NC STATE UNIVERSITY

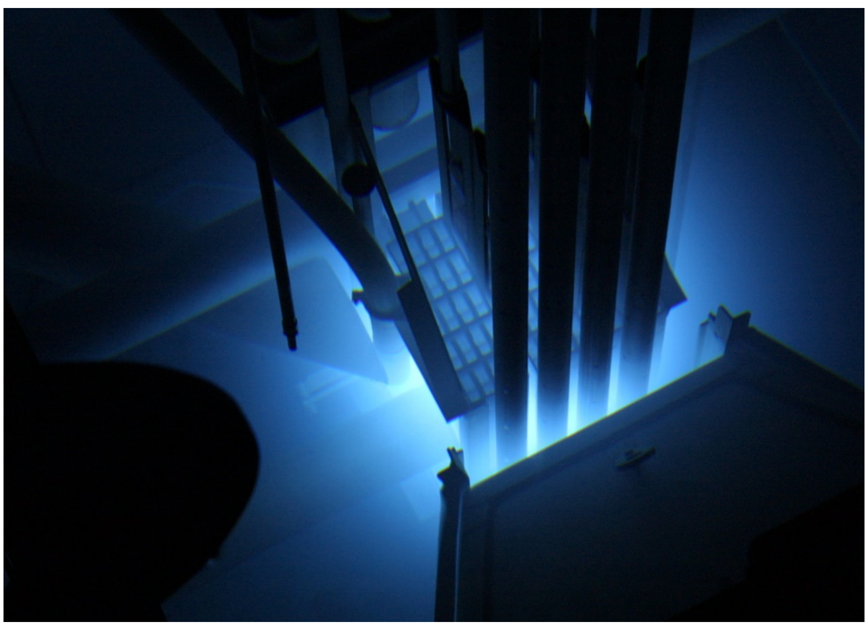
PULSTAR REACTOR

UPDATED

SAFETY ANALYSIS REPORT

NORTH CAROLINA STATE UNIVERSITY

RALEIGH, NORTH CAROLINA 27695



LICENSE NO. R-120

DOCKET NO. 50-297

9-AUGUST-2019

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1. THE FACILITY

1.1. Introduction

The PULSTAR nuclear research reactor facility is licensed to and operated by North Carolina State University (NCSU). NCSU is part of the University of North Carolina system, a public university system, and the largest four-year university in North Carolina. NCSU also has a large graduate school offering advanced degrees in many disciplines, including the sciences and engineering.

The mission of the University is to carry out teaching, research, and service functions. The reactor and experimental facilities at the PULSTAR reactor are used to fulfill this mission. Users of the facility include students and faculty at NCSU and other universities and researchers and clients from governmental agencies and industry.

NCSU and the reactor facility are located in Raleigh, North Carolina. The PULSTAR reactor is located on the North Campus at NCSU. Figure 1-1 indicates the reactor location on the NCSU campus. This central location is instrumental in fulfillment of the mission by promoting interaction of students and researchers with the facility.

The PULSTAR reactor was manufactured by the American Machine and Foundry Company (AMF) and its design, fabrication, and installation are based on the proven prototype that was located at the Buffalo Materials Research Center (BMRC) at the State University of New York at Buffalo.[1-1] It is a light water moderated and cooled thermal reactor using heterogeneous, pin-type, low enriched ^{235}U fuel. The PULSTAR reactor core design and operation were originally based on the testing results and experience from the BMRC. The PULSTAR reactor has operated at steady-state power levels up to one megawatt (1 MW) since initial criticality in 1972 to present. The BMRC was licensed for a steady-state power of 2 MW, but was fully decommissioned in 2016. The PULSTAR Reactor was originally designed to be pulsed to 2200 MW peak power and 38 MW·sec total energy release. Pulsing has since been discontinued and removed from the NCSU PULSTAR reactor license.

1.2. Summary and Conclusions on Principal Safety Considerations

The reliable operation of the NCSU PULSTAR since 1972 and the operational experience of the BMRC support the conclusion that no additional research or developmental testing is necessary to confirm proven safety features. The PULSTAR is operated in regions of proven technology and predictable responses.

Safety and design features of the PULSTAR reactor include the following:

1. Strong negative reactivity responses to reactor power increases. Refer to Sections 4 and 13.
2. A reactor design that takes advantage of proven safety characteristics of the zircaloy clad UO_2 fuel type. Specifically, the Doppler broadening, high retention rate of fission products by the fuel matrix, and low thermal conductivity. Refer to Sections 4 and 13.
3. A well-established intrinsic shutdown mechanism experienced at both the BMRC prototype and at the NCSU PULSTAR. Refer to Sections 4 and 13.
4. Heat capacity to accommodate fission product heating, even if the core is completely uncovered. Decay heat is not sufficient to cause fuel failure resulting

in the release of fission products. Refer to Section 13.

5. A building confinement system to control the release of radioactive materials, including fission products. Refer to Sections 6, 11 and 13.

6. A tall stack for dilution of radioactive materials, including fission products, and their subsequent release to the environment. Refer to Section 11 and 13.

7. Auxiliary power system and instrumentation to automatically perform safety related control and monitoring functions. Refer to Section 7 and 8.

As discussed in Section 13, the maximum hypothetical accident (MHA) associated with the PULSTAR reactor facility is one that results in radiation dose to personnel or members of the public. Accident scenarios that could result in radiation doses are:

1. Complete loss of coolant (LOCA) leading to the uncovering of the reactor core.

2. A fuel cladding failure accident (CFA) resulting in fission product release.

3. A fueled experiment failure resulting in the release of activation products, fission products, and the target fissionable material.

Fuel cladding failure results in the greatest potential for significant radiation doses to personnel and to the public, therefore it is considered to be the MHA. The analysis for this accident is detailed in Section 13.2.6 and it is concluded that the postulated 24 hour radiation dose is within 10 CFR Part 20 dose limits.

1.3. General Description of the Facility

1.3.1. Geographic Location

The reactor is located on the NCSU North campus in the Burlington Engineering Laboratory (BEL). The campus map is provided in Figure 1-1. Floor plans of the reactor building and BEL are provided in Figure 1-4 through Figure 1-9.

1.3.2. Principal Characteristics of the Site

The site is characterized by typical university campus surroundings, relatively light to moderate population areas, and a natural environment including no unusual characteristics. The area is free of any abnormal or hazardous conditions, e.g., explosive or chemical plants, etc. which would present a hazard to the reactor or reactor building. Experiments in other areas in the BEL and laboratories in nearby buildings that involve hazardous materials or processes are limited in scale and are reviewed for safety and regulatory compliance in accordance with NCSU policies and applicable regulations. There is no evidence that any undue risks to the public would result from site characteristics in the unlikely event of environmental release of radioactive materials from the reactor building.

1.3.3. Principal Design Criteria, Operating Characteristics, and Safety Systems

The NCSU PULSTAR reactor is designed to meet basic operational and safety practices. The criteria which represent the framework of reference and on which more detailed design may occur are as follows:

<u>Criterion 1</u>

Those features of reactor design essential to the prevention of accidents must be designed, fabricated, erected, and tested to establish quality and performance standards. This was accomplished by the selection of a tested and proven reactor system from a reliable vendor and the use of nuclear consultants to assist in the design of structures, systems and components.

Criterion 2

The maximum reactivity worth of control rods and the rate at which reactivity can be inserted are held to values such that no single credible malfunction could create a transient capable of causing fuel failure.

Criterion 3

Shutdown reactivity is provided for any credible operating condition even with the highest worth control rod fully withdrawn.

Criterion 4

Capability for control rod insertion under abnormal conditions is provided.

Criterion 5

The reactor facility has a single control room from which all actions can be controlled or monitored to ensure safe operation at all times. This room is provided with adequate means to ensure the protection of its occupants.

Criterion 6

A reliable reactor protection system is provided to automatically initiate appropriate action and prevent exceeding safety limits. Testing of these systems for continued operability is provided. Redundancy and independence of vital channels are provided.

Criterion 7

The confinement structure accommodates the environment for the largest credible energy release and site characteristics.

Criterion 8

Sufficient normal and auxiliary sources of electrical power have been provided to assure the capability for prompt shutdown of the reactor facility to a safe condition.

Criterion 9

Suitability of the facility and sub-systems will depend on demonstrated performance, reliability, and the extent of tests and inspections during the life of the plant. This has been confirmed throughout the construction, commissioning and operation of the NCSU PULSTAR.

Criterion 10

The fuel handling and storage facilities are designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel elements.

Criterion 11

The facility is provided with systems that are capable of monitoring the release of radioactivity under normal and accident conditions.

The reactor was selected and designed to take advantage of proven characteristics of a nuclear power reactor type fuel, specifically the Doppler broadening, high retention rate of fission products by the matrix, and low thermal conductivity. The design and operating limits are consistent with the previously licensed and proven safety limits of the BMRC. Neither the reactor nor its support systems require any research or development to demonstrate continued inherent safety of design and operation.

The intrinsic shutdown mechanism of the PULSTAR type reactor is well established by experience at both the BMRC and the NCSU PULSTAR. The reactor building use of confinement continues to be an acceptable and successful substitution for containment. Therefore, the system (structure) required to confine the maximum hypothetical accident (MHA) as detailed in Section 13 has been modeled after proven trends in reactor building envelopes, specifically confinement. The reactor building is a reinforced, monolithic concrete structure faced with a brick veneer and has proven to be a relatively air-tight structure. The primary piping vault (PPV) is part of the reactor building that houses primary piping and the delay tanks. The PPV is an underground reinforced concrete structure with minimal penetrations which is coupled to the reactor building. The reactor building and PPV are designed to confine radioactive releases and at the same time provide for controlled release to the surrounding environment at levels below regulatory limits. The remainder of the BEL is comprised of administrative offices and laboratories, which are conventional structures for their particular functions.

Table 1-1 summarizes the existing characteristics of the reactor core. The PULSTAR reactor is licensed for a maximum of twenty-five fuel assemblies of 4% and/or 6% enrichment. Graphite and/or beryllium reflectors may be located around the periphery of the core to enhance core performance.

1.3.4. Engineered Safety Features

Engineered safety features (ESF) are used to reduce the potential radiation dose to occupationally exposed personnel and members of the general public to levels less than the regulatory limits. The major step in accomplishment of this objective was the selection of a proven reactor type and support systems. The engineered safety features limit the consequences of a credible but most improbable event, if it should occur. This is accomplished by automatically isolating the reactor building and therefore controlling the release of activity. Section 6 describes the engineered safety features in detail.

1.3.5. Instrumentation, Control and Electrical Systems

The instrumentation for the NCSU PULSTAR reactor includes both nuclear and non-nuclear channels using electronic signals. Also included is the scram logic unit and associated trip circuits that make up the Reactor Safety System. A combination of alarms, interlocks, drive inhibits and reverse drive functions are provided for the safe and efficient operation of the reactor. Trips are fail-safe, meaning that upon loss of electrical power to an instrumentation channel, all trip circuits associated with that channel will act to limit reactor power or initiate reactor shutdown.

The design of the PULSTAR reactor is such that the reactor can be shut down and safely maintained in a shutdown condition under a complete loss of electrical power. There are no electrical power supplies that are critical for maintaining the facility in a safe shutdown condition, even for extended periods of time. The electrical power for Burlington Engineering Laboratories is supplied from the university distribution system. In the event that commercial power is lost, emergency lighting is supplied by backup batteries and selected radiation monitors and reactor instrumentation are supplied by uninterruptible power supplies (UPS). The PULSTAR reactor is equipped with an auxiliary electric generator to assist with post shutdown monitoring and ventilation in the event that commercial power is lost. Section 7 and Section 8 describe these systems in detail.

1.3.6. Reactor Coolant and other Auxiliary Systems

The reactor primary coolant system, utilizing demineralized water, is designed to remove up to two megawatts (2 MW) of heat from the PULSTAR reactor operating in the steady-state mode with forced convection cooling and has sufficient capacity to operate at power levels up to 100 kW with natural convection cooling. The heat from the primary coolant system is transferred to the secondary cooling system via a heat exchanger which in turn dissipates the heat through standard induced draft, counter-flow cooling tower located on the east side of Burlington Engineering Laboratories.

The reactor also has numerous auxiliary support systems such as HVAC systems, water purification systems, and overhead hoists. Section 5 and Section 9 describe these systems in detail.

1.3.7. Radioactive Waste Management Systems and Radiation Protection

The PULSTAR Radiation Protection Program at the reactor facility is approved by the University Radiation Safety Committee. This program meets the requirements given in 10 CFR Part 20 – *Standards for Protection against Radiation* and emphasizes the ALARA philosophy in all areas of operation that affect radiation safety at the reactor facility.

The amount of radioactive waste from the use of a reactor facility for teaching and research purposes is minimal compared to a test reactor, nuclear power reactor or fuel processing facility. The PULSTAR reactor contains systems designed to collect the various liquid, solid and gaseous waste streams where they are monitored and prepared for disposal in accordance with 10 CFR Part 20, 10 CFR Part 61 – *Licensing Requirements for Land Disposal of Radioactive Waste* and other applicable regulations. Section 5, Section 6 and Section 11 describe these systems in detail.

Spent reactor fuel is returned to the US Department of Energy as discussed in Section 1.7.

1.3.8. Experimental Facilities and Capabilities

The PULSTAR reactor has several experimental facilities which are available for use by educators and researchers from academic institutions, government, and industry. Each of these facilities harness the radiation fields emanating from the reactor core creating unique educational, research, and testing capabilities. Experimental facilities are located in the reactor pool and beamtubes. Experimental applications include material testing and analysis by various instruments (e.g. neutron diffraction, positron lifetime, neutron radiography, etc.), evaluation of radiation effects on materials, neutron activation analysis, radioisotope production, isotope doping and nuclear instrumentation testing. In addition, the reactor is used for educating and training users about nuclear reactor operation, nuclear parameters and nuclear engineering principles. Distance education has occurred allowing students from throughout the world to observe and participate in reactor operations and experiments.

Section 10 provides details on the experimental and teaching programs. All experimental facilities are reviewed and approved following internal procedures prior to being installed. All experiments using the experimental facilities and training activities are arranged in advance and reviewed and approved prior to being conducted.

1.4. Shared Facilities and Equipment

The reactor building structure is adjacent and connected to the surrounding Burlington Engineering Laboratory.

There are numerous systems used in the PULSTAR reactor, most of which are dedicated to the facility and not shared. Details on these systems are provided in Section 5 for coolant systems, Section 6 for engineered safety systems, Section 7 for instrumentation and control systems, Section 8 for electrical systems, Section 9 for auxiliary systems (e.g. ventilation, compressed air, etc.), Section 10 for experiments, and Section 11 for radiation monitoring and radioactive waste systems.

The following shared systems are present within the reactor building:

1. Electrical supply – Burlington Engineering Laboratories (BEL) receives its power from the campus distribution system. All feeder circuits in BEL originate from the switchgear in room 1106A. The switchgear provides a 600 ampere feeder to reactor building switchboard panel SB-1 and a 400 ampere feeder to reactor building motor control center M (MCC-M). The switchgear can be used to de-energize electrical power to all reactor equipment in the event of fire or other emergency conditions. Refer to Section 8 for more details on the electrical supply.

2. Natural gas supply – The natural gas supply enters BEL on the west side where it branches off to supply various loads throughout the laboratories, mainly fume hoods. The auxiliary generator is the only reactor equipment that relies on natural gas, refer to Section 8.2.

3. Domestic water supply – The domestic water supply enters BEL from the west side through room B113. The reactor building is supplied from a service branch line in room B103. The water supplies sinks, primary water makeup, and secondary water makeup.

4. Liquid waste water system – All drains in the reactor building, along with select drains from BEL collect in the sump located in the mechanical equipment room. The waste water can then be processed for disposal. Refer to Section 5.7.1.

5. Hot water for space heating – Hot water for space heating is provided by the campus steam distribution system.

6. Chilled water for space cooling – Chilled water for space cooling is supplied by the campus chilled water system.

7. Chilled water for experiment cooling requirements – Chilled water for experiment cooling is provided by the campus chilled water system. Cooling for the reactor is provided by the completely independent secondary cooling system. Refer to Section 5.3.

8. Roof drain system and associated storm sewer system – The roof drains for the reactor building are independent of the rest of the BEL, but do tie into the campus storm sewer system located on the southwest side of the building.

9. Fire Detection System – The reactor building fire detection system is part of the BEL fire detection system but is an independent loop. The reactor building does not have any sprinklers but there are fire extinguishers available throughout the building. There is no significant amount of combustible material present in the

reactor building. The building is regularly inspected by the NCSU Fire and Life Safety Office. Refer to Section 9.3.

1.5. Comparison with Similar Facilities

The PULSTAR reactor facility at NCSU is most comparable to the BMRC that was licensed to operate at a steady-state power of 2 MW. The BMRC was licensed from 1964 to 1994, and was subsequently fully decommissioned in 2016. Variations or differences from the BMRC are conservative and were made to support the functional needs of NCSU and AMF product improvements rather than any major technological or safety limitation of the BMRC.

The PULSTAR reactor core design and operation were originally based on the testing results and experience at the BMRC. The reliable operation of the NCSU PULSTAR since 1972 supports the conclusion that no additional research or developmental testing is necessary to confirm proven safety features, and that the reactor is operated in regions of proven technology and predictable responses.

1.6. Summary of Operations

The mission of the University is to carry out teaching, research, and service functions. The reactor and experimental facilities are used to fulfill this mission.

Organizationally, the PULSTAR reactor is operated by the Nuclear Reactor Program (NRP) within the Department of Nuclear Engineering. The Department of Nuclear Engineering is within the College of Engineering. The NRP has a Director and a dedicated staff similar to that employed at other university reactor facilities and is consistent with ANSI/ANS 15.1-2007 – *The Development of Technical Specifications for Research Reactors.*[1-2] The NRP is responsible for the safe and efficient operation of the PULSTAR reactor and its support facilities.

In addition, operational activities at the reactor are reviewed and approved by safety committees at NCSU. The Radiation Safety Committee (RSC) has the primary responsibility to ensure that the use of radioactive materials and radiation producing devices, including the nuclear reactor, at NCSU are in compliance with state and federal licenses and all applicable regulations. The RSC reviews and approves all license changes and experiments involving the use and potential release of radioactive material conducted at NCSU. The RSC also provides oversight of the NCSU Radiation Protection Program and is informed of the actions of the Reactor Safety and Audit Committee (RSAC) and may require additional actions by the RSAC and the Nuclear Reactor Program. RSAC has the primary responsibility to ensure that the reactor is operated and used in compliance with the facility license and all applicable regulations. RSAC reviews and approves license changes, experiments, procedures, and design changes made to the reactor facility. RSAC also performs an annual audit of the operations and performance of the NRP. The annual audit report, including any recommendations, is provided to the RSC. Occupational health and safety aspects of reactor operations are reviewed and approved by NCSU Environmental Health and Safety. Further details on facility organization and operations are given in Section 12.

The PULSTAR reactor currently operates up to the licensed steady-state power level of 1 MW and upon approval of this license renewal, will operate at steady-state levels up to 2 MW. Operation of the facility occurs on an as needed basis in support of the mission. This is typically accomplished during routine business hours. Extended hours beyond the normal working hours and continuous operation over extended periods of time have occurred when there is a need. Prolonged operation at the full licensed power level may occur when needed and has periodically occurred in the past.

Operation of the PULSTAR reactor produces radiation and radioactive materials. In addition to the fission product production in the reactor fuel, other radioactive by-products from activation of impurities in the primary coolant and experimental facilities are produced. Occupational radiation doses are taken into consideration for reactor experiments and operations and have been well below regulatory limits. Radioactive effluent is also considered in reactor experiments and operations and has also been below regulatory limits. Various methods are used to prevent, reduce, and mitigate radiation dose and production of radioactive materials. Radiation and radioactivity are closely monitored as described in Section 11. Limitations on radiation and radioactivity are also described in Section 11.

1.7. Compliance with the Nuclear Waste Policy Act of 1982

The US Department of Energy (DOE) provides fresh fuel to, and receives spent nuclear fuel from, university research reactors. DOE has informed the US Nuclear Regulatory Commission (NRC) in a letter dated May 3, 1983 from Robert Morgan (DOE) to Harold Denton (NRC) that universities operating non-power reactors have entered into contracts with DOE that provide that DOE retain title to the fuel and be obligated to accept spent nuclear fuel (SNF) and/or high level waste for storage or reprocessing.[1-3]

DOE Research Reactor Infrastructure program contract number 78287 applies to the NCSU PULSTAR reactor. Because NCSU has entered into a contract with DOE, the applicable requirements of the Nuclear Waste Policy Act of 1982 have been satisfied.

1.8. Facility Modifications and History

Construction permit CPRR-106 was issued for the construction of the NCSU PULSTAR Reactor.[1-4] Construction began in June 1969 and was completed in September 1972.

1.8.1. Reactor Designer

The PULSTAR reactor was manufactured by the American Machine and Foundry Company (AMF) and its design, fabrication, and installation were based on the proven PULSTAR reactor located at the BMRC at the State University of New York at Buffalo. AMF served as the checkout supervisor of the NCSU PULSTAR Reactor. AMF also furnished the fuel.

1.8.2. Architect Engineer

Charles W. Wheatley and Associates were the architects for the BEL complex and were supported by the nuclear consulting firm of Parson-Jurden Company of New York City. Parsons-Jurden was responsible for the design and initial review of the reactor support systems. Detailed design of the waste handling, radiation monitoring and process systems outside the reactor was handled by the architect. The Title III services (field supervision, inspection, and acceptance test of all but the AMF design) were handled by the architect and their subcontracted engineers. The architect appointed a qualified individual to serve as the full time on-site inspection coordinator responsible for the scheduling and monitoring of inspection services by the design engineers, giving reasonable assurance to the University and NRC that compliance by the contractors with the contract documents was satisfied, and ensuring that adequate records were available to substantiate the effectiveness of the quality control and assurance programs.

1.8.3. NCSU Consultants and Advisors

NCSU established an Ad Hoc Building Planning Committee to manage the overall planning for delivery of the PULSTAR reactor and BEL complex. The technical management of the project was handled by the Department of Nuclear Engineering. The faculty of the College of Engineering actively served as advisors to the NE project engineer. The NE Committee was responsible for evaluating and approving the safety analysis report (SAR). Also within the NE Department were experts in instrumentation and controls, health physics, and experiment planning. The chairman of the former Radiation Protection Council (now the Radiation Safety Committee, RSC) appointed a subcommittee to review the SAR.

NCSU defined and implemented a Quality Assurance Program (QAP)[1-5] with a quality assurance coordinator (QAC) who directed/coordinated all quality assurance measures for the project, both on-site and off-site, as they related to the nuclear safety and facility operational aspects. The three levels of quality assurance were:

1. control by contractor
2. surveillance by design engineers
3. audit by the QAC

1.8.4. License Amendments

Since 1972, graphite and beryllium reflectors have been added to the periphery of the reactor core to allow continued use of the reactor fuel initially received. Also, various instrumentation systems have been replaced with newer components. These include the radiation monitoring system, nuclear instrumentation channels, temperature measuring devices, scram logic unit, and data recorders.

The R-120 license renewal application was submitted in 1989 and approved in 1997. Under the first license renewal, no significant deviations from the proven design and operation were made. The major changes were: solid graphite reflectors had been added to the core periphery, pulsing has been discontinued, various instrumentation channels have been updated, and subterranean primary piping vault was added to contain the previously buried delay tank and primary piping. The PPV construction was completed in 1995.

The following amendments to the license have been made since the last license renewal in 1997:

- Amendment 12 – Revision to the Technical Specifications – October 30, 1997
    - Update of Technical Specification Figure 5.2-1, Reactor Site Map to reflect changes to street names.
    - Update Technical Specification 6.7, Reporting Requirements from Region II to the Office of Nuclear Reactor Regulation.
- Amendment 13 – Revision to the Technical Specifications – March 1, 1999
    - Allow for the use of beryllium as a reflector material on the periphery of the reactor core.
    - Removes the limitation from the Technical Specifications requiring fuel assemblies to be arranged in the reactor core in a five-by-five array.
- Amendment 14 – Revision to the Technical Specifications – May 29, 2001
    - Changes the name of the Radiation Protection Committee to the Radiation Safety Committee.

  - Changes the administrative requirements for the Radiation Safety Committee.
- Amendment 15 – Revision to the Technical Specifications – March 3, 2005
  - Removes reference to the service hatch entrance in the primary piping vault.
  - Removes the prohibition of cryogenic liquids inside the biological shield.
  - Changes the annual reporting period from the academic year to the calendar year.
- Amendment 16 – Revision to the Facility License – August 15, 2006
  - Permits the receipt, possession, and use of byproduct material.
- Amendment 17 – Revision to the Technical Specifications – September 8, 2008
  - Revised the Technical Specifications in its entirety to make it more consistent with ANSI/ANS-15.1-1990.
  - Changes to Technical Specification 3.5 regarding radiation monitor setpoints
  - Changes to Technical Specification 3.8 regarding fueled experiments
  - Changes to Technical Specification Section 6 regarding the facility line organization.
- Amendment 18 – Revision to the Facility License and Technical Specifications – June 27, 2016
  - Allows for the operation of the PULSTAR reactor with mixed enrichment cores containing 4% and 6% enriched fuel assemblies.

1.8.5. 10 CFR 50.59 Modifications

The following major modifications have been made to the facility under 10 CFR Part 50.59 since the last license renewal in 1997:

In 2006, the ventilation system for the facility was replaced with newer components and relocated from the mechanical equipment room (MER) to an area in the reactor building above the control room now referred to as the ventilation equipment room (VER). This change was made to accommodate experimental needs for space and to add air conditioning, temperature and humidity control, of the reactor building to provide a more stable environment for experimental equipment. The new system was designed by mechanical engineering firm Dewberry and Associates while NCSU Facilities Design and Construction Services provided oversight for these modifications. All work was performed by qualified contractors.

In 2006, the electrical distribution system was upgraded with an increase in capacity. NCSU Facilities Design and Construction Services provided both design and oversight for these modifications while the work was performed by qualified contractors.

In 2013, the primary and secondary coolant systems were upgraded to support future operations at an increased steady-state power of 2 MW. The designer was Enercon Corporation of Kennesaw, Georgia with support provided by Edmondson Engineering of Durham, North Carolina. NCSU Facilities Design and Construction Services provided oversight for these modifications. The work was performed by qualified contractors.

All changes to the facility have been made in accordance with applicable regulations, license conditions, and accepted engineering and building practices. Changes to the facility are documented and reviewed and approved by the RSAC and RSC, as necessary. These include changes to procedures, facility design, experiments and experimental facilities, and license documents. License amendments have been made and approved by the NRC prior to being implemented as necessary.

1.9. References

1-1 Buffalo Materials Research Center, *Hazards Summary Report Revision II*, September 23, 1963.

1-2 ANS/ANSI-15.1-2007-*The Development of Technical Specifications for Research Reactors.*

1-3 NUREG 1537 Part 2, *Guideline for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors-Standard Review Plan and Acceptance Criteria*, February 1996, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1537/.

1-4 North Carolina State University at Raleigh Docket No. 50-297 Construction Permit No. CPRR-106, October 1, 1968.

1-5 North Carolina State University-Nuclear Science and Engineering Research Center, *Quality Assurance Instructions Manual*, July 1, 1969.

1-6 Nuclear Regulatory Commission, *Issuance of Amendment 18 to Allow for Mixed Enrichment Core Configurations (TAC NO.MF6088),* June 27, 2016, NRC Accession Number: ML16146A785, https://www.nrc.gov/docs/ML1614/ML16146A785.pdf

Table 1-1 – Summary of the NCSU PULSTAR Reactor Characteristics

Core Dimensions	
Overall (in)	$15^7/_8 \times 15 \times 24$
Minimum Critical No. of Assemblies (4%)	21
Maximum Assemblies per Core	25
Fuel	
Material	UO_2
Form	Sintered Pellets
Enrichment (w% ^{235}U)	4% / 6%
Design Inventory Core (kg UO_2)	[redacted]
^{235}U per Fuel Pin (gm)	[redacted]
Fuel Pin	
Clad Material	Zr-2
Pin diameter (in)	[redacted]
Pellet Diameter (in)	[redacted]
Height of Pellet (in)	[redacted]
Height of Pellet Stack (in)	[redacted]
Pellets per pin	[redacted]
Fuel Box	
Material	Zr-2 or Zr-4
Outer Dimensions (in)	2.74 × 3.15 × 38
Fuel Pins per Assembly	25
Weight (lbs)	44
Moderator – Coolant	
Material	Light Water
Nominal Inlet Temp (°F)	105
Nominal Outlet Temp (°F)	118.8
Primary Flow Rate (gpm)	1000
Secondary Flow Rate (gpm)	1000
Reflector Material	Light Water, Graphite, Beryllium
Control Rods	
Absorber Material	Ag-In-Cd (80-15-5)
Guide Material	Aluminum
Shape	Rectangular
Clad Material	Sn/Ni
Number of Control Rods	4

Figure 1-1 – North Carolina State University North Campus Map

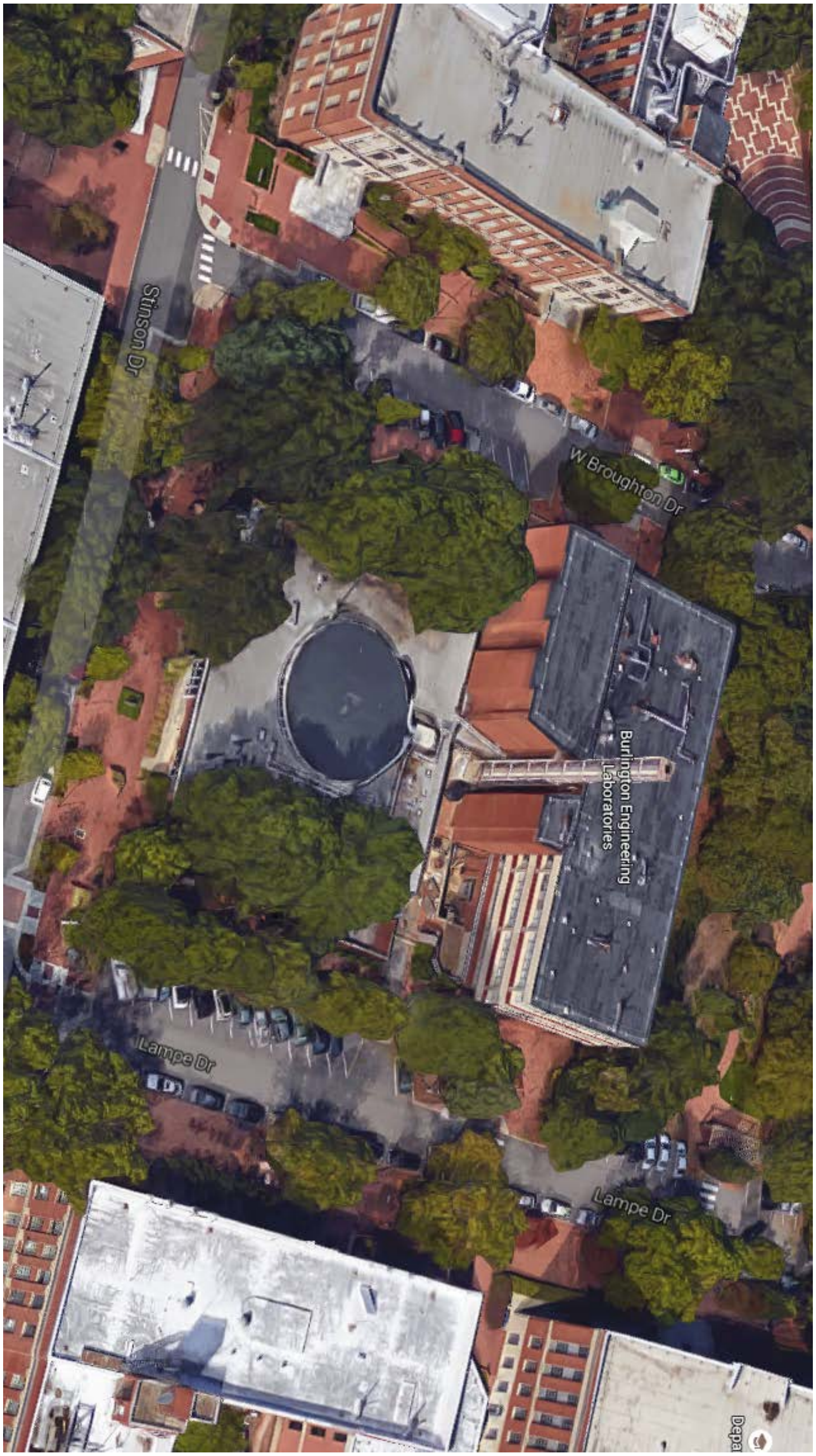

Figure 1-2 – Burlington Engineering Laboratory – Aerial View

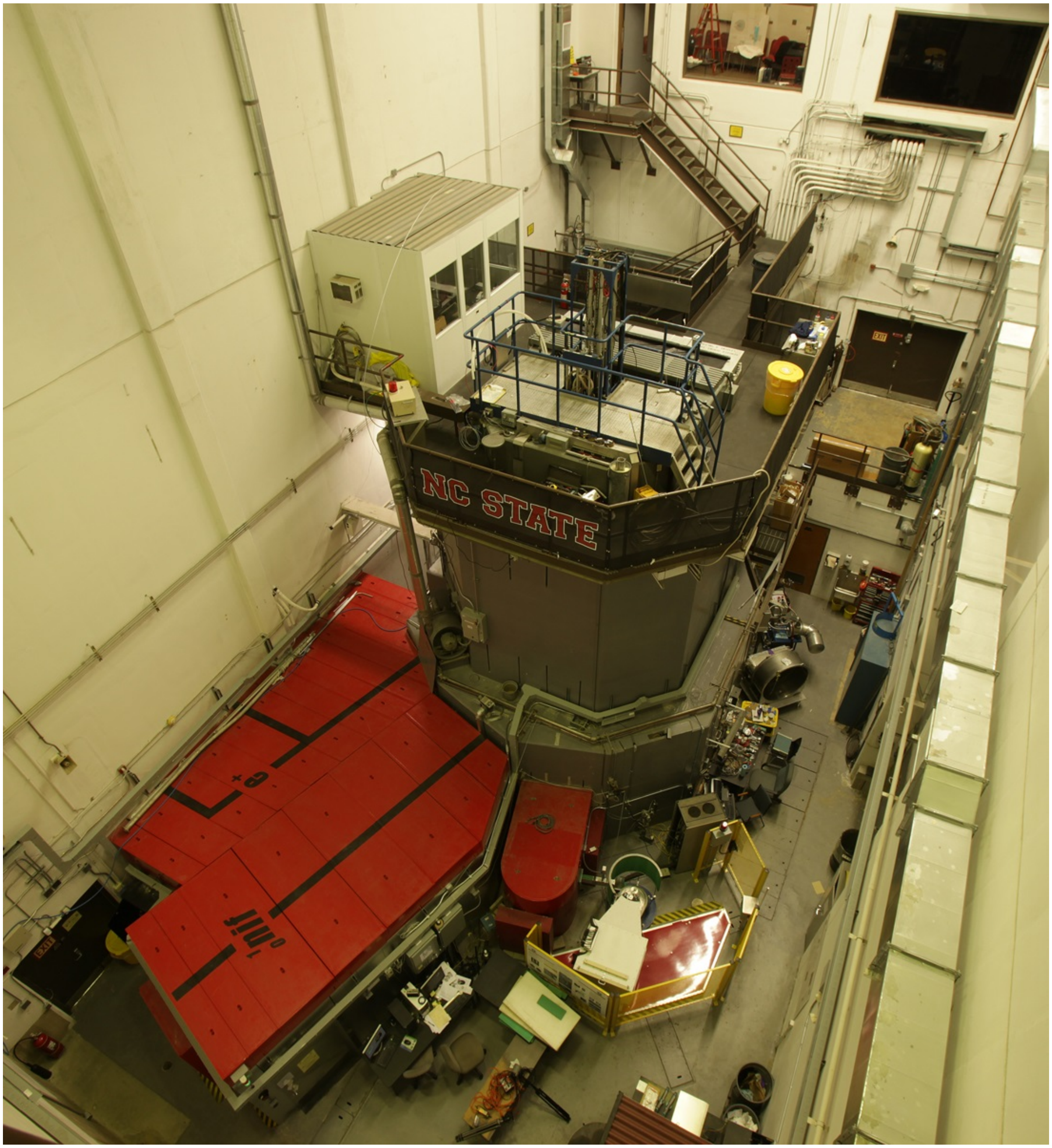

Figure 1-3 – Reactor Bay – Projection View

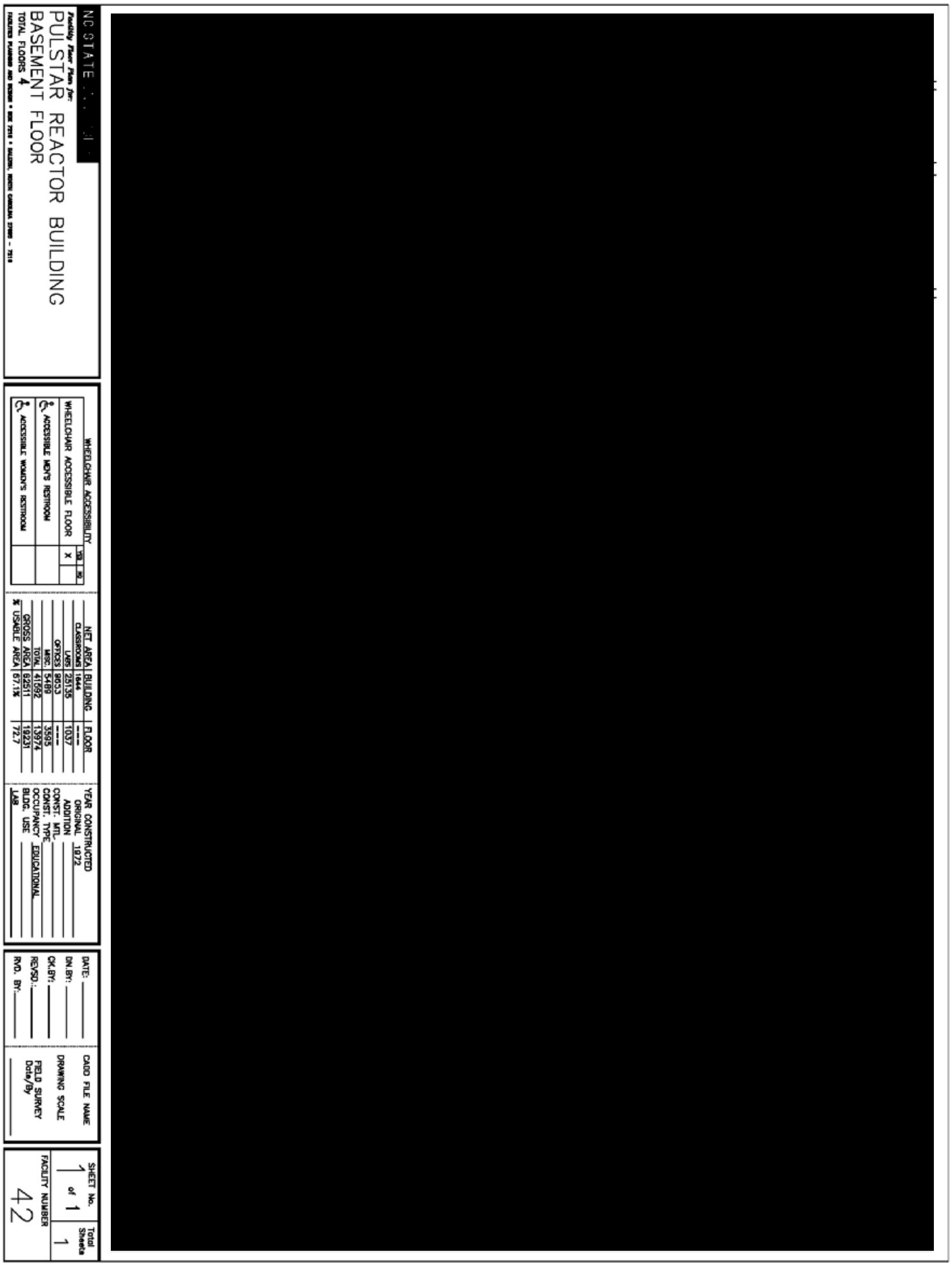

Figure 1-4 – Reactor Building Floor Plan – Basement Floor

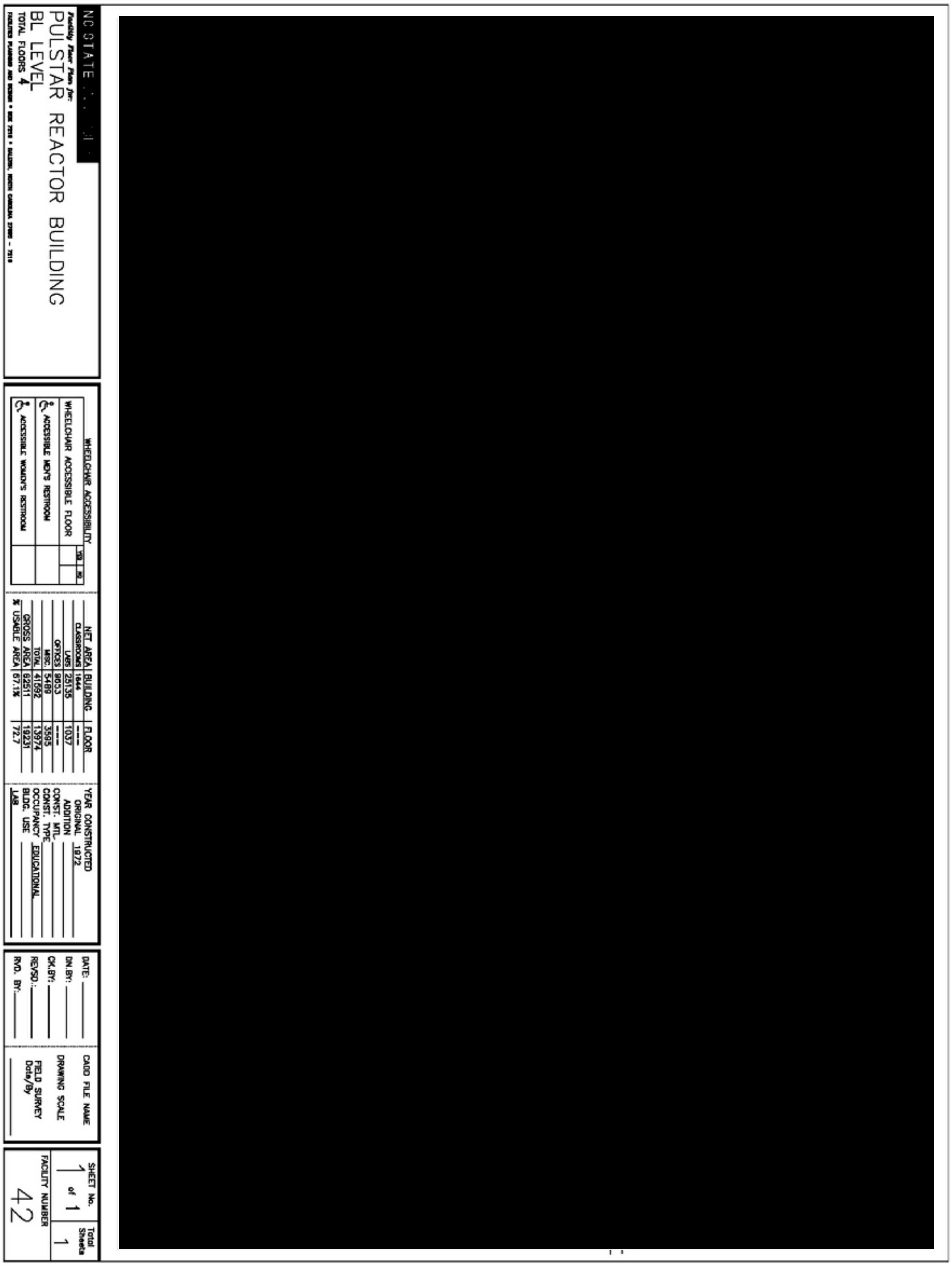

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Figure 1-5 – Reactor Building – BL Floor

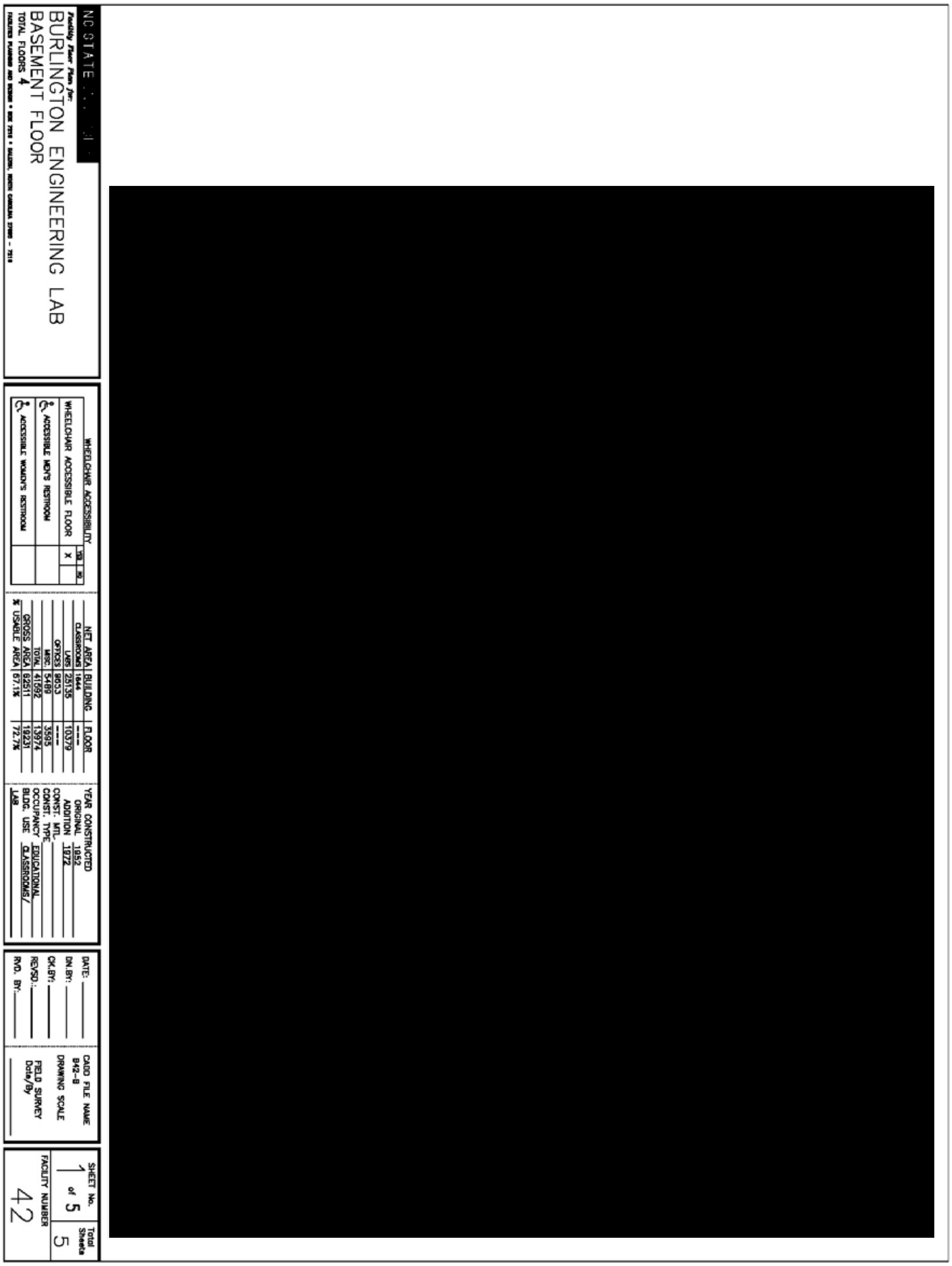

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Figure 1-6 – Burlington Engineering Labs – Basement Floor

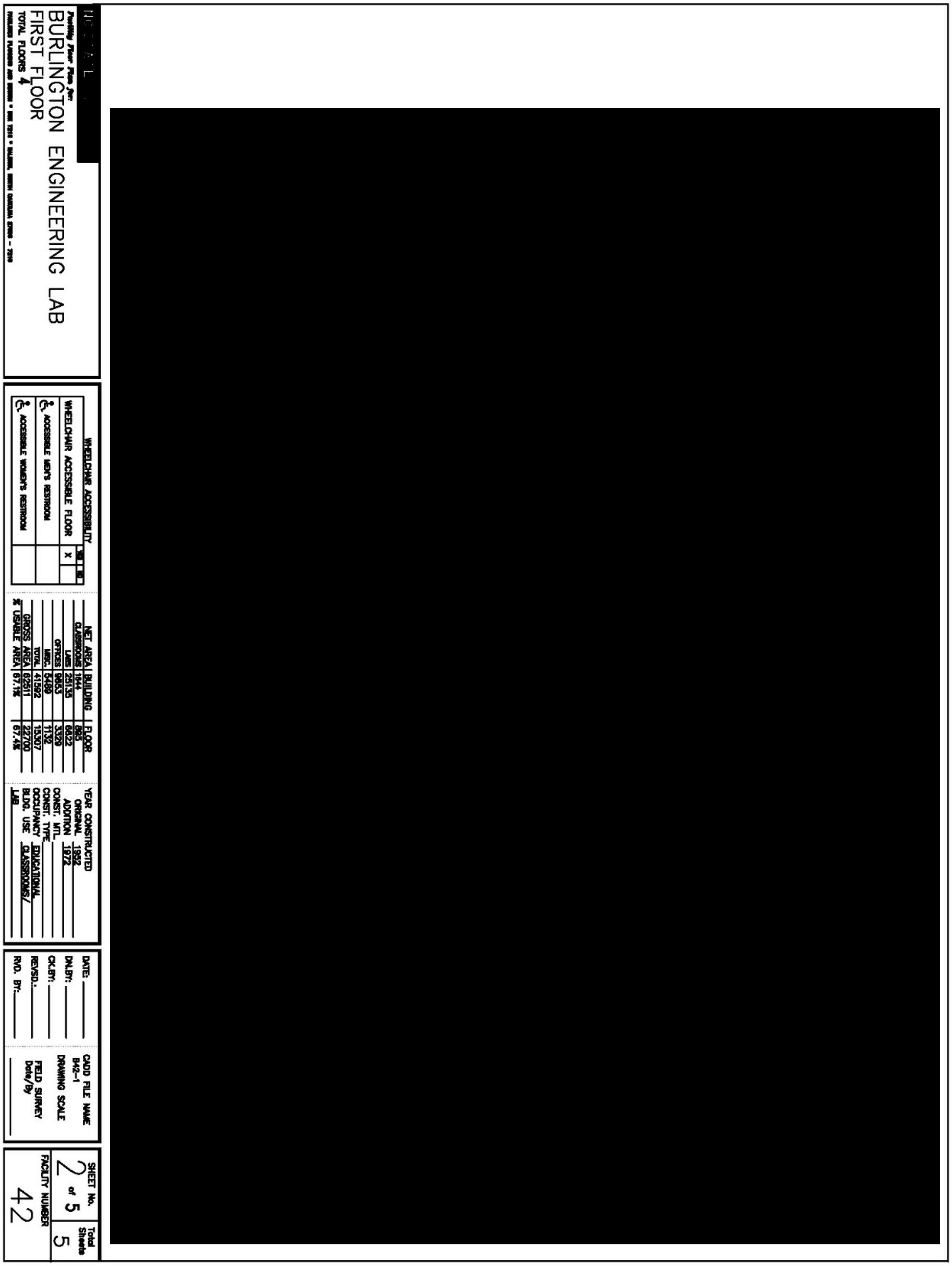

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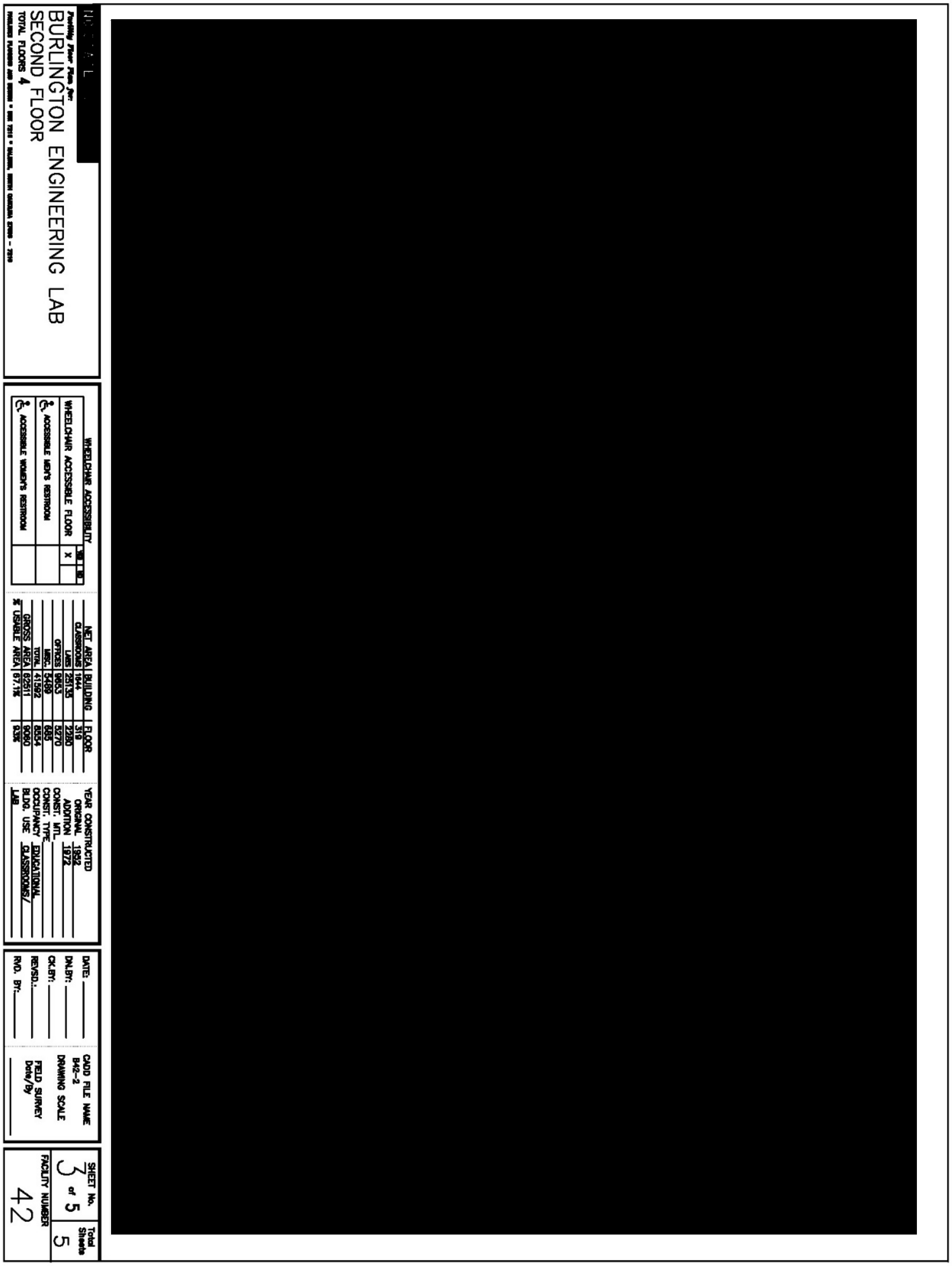
Figure 1-7 – Burlington Engineering Labs – First Floor



[redacted]

Figure 1-8 – Burlington Engineering Labs – Second Floor

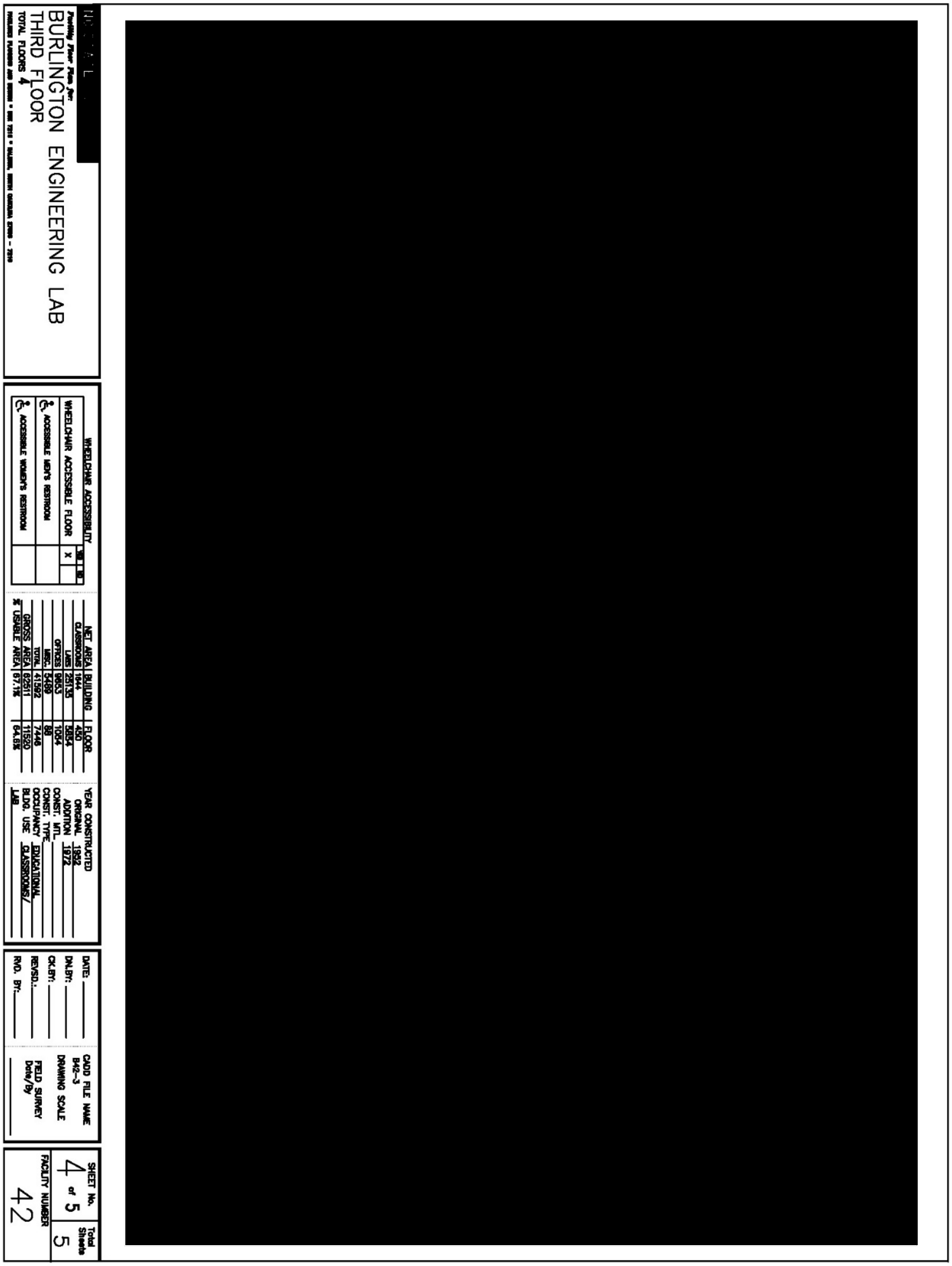

[redacted]

Figure 1-9 – Burlington Engineering Labs – Third Floor

2. SITE CHARACTERISTICS

Pertinent information related to the NCSU PULSTAR reactor site, including demographics, land use, and meteorological, hydrological, geological, and seismological factors are presented in this section.

2.1. Geography and Demography

2.1.1. Site Location and Description

2.1.1.1. Specification and Location

The North Carolina State University (NCSU) PULSTAR reactor is located in a separate building within the Burlington Engineering Laboratories on the North campus. The University campus is located in the western part of the city of Raleigh near the center of Wake County, North Carolina. Latitude and longitude are [redacted] latitude, [redacted] longitude. Universal transverse Mercator coordinates are [redacted]. The reactor building is located in a quadrangle surrounded by streets on all sides. Relative positions of the nearby campus buildings and the railroad line are found on maps of the surrounding area. Areas within 8km of the PULSTAR facility are provided. Figures also display highways and railways within close proximity to the reactor.[2-1]

2.1.1.2. Boundary and Zone Area Maps

The reactor site is shown on provided figures indicating the site boundary and operations boundary. The nearest building and exhaust stacks of equivalent height to the reactor stack are located approximately 150 m away (DH Hill Library and Yarborough Steam Plant). The nearest agricultural area is located approximately 6 km (~ 4 miles) away. The surrounding areas would have little effect on dispersion as there is minimal elevation change. For this facility it is noted that the site boundary is the emergency preparedness zone. The reactor floor is situated at [redacted]. The ground slopes away from the reactor site to the east, south, and west and affords good natural drainage and freedom from flooding. The terrain of the area inside a radius of more than 15 miles around the reactor site is made up of gently rolling land ranging in altitude from about 350 feet to 450 feet. Except for a few small lakes, the land is well drained; creeks carry the surface water to rivers that flow toward the southeastern part of the state.[2-1,2-2,2-3]

Figures are provided which depict the following:

1. Reactor site and NCSU campus[2-1]
2. City of Raleigh and Wake County, North Carolina[2-4]
3. Topographic details[2-1]
4. Flood zones and flood prone soils[2-1]
5. Storm sewer system[2-1]
6. Zoning[2-4]
7. Current and planned uses[2-4,2-5]
8. Agriculture and forests use[2-6,2-7]

Note: In 2016, Harrelson Hall (shown in Figures 2-2, 2-6, and 2-9) was demolished and replaced with a landscaped park. Figures 2-3 and 2.4 have been updated.

From these figures it is concluded that the reactor facility is located away from major highways, railways, and airports and surrounded by predominantly residential, educational, governmental, and retail use areas.

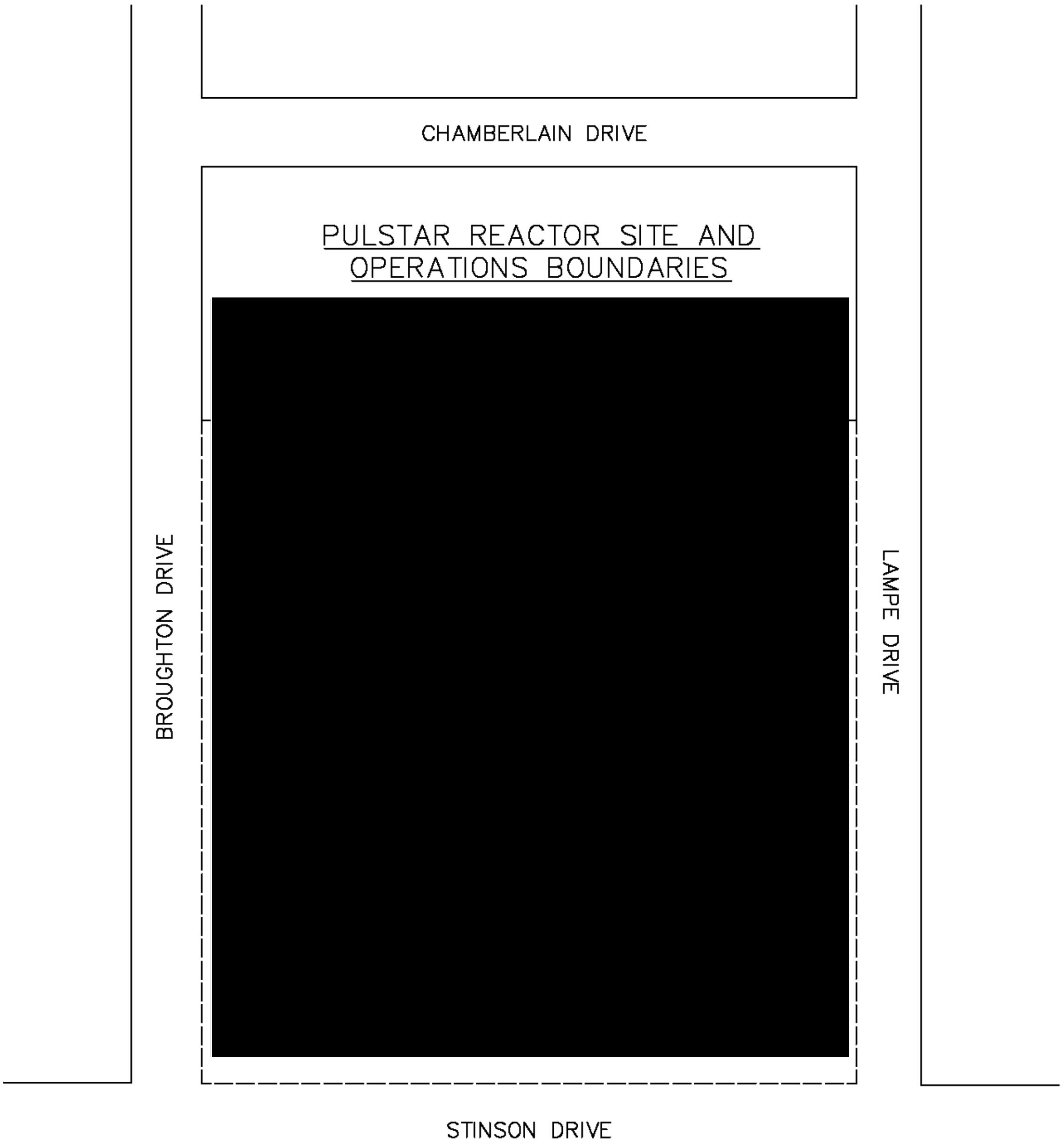


Figure 2-1 – Reactor Site

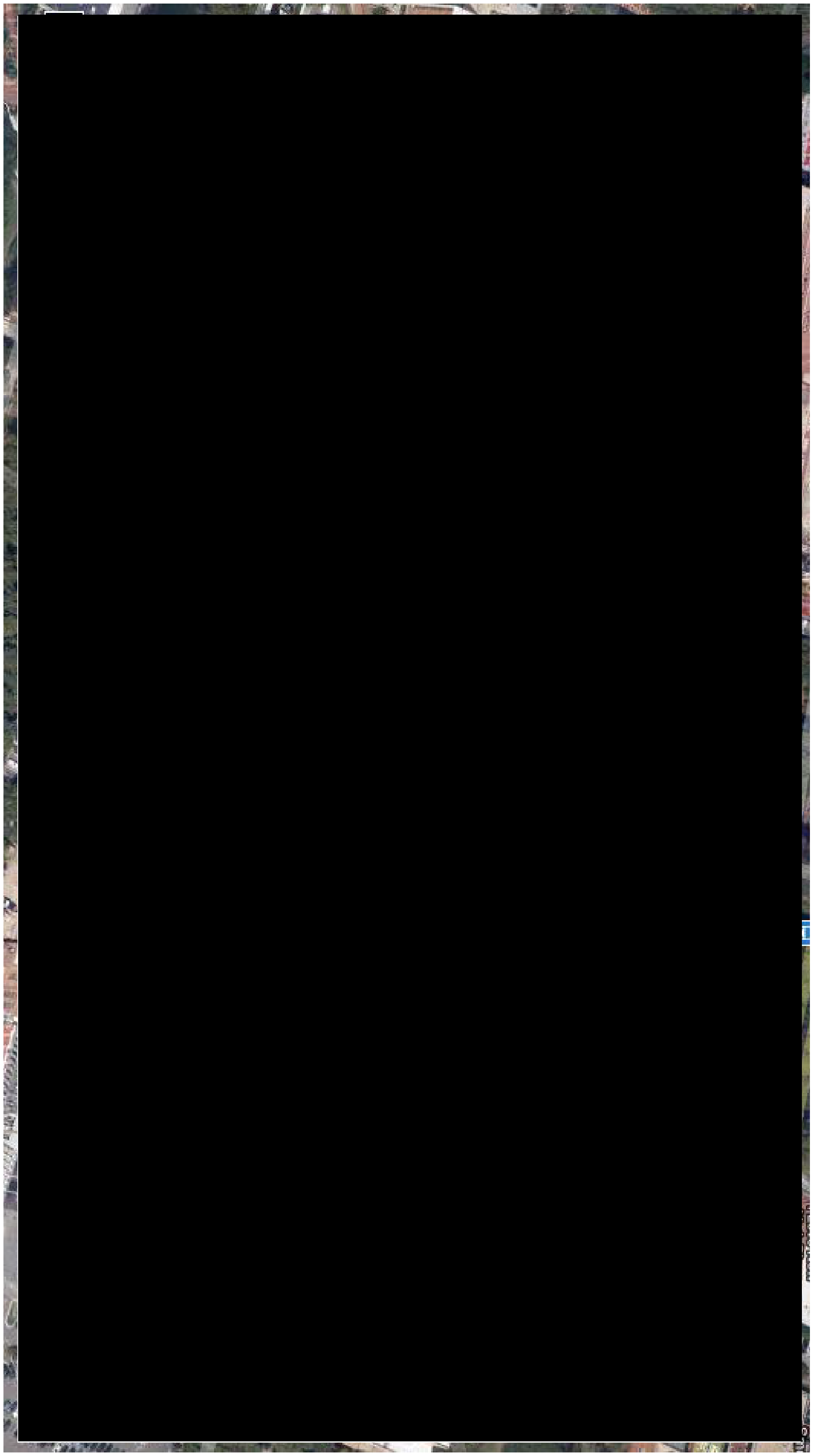


Figure 2-2 – Aerial View of the Reactor Building

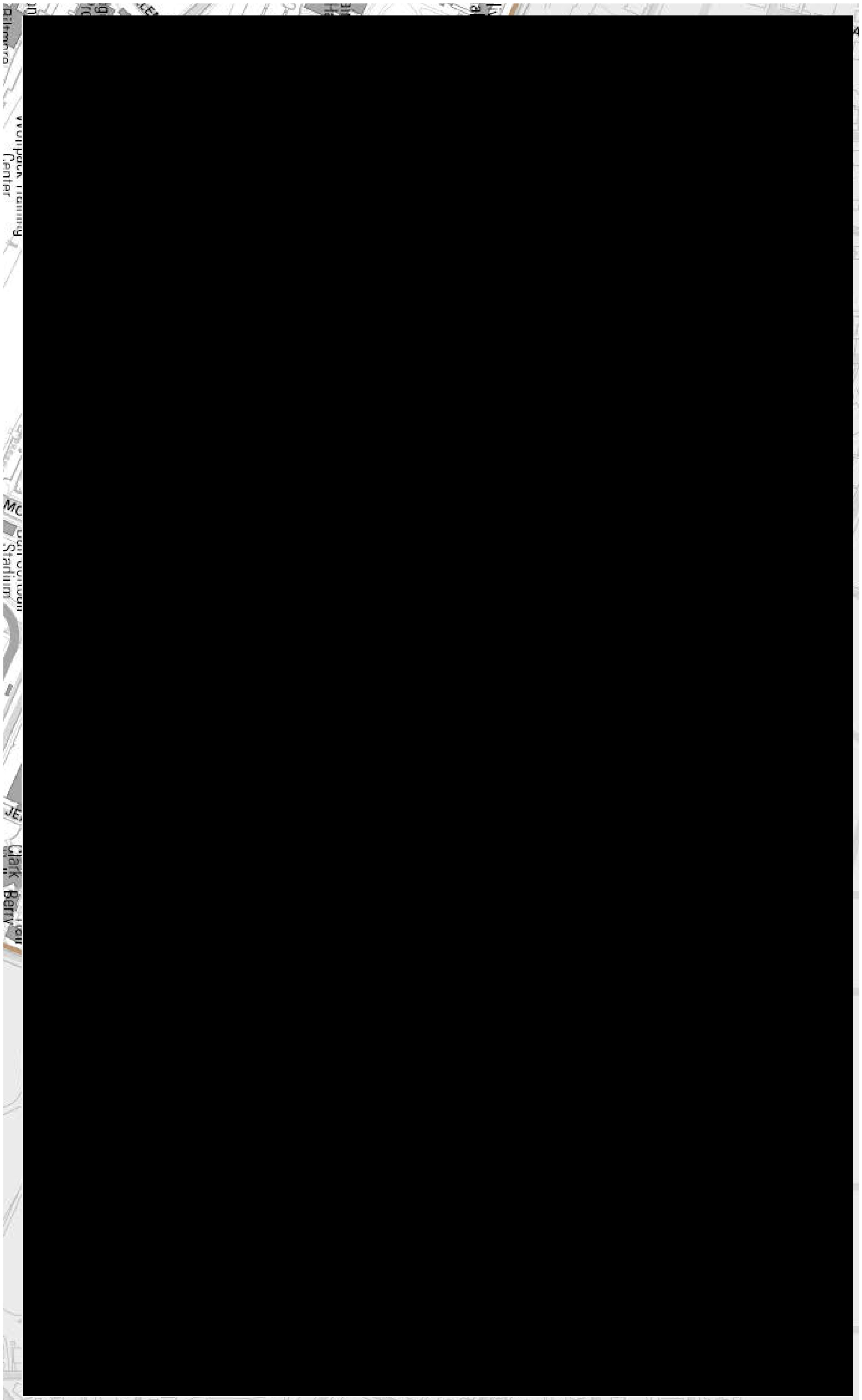

[redacted]

Figure 2-3 – North Carolina State University North Campus Map

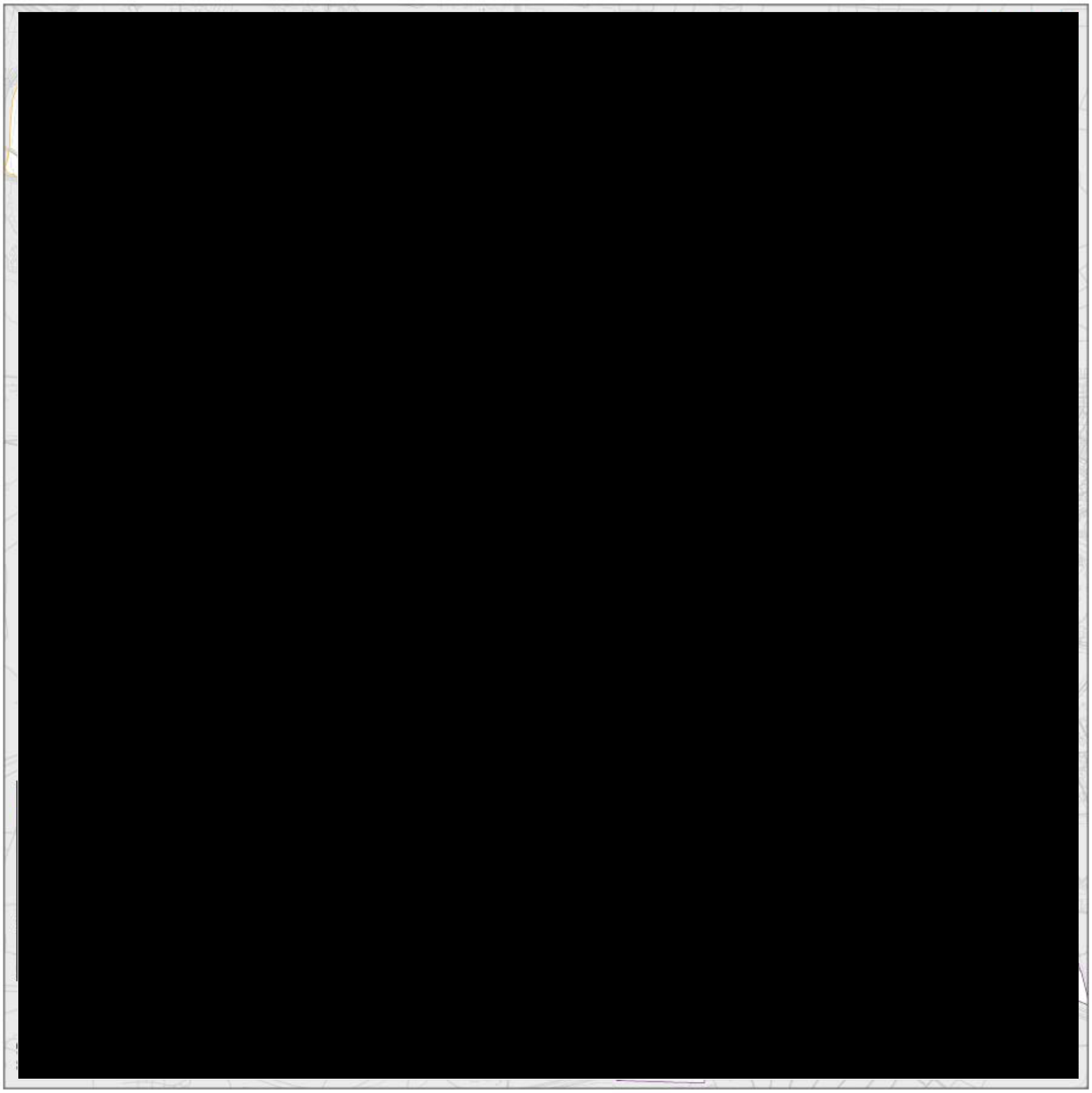

[redacted]

Figure 2-4 – North Carolina State University Campus Map

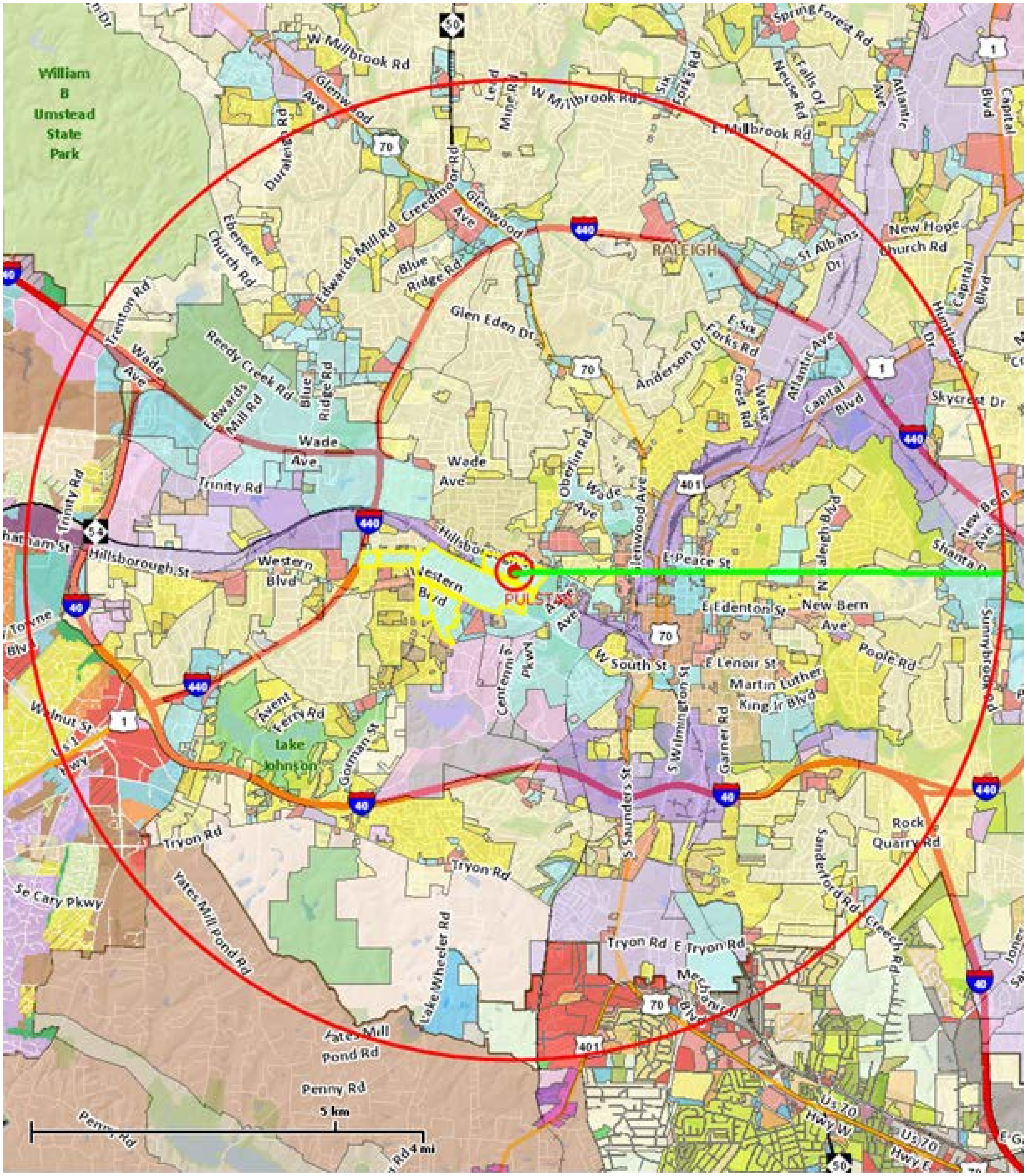


Figure 2-5 – 8 km Area Surrounding the Reactor Site

Figure 2-6 – Topographic Map of North Carolina State University North Campus

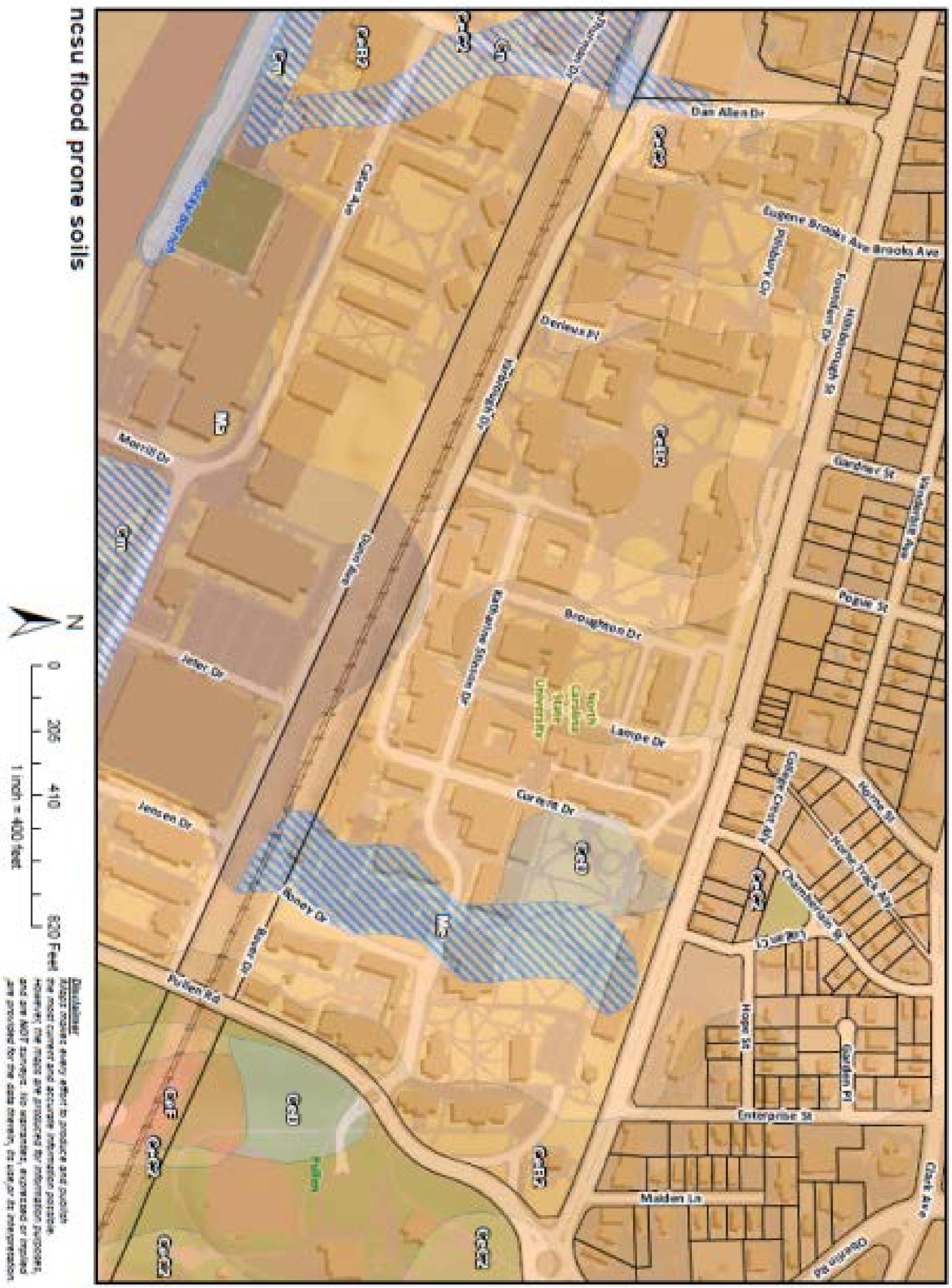


Figure 2-7 – Flood Prone Soils on North Carolina State University North Campus

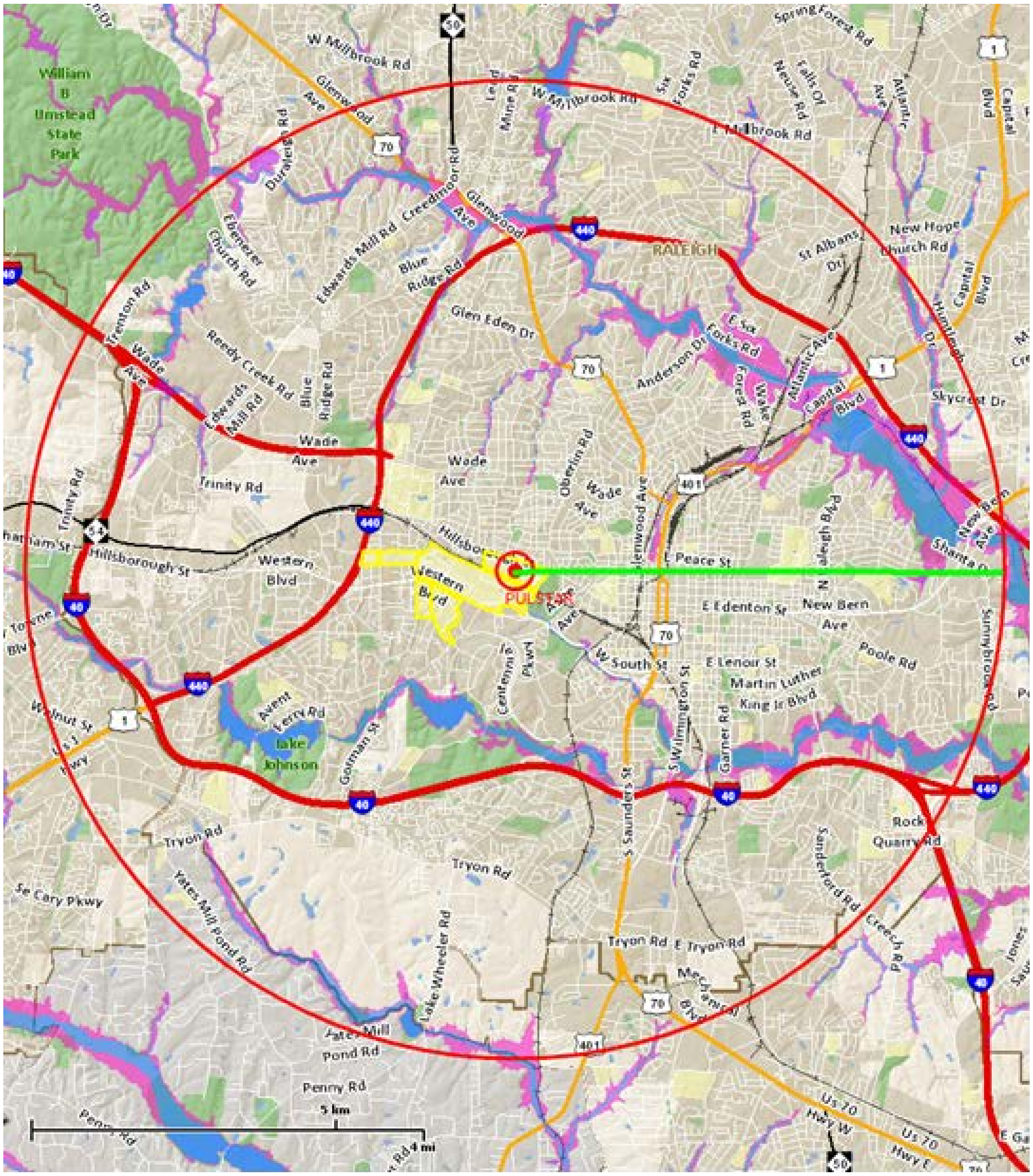


Figure 2-8 – Floodplains with 8 km of Reactor Site

Figure 2-9 – Storm Sewer System at North Carolina State University

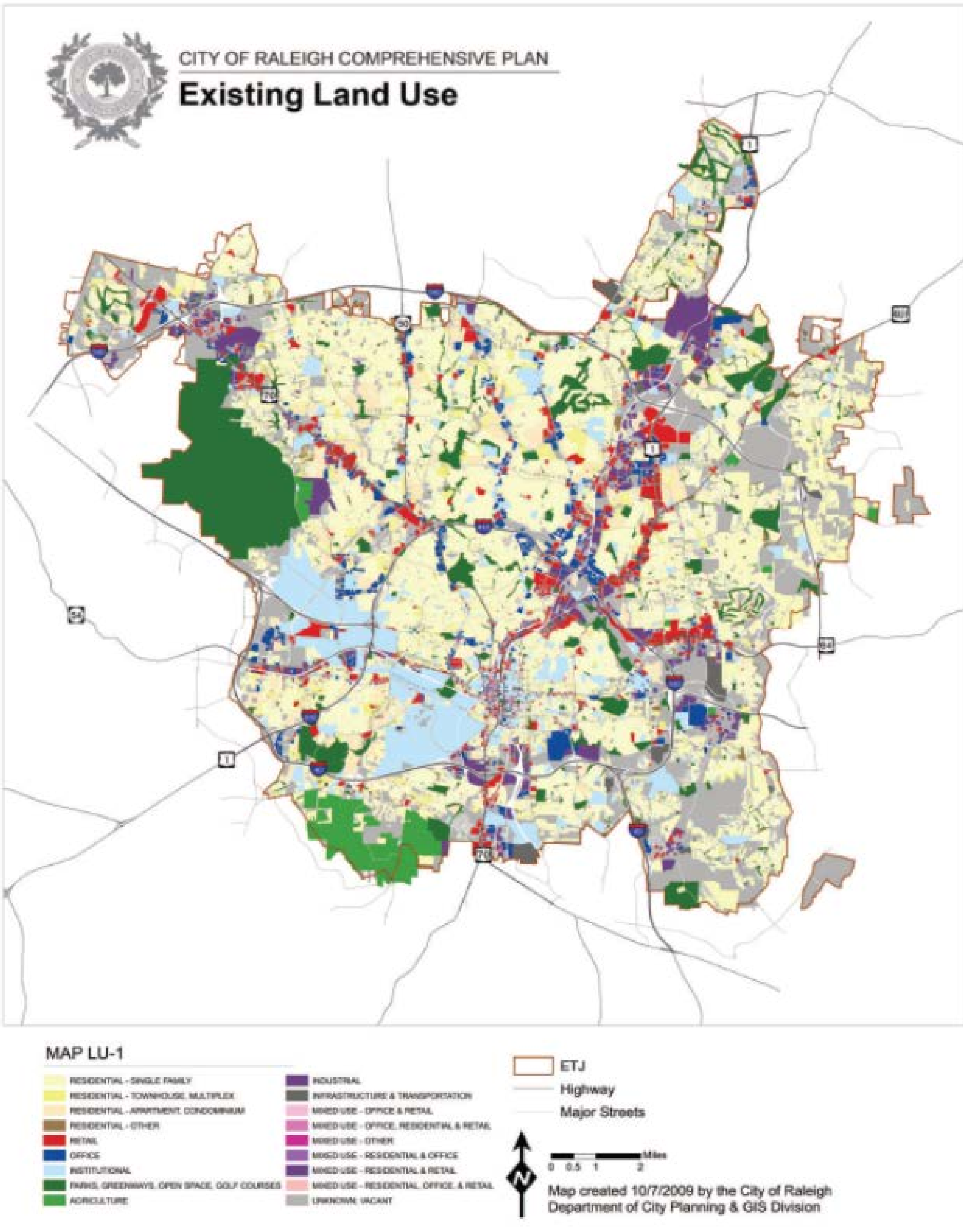


Figure 2-10 – City of Raleigh, NC Existing Land Use Map

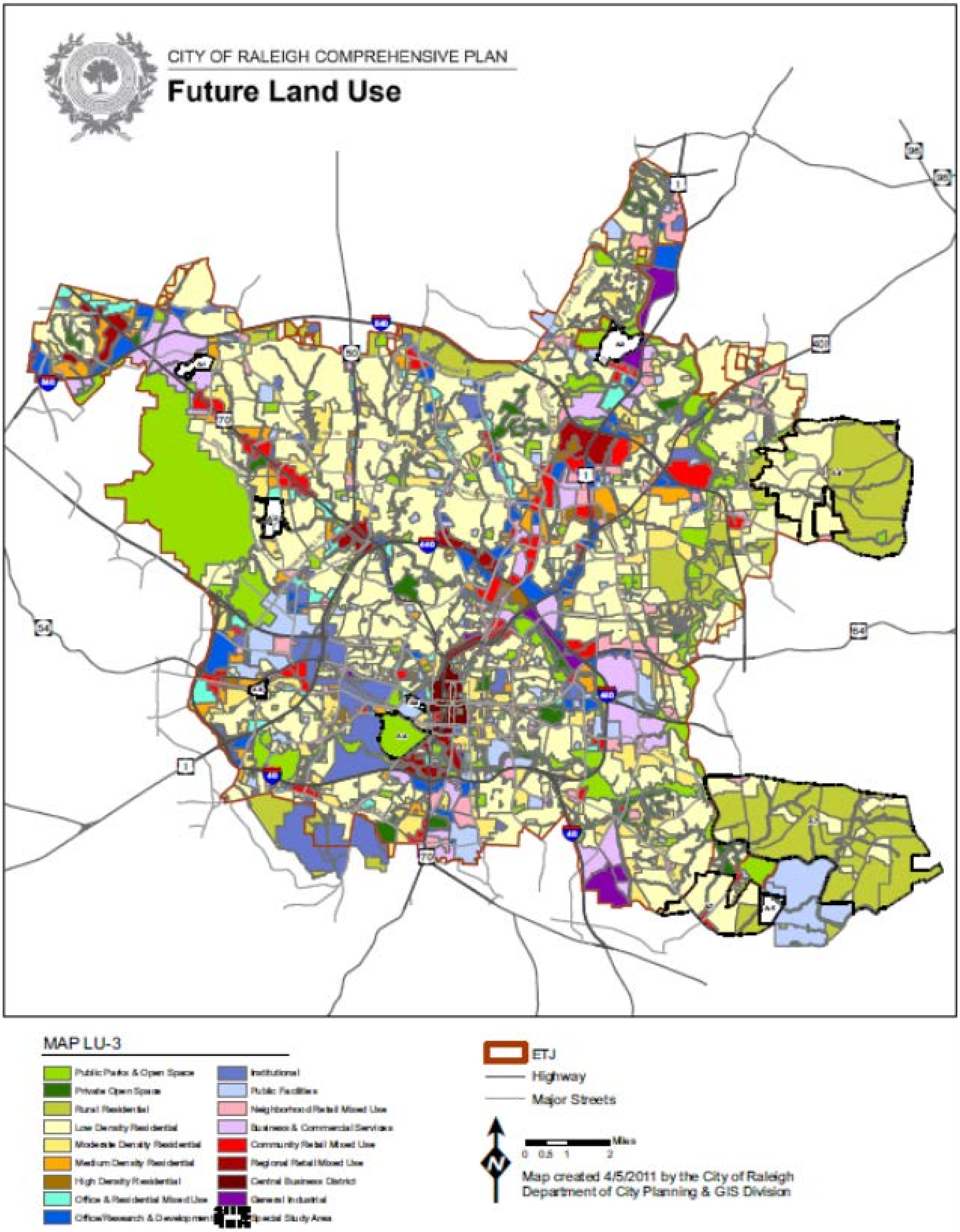


Figure 2-11 – City of Raleigh, NC Future Land Use Map

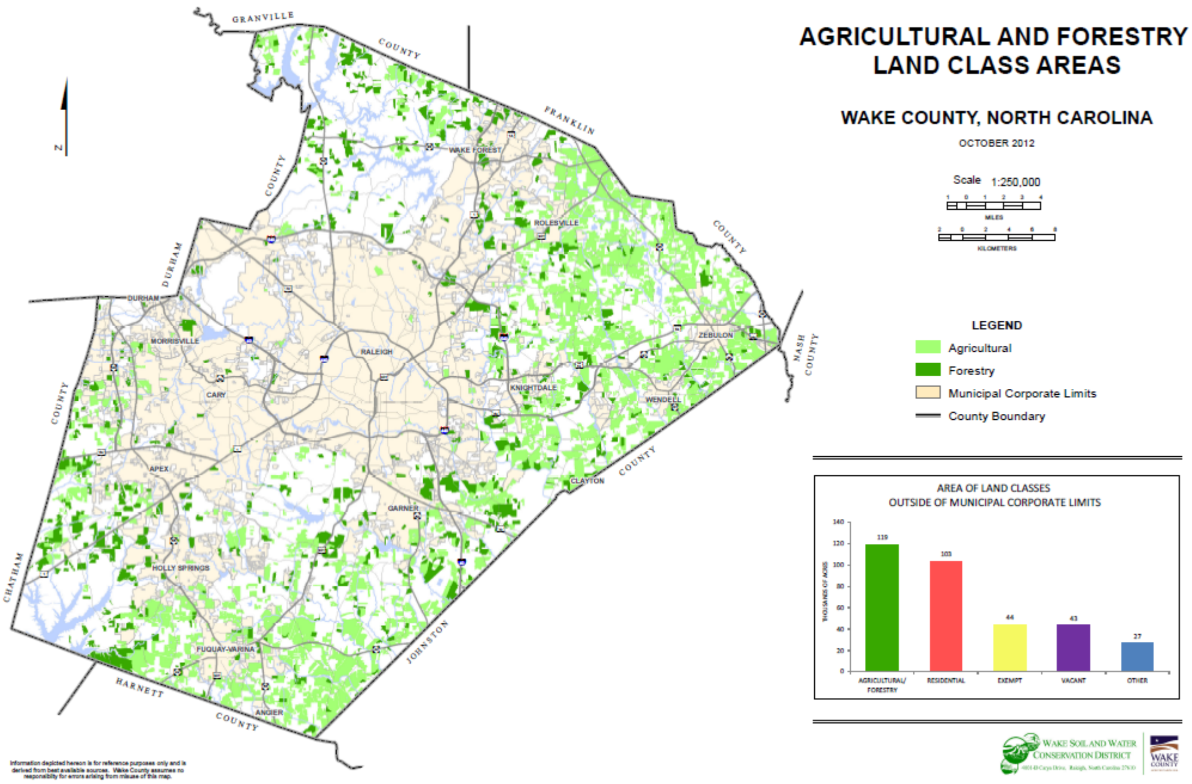


Figure 2-12 – Wake County, NC Agricultural and Forestry Land Class Map – 2012

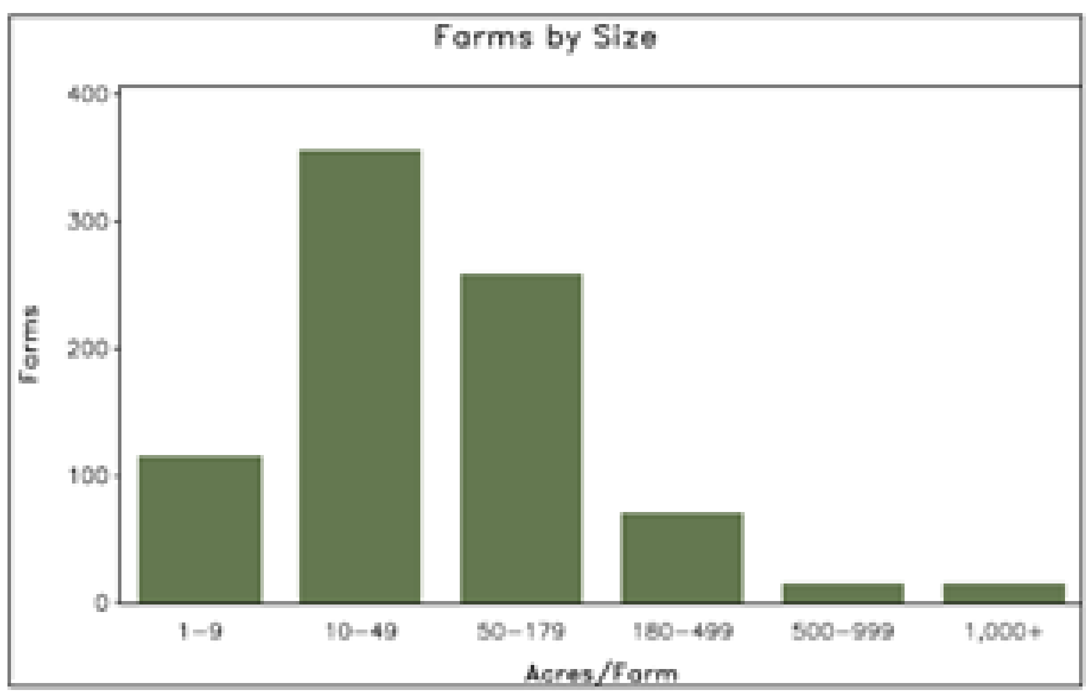


Figure 2-13 – Wake County, NC Farms by Size – 2007

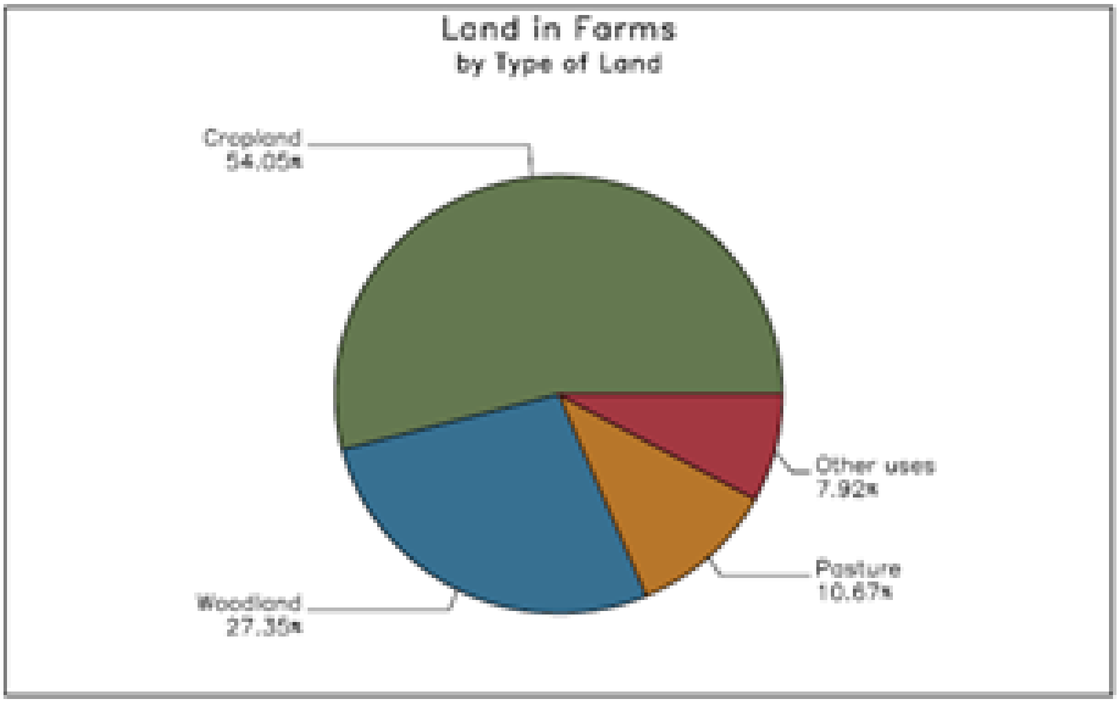


Figure 2-14 – Wake County, NC Land in Farms by Type – 2007

2.1.2. Population Distribution

The NCSU Office of University Planning and Analysis estimates that the student population will increase to 37,000 by the year 2020. This estimate includes approximately 3000 students located in off-campus facilities, giving 34,000 students on-campus. Projections for 2020 give a summer population of approximately 22,000.[2-5]

During summer school the typical number of students on campus during the day is approximately 8,000 (ranges from 6,000 to 10,000) based on data from 2010 through 2012. Week day population in the summer on campus is estimated at approximately half that of the regular academic year, or 20,000 for students, faculty, and staff.

Typically campus visitors are present for brief times during days. Numbers of visitors vary and are typically low. Also students, faculty and staff may be absent. Specific estimates for visitors and absences are not available.

Except for occasional changes in campus student population, no marked variations in the local population exist. Tourism and other seasonal industries are not important factors in population size. On campus, important changes in population takes place in the evenings and nights of the work week, on weekends, and during summer months. Buildings near the PULSTAR reactor are nearly empty at night.

Population data for the area surrounding the PULSTAR reactor site are presented in the 2010 census tracts. 2010 census data for census tracts are listed. The facility is located in census tract 511.02. Three population areas were chosen based on the size, shapes and location of the neighboring census tracts. The census tracts listed are for Raleigh and Wake County, NC.[2-8,2-9]

The population of Raleigh is projected to grow to 580,000 in 2030. The city of Raleigh indicates that greater growth is possible based on an analysis of the land capacity within current jurisdiction and zoning. The growth is expected to occur from development of vacant land within city limits and planned land annexation. Future residential land use near campus is projected to be mostly low population density with small pockets of medium to high density population. Projected population for Raleigh and Wake County are summarized in the tables provided.[2-4]

A portion of the land use near campus is expected to become park and open land in the future. Agricultural land use near campus is expected to become developed office or research use in the future. Retail use is expected to be stable. Figures with current and expected future land use are provided.[2-4]

Table 2-1 – North Carolina State University Population Statistics for Academic Year 2012 – 2013

Group	Week Days	Week Evenings	Week Nights and Week Ends
Students	31,200	14,000	
Faculty and Professional Staff	2,100	900	
Administrative and Support Staff	6,700	100	100
Students Residing in University Owned or Operated Housing		10,000	10,000
TOTALS	40,000	25,000	~10,100

Table 2-2 – North Carolina State University Population Statistics for2020

Group	Week Days	Week Evenings	Week Nights and Week Ends
Students	34,000	15,500	
Faculty and Professional Staff	2,300	1,000	
Administrative and Support Staff	7,300	100	100
Students Residing in University Owned or Operated Housing		11,000	11,000
TOTALS	43,600	27,600	~11,100

Table 2-3 – City of Raleigh, NC Census Tract and Population Data

Distance	Census Tracts					Population
0 to 2.4 km (0 to 1.25 miles)	510	511.02	514	524.08		34,582
	511.01	512	523.02	524.09		
2.4 to 6 km (1.25 to 3.75 miles)	501	507	516	524.01		79,971
	503	508	517	524.04		
	504	509	518	524.07		
	505	515.01	523.01	525.03		
	506	515.02	524.01	525.04		
				545		
6 to 12 km (3.75 to 7.5 miles)	519	528.01	535.06	535.16	540.06	385,603
	520.01	528.02	535.07	535.17	540.08	
	520.02	528.03	535.09	535.19	540.14	
	521.01	528.06	535.12	535.2	540.15	
	521.02	528.07	535.13	535.21	540.17	
	524.06	528.08	535.16	535.24	540.18	
	525.05	528.09	535.17	536.08	541.04	
	525.06	530.03	535.19	536.09	541.05	
	525.07	530.04	535.2	536.1	541.06	
	526.01	530.05	535.21	537.07		
	526.02	530.07	535.24	537.09		
	526.03	530.08	536.08	537.13		
	527.01	530.09	536.09	537.14		
	527.04	532.02	536.1	537.16		
	527.05	532.03	537.07	537.26		
	527.06	532.07	537.09	540.01		
	527.07	534.05	537.13	540.04		

Table 2-4 – Projected Population Estimates

Location	2010	2020	2030
Raleigh	403,892	500,000	> 580,000
Wake County	900,993	1,111,847	1,320,437

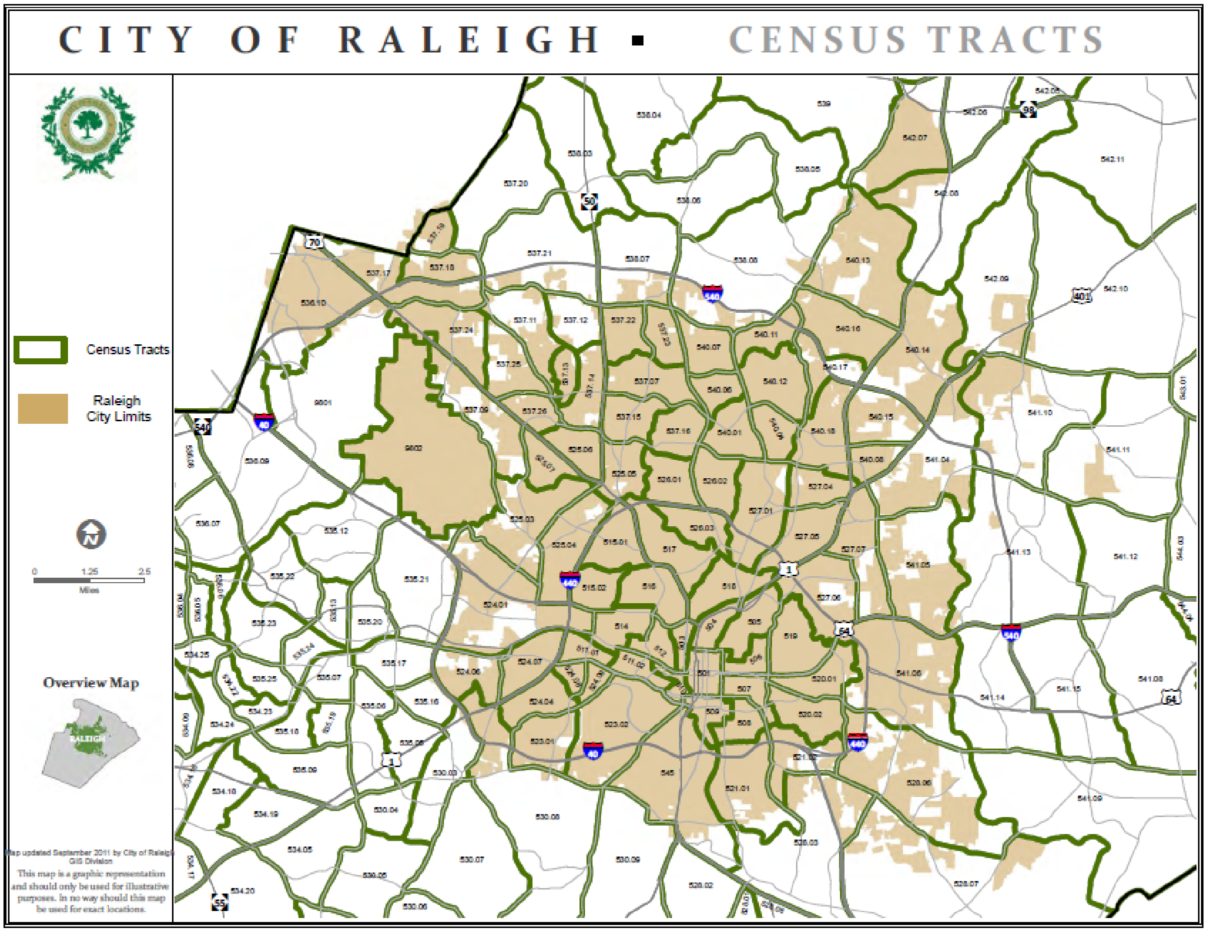


Figure 2-15 – City of Raleigh, NC Census Tracts – 2010

2.2. Nearby Industrial, Transportation, and Military Facilities

Raleigh is the capital city of North Carolina and, as such, contains much of the State governmental facilities and several State institutions. Much of the commercial activity in the area is associated with the operation of State government and State institutions. Heavy industry is less common within an 8 km (5 mile) radius of the reactor site. The industry that is present is mostly located east of the reactor site. Commercial activity within the 8 km radius is mostly retail business activity. The site is largely free from external hazards associated with regular commercial or military air traffic, explosions or fire in industrial areas, strong winds, earthquakes, storm water runoff, or floods. There are no wells for drinking water, no gas or oil pipelines, or tank farms within 8 km of the reactor site.[2-4]

Aggregate and stone quarries are located in the area, but all are more than 8 km away.

According to the Wake County Geographic Information Service, there are no major wells that are still in service located between the reactor site and Rocky Branch Creek to the south.[2-10] A monitoring well has been installed near the corner of Stinson Drive and Broughton Drive southwest of the reactor building. The monitoring well is approximately 12 m deep.[2-11]

Automobile gasoline stations with underground storage tanks are located approximately 1 km from the reactor facility.

The only underground water storage tanks near the reactor are the reactor waste water tanks. There are three 3420 liter (904 gallon) tanks that accumulate reactor and lab waste water located in an underground vault on the reactor site. Waste water is released to the sanitary sewer by gravity drainage. The waste water collection system and discharges are discussed in more detail in Section 11. Storm drains use underground pipes. A roof drain from the reactor building roof is routed through the reactor building and PPV. Other underground pipes near and on the site boundary are used for chilled water and electric and natural gas utilities.[2-12]

A 5 million gallon aboveground potable water tank is located approximately 1.5 km from the reactor site. This water tank is located near residential and retail businesses.

2.2.1. Locations and Routes

Major highways in proximity to the reactor are given in several of the figures provided. The streets surrounding the reactor facility are controlled by NCSU and provide for the transportation needs of the campus population. City streets with significant vehicular traffic near the reactor facility are Pullen Road, Hillsborough Street, Western Boulevard, and Gorman Street. Hillsborough Street is the closest city street to the reactor facility and is approximately 100 m away at its closest point. The nearest interstate is I-440 and is approximately 2 km away from the reactor facility. I-40 is approximately 3 km away.[2-1]

A railway is located approximately 150 m from the reactor facility on the NCSU North campus. The railway is used for passenger and freight trains. The closest rail yard is approximately 2 km away from the reactor site.[2-1]

2.2.2. Air Traffic

The PULSTAR reactor is approximately 10 miles from the Raleigh-Durham International (RDU) airport. Aircraft takeoff and landing trajectories for the major runways at RDU airport are not in line with the PULSTAR facility. Commercial air traffic at the RDU airport accounts for approximately 400 daily flights. Flights by private aircraft and the National Guard also occur, but are less frequent. Air traffic over Raleigh meets Federal Aviation Agency regulation (14 CFR Part 91.119) which requires aircraft to fly at least 1,000 feet above the highest obstacle in congested areas.[2-13,2-14]

2.2.3. Analysis of Potential Accidents

Based on the facilities and activities conducted near the reactor facility, the following accidents are considered (1) railway accident, (2) vehicular accident, (3) aircraft accident, (4) industrial accidents, (5) academic laboratory accidents, (6) construction accidents, (7) fire, (8) storm damage (wind, rain, ice and snow, flood, tornado), and (9) earthquakes. If necessary, the emergency plan would be activated should any accident occur that adversely affects the reactor facility.

Sufficient distance and intervening structures exist to prevent damage to the reactor facility from a railway train accident. Evacuation is the most likely response to a railway accident affecting the reactor facility, e.g. hazardous material spill or fire. Active monitoring of the reactor facility status would be maintained during any evacuation period.

City streets and interstate or expressways are sufficiently distant to preclude a vehicle crash into the reactor facility. Vehicular traffic near the reactor facility is limited to those serving the NCSU campus. Traffic is restricted by NCSU Transportation and Campus Police. Speed limits are low allowing for pedestrian traffic. The reactor building is protected from vehicle crashes due to its structure and surroundings, e.g. reactor building exterior walls are made of reinforced, [redacted] and intervening structures such as other buildings and landscape retaining walls. A vehicle crash causing significant damage to the reactor building is not considered to be credible and therefore is not analyzed.

The RDU airport is sufficiently distant to preclude any aircraft accidents affecting the reactor facility. All air traffic is required to be at least 1000 feet above the tallest structure in the area by FAA regulations. An aircraft impacting the reactor facility is not considered credible due to the low amount of air traffic, air traffic patterns and altitude requirements, and small dimensional size of the reactor.

No industrial or manufacturing processes occur within a sufficient distance to affect the reactor facility. The immediate surrounding area near the reactor facility is predominantly residential, educational, governmental, and retail.

There are no academic laboratory processes occurring that would affect the reactor facility. Laboratory processes are reviewed and limited by NCSU Environmental Health and Safety to keep hazardous materials and processes within regulatory limits for members of the public.

Construction, including renovations and building modifications, on the NCSU campus is reviewed and controlled by NCSU and other parties to meet all applicable safety regulations for occupational personnel, faculty, staff, students and members of the public.

A major fire is a concern within the reactor building and in surrounding buildings and areas. The reactor facility, Burlington Engineering Laboratory (BEL), and all campus buildings have fire detection systems. Smoke and heat sensors are used in the fire detector system at the reactor facility and BEL. There is no sprinkler system in the reactor building or BEL. Upon activation of a fire alarm, audible and visible alarms are activated within the reactor facility and BEL and notification is made by the fire alarm panel to the NCSU Emergency Communications Center. Upon activation of the fire alarm system or being ordered to leave the building by firefighting personnel, the reactor building would be evacuated. After clearance for occupancy is given by the responsible authorities, the reactor facility would be re-entered. If the reactor facility is not involved in the fire, monitoring of the reactor facility would remain operable during the evacuation period.

Storm and earthquake damage is considered to be minimal for the reactor facility based on historical data and on applicable building codes and building ratings for the reactor site.

Risk of flooding of the reactor facility is considered to be minimal based on history, drainage, and topography. The top of reactor biological shield wall [redacted]. The street elevation is approximately elevation [redacted]. A flood of approximately [redacted] the street level would have to occur for the reactor pool to be flooded. The reactor building is unlikely to be flooded based on the drainage and topography of the reactor site. Flash flooding is not a concern for the reactor site. Minor water drainage into the basement of the reactor facility may occur as a result of heavy rains that could briefly overwhelm the storm sewer system. Any water in the reactor facility would drain to the lower level of the PPV, which can hold approximately 14,000 gallons before the remaining water spilled over onto the reactor bay floor. Reactor equipment in the MER is on pedestals ranging from 3 to 10 inches providing an additional 2000 gallons of water storage capacity before being affected by flooding. The reactor facility also has 2700 gallons of water storage capacity in the waste water tanks. The facility waste water systems are described in Sections 5 and 11.

Two dedicated storm drains are located on the reactor building roof and are routed through the reactor bay and PPV. The storm drains enter the storm sewer from the PPV. If a leak occurs, rain water may spill into the reactor pool or into the reactor building or into the PPV. Upon being noticed, the rain water spill could be diverted or stopped. The primary coolant purification system would remove rain water contaminants from the primary coolant. Resistivity of the primary coolant is measured and limited by the facility technical specifications. Flooding by leaking rain water drains is not a concern. Reactor equipment is protected from potential rain water spills, e.g. mounted on elevated equipment pads, and would continue to operate normally.

2.3. Meteorology

Meteorology of the site and the surrounding areas is described in this section. This description includes the following:

1. Historical data on temperature, rainfall, relative humidity, and atmospheric pressure
2. Wind speed and direction, and weather stability
3. General climate and synoptic scale atmospheric processes
4. Historical seasonal and annual frequencies of severe weather phenomena, such as tropical cyclones, tornadoes, severe thunderstorms, high winds, and significant precipitation (rain, hail, sleet, freezing rain, and snow).

2.3.1. General and Local Climate

In general, the climate is of a modified continental type, i.e., warm, temperate and moderately wet with no particularly dry season. In the summer months a maritime tropical air mass predominates. The high humidity, combined with surface heating, tends to make this air mass unstable, with the result that frequent thunderstorms occur from late spring to early fall. Wintertime sees the principal air mass changed to a modified polar continental type with occasional replacement of modified maritime polar air from the Pacific area. Extreme cold waves are rare in this region due to the mountain barrier to the west and the modifying influence of the Atlantic Ocean to the east. Inversions usually accompany the incursion of maritime tropical air in wintertime, due to the interaction of warm air from the Gulf of Mexico with cold polar air, which usually results in widespread overcast skies. Winter snowstorms sometime accompany low pressure centers moving up the coast. North or northeast winds predominate during these periods, bringing in moist cold maritime air from the Atlantic Ocean.[2-15,2-16]

General wind direction and speed, temperature, relative humidity, atmospheric pressure, and precipitation data at the RDU airport and near the reactor facility are provided.[2-16,2-17]

2.3.1.1. Severe Weather Phenomena

Tropical Cyclones (Depressions, Storms, and Hurricanes)

Tropical cyclones are warm-core, intense low-pressure weather systems producing high winds, flooding from intense rainfall, and thunderstorms. Tornadoes may also occur. Tropical cyclones vary greatly in size and intensity, making their impact storm-dependent and difficult to fully predict. Tropical cyclones affecting Raleigh, NC usually take one of three tracks: Coastal Track with a potential damage from high winds, Inland Track with potential damage from flooding and high winds, and Gulf Track with potential damage from flooding.[2-16,2-18]

In the Atlantic Ocean, hurricane season lasts from June through November with the peak of hurricane season in early to mid-September. Storms rarely form outside this season. Winds during a hurricane are generally from the northwest; with the eye of hurricanes usually passing east of Raleigh, and likely off the coast.

Tropical cyclone events in North Carolina from 1851 to 2012 are summarized in Table 2-5. Direct land-falling storms are only those that directly strike the North Carolina coast. Storms affecting North Carolina include all tropical cyclones that have had an impact on the state but did not make direct landfall. Total storms affecting North Carolina include all tropical cyclones that have somehow affected the state.[2-15,2-16,2-18,2-22]

Table 2-5 – Tropical Cyclones Affecting North Carolina from 1851 to 2012

Statistic	Direct Landfall Storms in NC	Non-Landfall Storms Affecting NC Within 150 miles	Total Storms Affecting NC
Number of Storms	45	49	94
Percentage of Storms	2.61%	2.84%	5.46%
Average Years Between Storms	3.6	3.31	1.72
Average Storms per Year	0.28	0.3	0.58

Most tropical cyclones from 1970 to 2012 are tropical storms or depressions and it is noted that Category 4 and 5 hurricanes are rare. An additional 22 extra-tropical and subtropical storms affected North Carolina from 1970 to 2012. Hurricanes come close enough to influence North Carolina weather about twice in an average year. Much less frequently, perhaps once every 10 years on average, these storms strike with sufficient force to damage inland property.[2-18]

An overall upward trend in the 1990s and 2000s is due to improvements in observational tools (especially satellites) and analysis techniques by the National Hurricane Center. Higher numbers are attributable to better detection of short-lived systems.[2-18]

Table 2-6 – Tropical Cyclones Affecting North Carolina from 1970 to 2012

Years	Tropical Depressions	Tropical Storms	Category 1 Hurricanes	Category 2 Hurricanes	Category 3 Hurricanes	Category 4 Hurricanes	Category 5 Hurricanes
1970-1979	20	8	1	1	0	0	0
1980-1989	9	12	3	3	1	1	0
1990-1999	7	4	6	7	3	0	0
2000-2009	10	12	4	2	0	0	0
2010-2012	0	0	2	1	0	0	0
1970-2012	46	36	16	14	4	1	0

Major Hurricanes Affecting North Carolina

On October 1, 1954 high winds from Hurricane Hazel affected Raleigh, NC (73 mph winds averaged over one minute). In this case, all major damage was due to trees falling on houses. Rainfall was approximately 8 inches. This storm pre-dates the current reactor facility.[2-18,2-22]

On September 22, 1989, Hurricane Hugo came ashore at Charleston, South Carolina, and its eye later passed over Charlotte, North Carolina (and therefore, west of Raleigh). No damage was recorded on the NCSU Campus or to the PULSTAR reactor facility as a result of the inland movements of Hurricane Hugo.[2-18]

Hurricane Fran made landfall on September 6, 1996 near Cape Fear, NC as a Category 2 hurricane with sustained wind speeds of 115 mph. Hurricane Fran then moved inland and weakened to a tropical storm as it passed through Raleigh, NC. Damage in Raleigh, NC was caused by high winds and heavy rainfall and flooding. In Raleigh, the peak wind gust was 86 mph with sustained winds of 70 mph and rain totaling more than 10 inches. There was significant damage from fallen trees, including damage from many trees on the NCSU campus. No damage occurred at the reactor facility.[2-18,2-19]

In September and October 1999, Hurricanes Dennis, Floyd and Irene affected North Carolina. Some areas in eastern NC received in excess of 37 inches of rain from the three storms. Raleigh received 24 inches of rain. Damage from the storms was associated with flooding. No damage occurred at NCSU or the reactor facility.

Tornadoes and Severe Thunderstorms

The National Weather Service (NWS) Weather Forecast Office (WFO) in Raleigh, NC reviewed severe weather climatology for the County Warning Area (CWA) in central North Carolina from January 1, 1950 to December 31, 2005.[2-18]

The Raleigh CWA is comprised of 31 counties (including Wake County) and covers 16,459 square miles including the metropolitan areas of Raleigh/Durham and Chapel Hill (Triangle), Greensboro/Winston-Salem and High Point (Piedmont Triad), Rocky Mount/Wilson and Fayetteville/Fort Bragg.

For central North Carolina, the combination of abundant low-level moisture from both the Atlantic Ocean and Gulf of Mexico along with frontal boundaries that interact with this moisture often set the stage for strong to severe thunderstorm development. As a result, the Raleigh CWA experiences a wide variety of weather phenomena, including severe thunderstorms that produce tornadoes, large hail, and damaging wind gusts.

Tornadoes

There have been 284 tornadoes reported across the Raleigh CWA from 1950 to 2002. All 31 counties in the Raleigh CWA have had at least 2 tornadoes confirmed. Tropical cyclones or their remnants can track through the Southeast and Mid-Atlantic region. Tornadoes frequently occur in the northeast quadrant of northward advancing tropical systems or their remnants. North Carolina is outside the principal tornado area of the United States. They occur mostly east of the Appalachian Mountains. On average, North Carolina experiences approximately 12 tornadoes per year and the Raleigh CWA experiences approximately 5 tornadoes per year.[2-18]

Tornado intensity was rated using the Fujita Scale (F1 to F5), which is based on the extent of the wind damage. Of the tornadoes that were reported to have occurred in the Raleigh CWA, nearly three-quarters (71%) were classified as weak F0 or F1 tornadoes, 26% were rated strong (F2 or F3), and 2% were rated as violent F4 tornadoes. Of the 284 tornadoes reported in the Raleigh CWA, 48 were associated with tropical systems or their remnants. Of these 48 tornadoes, 46 were classified as weak tornadoes (F0 or F1). Three of the five F4 tornadoes occurred in the March 28, 1984 Carolina Tornado Outbreak. During the period of 1884 through 1994, only six F4 severity class tornadoes have been recorded in North Carolina. There were no documented F5 tornadoes.

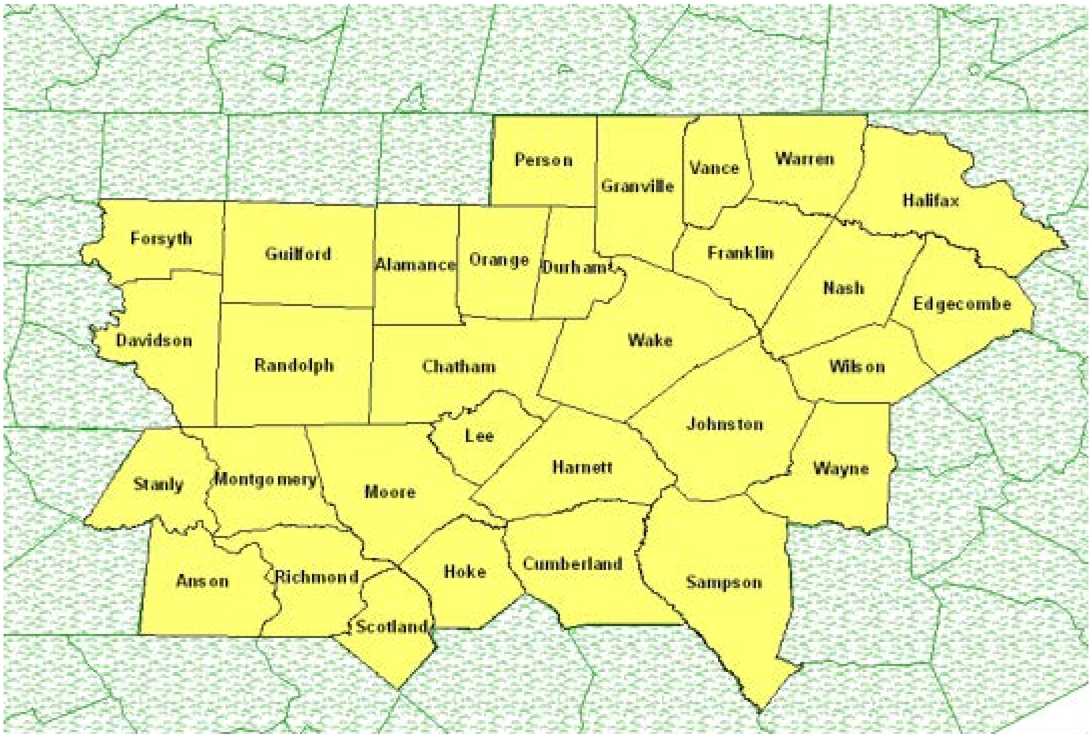


Figure 2-16 – Weather Forecast Office (WFO) Raleigh, NC (RAH) County Warning Area (CWA)

Within a five mile radius of the reactor, there have been approximately five occurrences of winds with tornado characteristics over the past 80 years. The two most severe tornadoes that occurred within the five mile radius occurred in 1988 and 2011.

On November 28, 1988, Raleigh was struck by one or more tornadoes. The track of the November 1988 tornado was 84 miles in length and was estimated at its onset as F4 severity on the Fujita Scale with winds estimated at 210 mph. This particular tornado had an estimated ground speed of 50 mph and passed through Wake, Franklin, Nash, Halifax, and Northampton Counties. At its closest point, it passed within approximately 5.5 miles of the PULSTAR reactor facility.

Nine tornadoes occurred in the Raleigh CWA on 16 April 2011 including two EF-3 tornadoes (Enhanced Fujita scale), four EF-2 tornadoes and three EF-1 tornadoes. The nine tornadoes in the Raleigh CWA were produced by four supercell thunderstorms, with each supercell producing at least two tornadoes. The Sanford-Raleigh tornado had maximum winds of 160 mph and a path length of 66.8 miles. This tornado started near Carthage, NC (southwest of Raleigh) and proceeded through Raleigh before ending near Wake Forest, NC (northwest Wake County). At its closest point, it was approximately two miles east of the reactor facility as it passed through Raleigh. No damage occurred at the reactor facility or on the NCSU North Campus.[2-18]

The probability of occurrence of any tornado is 0.027% with a return interval of 3647 years and a frequency of 1 per 1672 square miles. The probability of occurrence of a F2 or higher rated tornado is 0.023% with a return interval of 4351 years and a frequency of 1 per 10,518 square miles. The average number of days per year in the Raleigh CWA with a tornado is 12.[2-22]

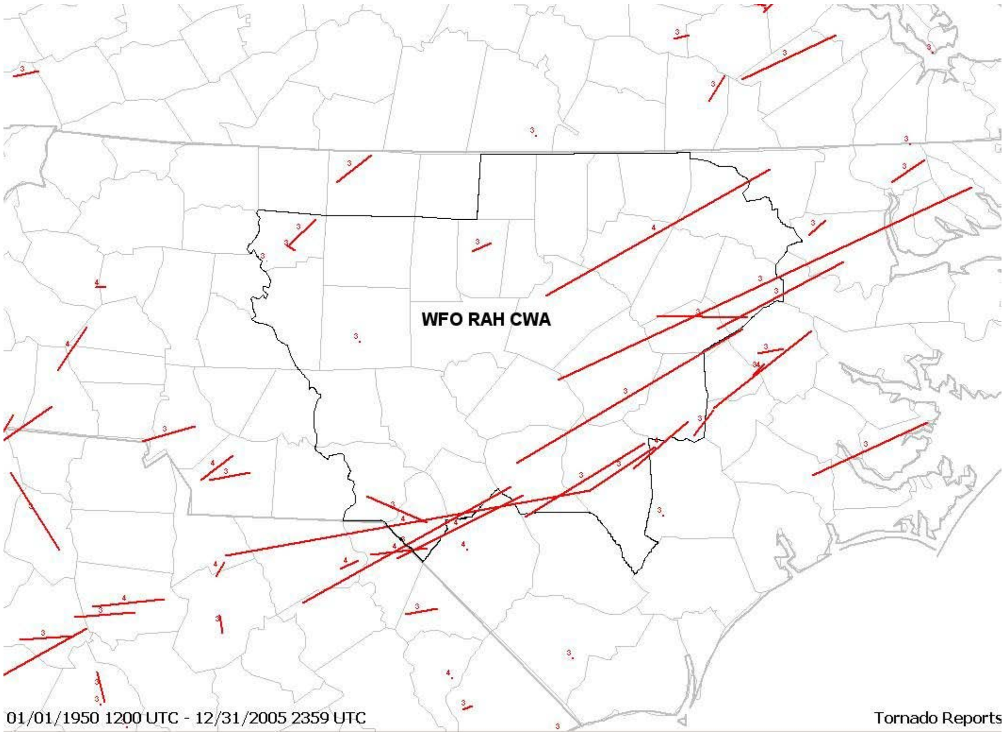


Figure 2-17 – Historical F3-F5 tornado Tracks (1950 – 2005) in the WFO Raleigh CWA

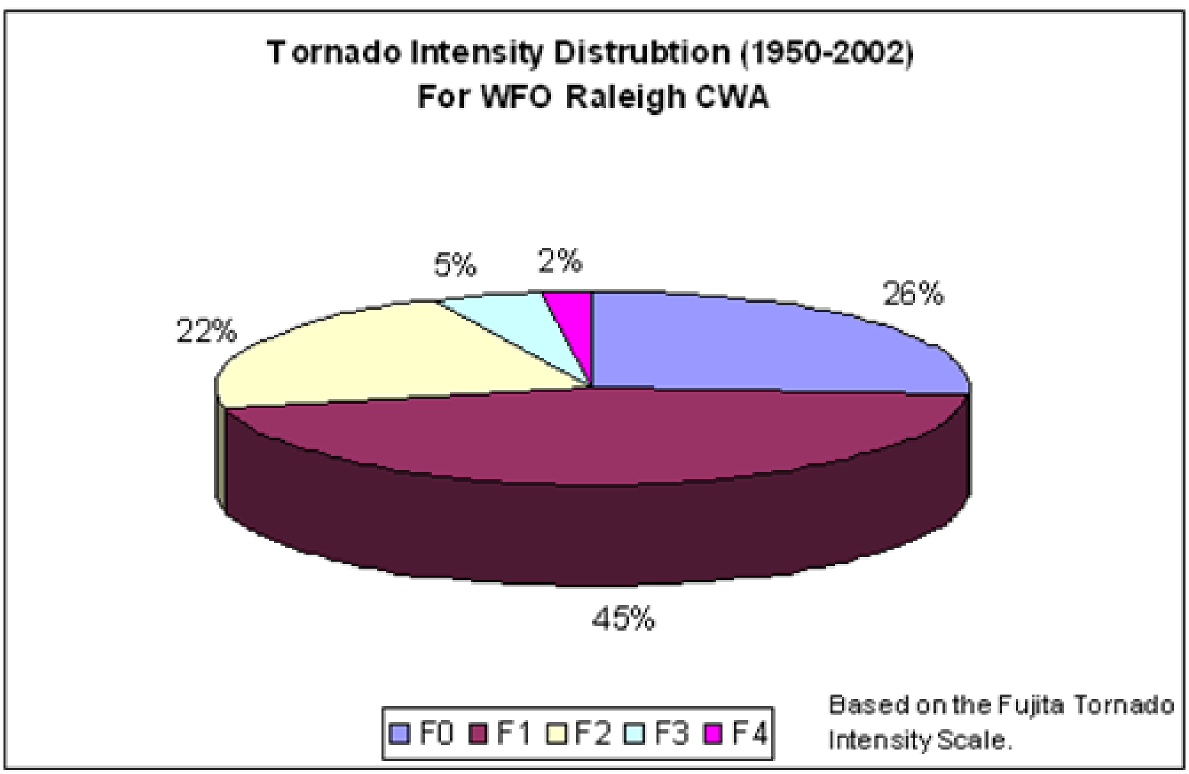


Figure 2-18 – Tornado Distribution by Fujita Scale (1950 – 2002) for Raleigh CWA

2011 Tornadoes

A historic outbreak of tornadoes occurred in North Carolina in April 2011. A total of nine tornadoes occurred in the Raleigh CWA on 16 April 2011 including two EF-3 tornadoes, four EF-2 tornadoes and three EF-1 tornadoes. There were 30 tornadoes across North Carolina on 16 April 2011. The map below highlights the tornado tracks across the entire state of North Carolina with the maximum EF scale rating for each tornado shown in red.[2-21]

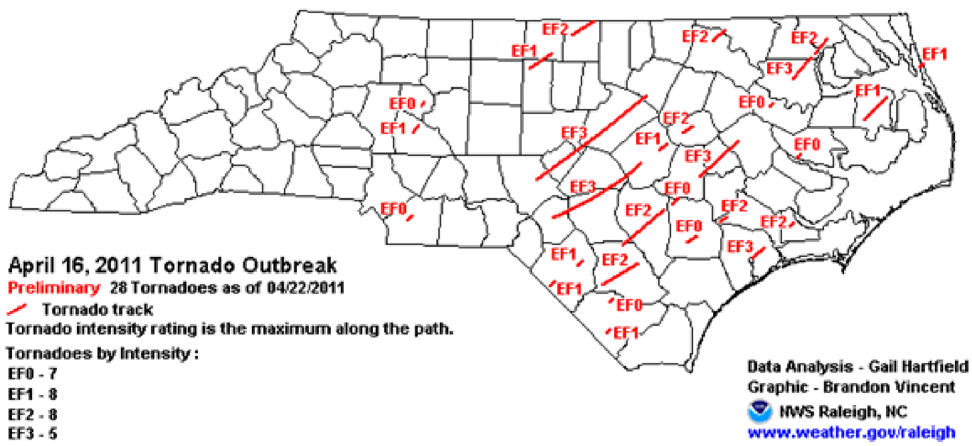


Figure 2-19 – North Carolina Map of Tornado Outbreak of April 16, 2011

A summary of the various statistics for each of the 9 tornadoes in the Raleigh CWA along with the supercell data is shown in the Table 2-7.[2-21]

Table 2-7 – North Carolina Statistics for Tornado Outbreak of April 16, 2011

Tornado	Supercell	Counties	Max EF Scale	Max Wind mph	Fatalities	Injuries	Damage	Max Width yards	Path Length miles
Alamance County	1	Alamance	EF-1	110	0	0	$580,000	75	7.4
Person County	1	Person	EF-2	125	0	2	$400,000	300	9.8
Sanford-Raleigh	2	Moore, Lee, Chatam, Wake, Franklin	EF-3		6	103	$172,075,000	500	66.8
Roanoke Rapids	2	Halifax	EF-2		0	0	$1,200,000	250	4.9
Fayetteville-Smithfield	3	Hoke, Cumberland, Harnett, Johnston	EF-3		2	109	$116,100,000	1700	58.5
Micro	3	Johnston	EF-1		0	67	$25,000,000	100	8.7
Wilson	3	Wilson	EF-2		0	10	$3,000,000	200	9
Cumberland-Sampson	4	Cumberland, Sampson	EF-2		0	13	$10,250,000	800	28.5
Wayne County	4	Wayne	EF-0		0	00	$5,000	400	2.8
Totals					**8**	**304**	**$328,610,000**		**196.4**

The Sanford-Raleigh tornado was on the ground for an estimated 67 miles (107 km) and the tornado was about 500 yards (450 m) in width at its peak. This tornado was approximately 2.25 km (or 1.5 mile) from the reactor facility at its closest point with an intensity of EF1.

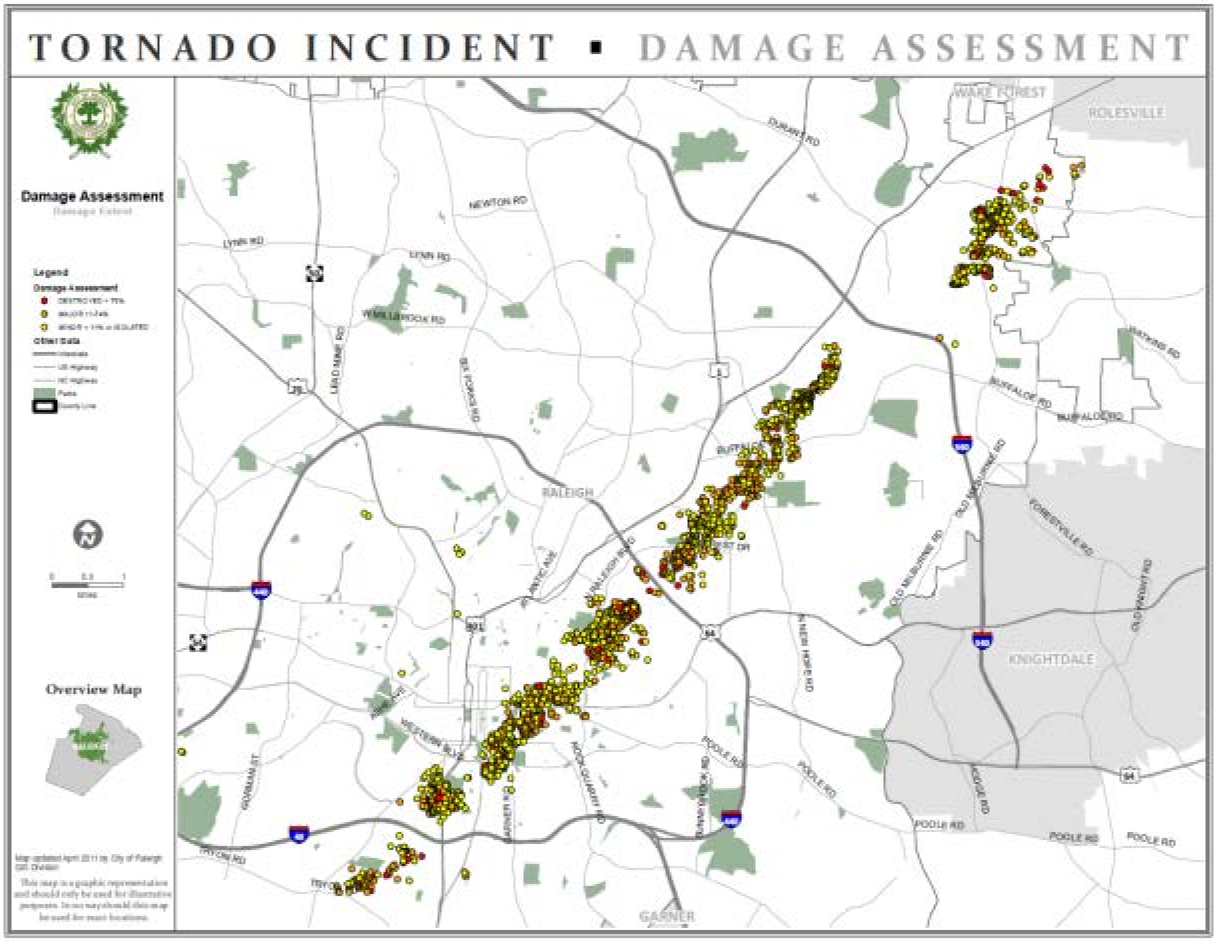


Figure 2-20 – Raleigh, NC Damage Assessment Map for Tornado Outbreak of April 16, 2011

The Sanford-Raleigh tornado ranged in intensity from EF3 to EF0. Damage from the tornado occurred along its entire path. Damage in Raleigh is indicated in Figure 2-20.[2-23] No damage occurred at the reactor facility.

Severe Thunderstorms

As defined by the NWS, a severe local storm is one that is sufficiently intense to threaten life and/or property, including thunderstorms with large hail, damaging wind, or tornadoes. More specifically, severe thunderstorms are further defined by the NWS as a storm that meets one or more of the following: a tornado, hail three-quarters of an inch in diameter or larger, or wind of at least 50 knots (58 mph) or wind which causes damage, including trees or power lines blown down. The Raleigh CWA averages about 40 to 50 severe thunderstorms a year.[2-18]

Topography is a contributing factor in the initial development of convective storms. The two principal topographic regions that encompass the Raleigh CWA are the piedmont and coastal plain regions. Differences in soil types and elevations in these regions are associated with convective storms. A notable increase in elevation occurs west of the “fall line” which divides the piedmont and coastal plain. The piedmont is characterized by rolling hills and ranges in elevation from 250 feet to 1100 feet while the coastal plain is characterized by flat land and ranges from 50 feet to 250 feet in elevation. It is along the “fall line” where the variety of soil types of the coastal plain, which are soft sediment or sand, is bounded by the clay-loam soil of the piedmont. The differences in these soil types create differential heating of the surface. As a result of daytime heating, convective thunderstorms occur in the afternoon and evening, especially in the late spring and summer months. The “fall line” is located 10 to 20 miles or more east of Raleigh and the reactor site.[2-18]

Damaging Winds

Strong, damaging winds resulting from severe thunderstorms, fast moving squall lines or bow echoes are the most frequent severe weather event across the Raleigh CWA. Over the 52-year period between 1950 and 2002, there were 2119 severe thunderstorm wind events (60% of all severe events). Severe thunderstorm damaging wind events from convective storms show a steady increase during the spring and peak in June. Nearly one quarter of the severe thunderstorm damaging wind events (23% or 488 events of the total) occurred in June. During the late spring and early summer months of May, June and July, 1268 events (60%) occurred.[2-18]

Thunderstorm wind damage is most common during the mid-afternoon through evening hours. However, thunderstorm wind events may occur during any hour of the day. A three second gust wind speed of 96 mph is reported for the 100 year mean recurrence interval.[2-16]

Lightning

The National Weather Service in Raleigh as well as collaborative researchers at NCSU evaluated cloud to ground (CG) lightning from 2003 through 2010 using data from the National Lightning Detection Network (NLDN).[2-20]

CG lightning data was constructed from local archives of the Advanced Weather Interactive Processing System (AWIPS). Statistical point data for selected cities were derived using a 25 km^2 grid box centered over the associated airport location.

At RDU airport, there were 2429 strikes per year on average. July had the most flashes. A dramatic decline in flashes from August to September was observed and reflects the climatologically drier fall. The minimum months were November, December, January, and February. Annual average flash density at RDU airport was approximately 4 flashes per km^2. Monthly average flash density ranged from 0 to approximately 1.4 flashes per km^2.[2-20]

The reactor facility electrical system meets building codes and backup electrical power is provided for designated systems, including the radiation monitoring and security systems.

Hail Climatology

The monthly distribution of severe hail (0.75 inch diameter or greater) indicates a strong inclination toward the spring season. The peak occurrence of hail in spring is largely due to the combination of relatively warm near surface temperatures with freezing temperatures in the mid-levels of the atmosphere.[2-18]

Similar to the hourly tornado distribution, there is a dramatic increase in severe hail after the noon hour. The peak occurrence of hail frequency during the early-to-midafternoons can be attributed to several factors including strong updrafts and atmospheric instability. Atmospheric instability is maximized during the afternoon.

Nearly half of severe hail reported (595 events or 49% of the total) in the Raleigh CWA was less than one-inch diameter. Occurrences of hailstones ranging from one to two inches accounted for 47% of the reports. Severe hail of over 2 inches in diameter accounted for only a small percentage (~4%). The largest hailstone measured in the Raleigh CWA during the period was 4.5-inch diameter. The softball size hail occurred in Montgomery County with a non-tornadic supercell thunderstorm.

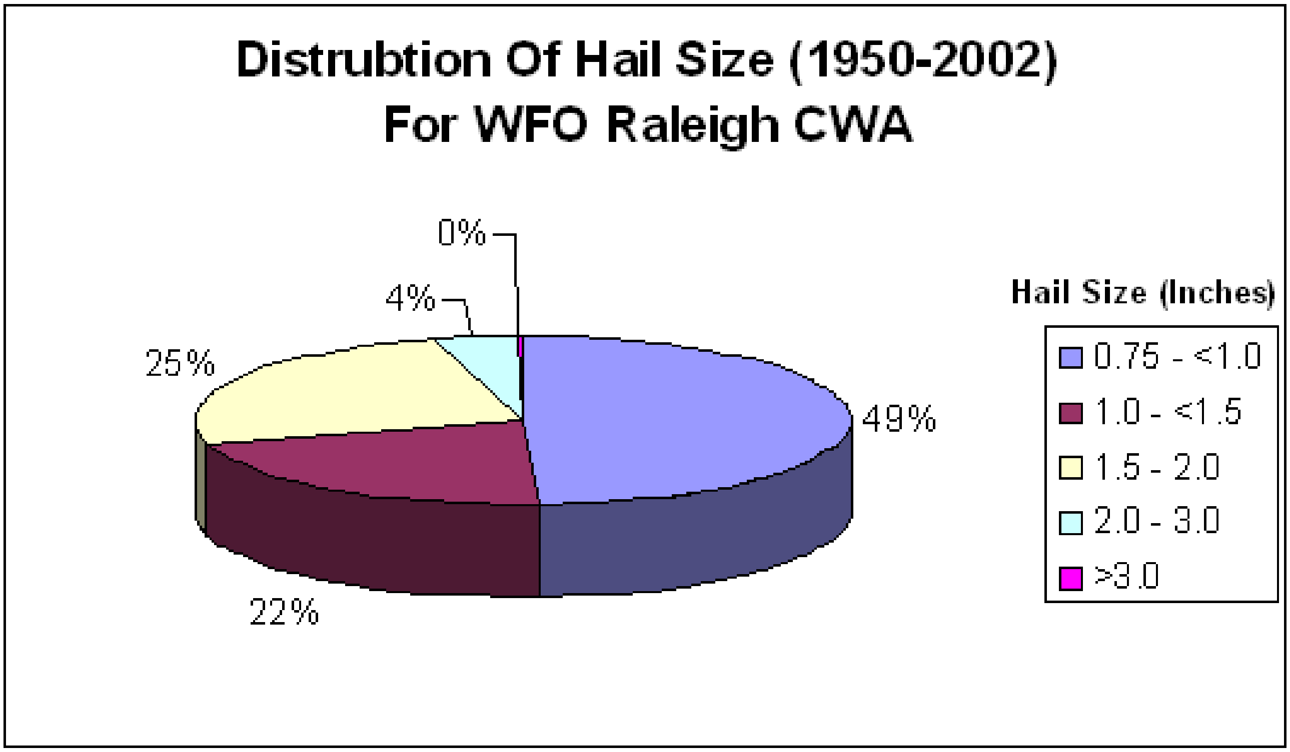


Figure 2-21 – Size Distribution of Hail Events (1950 – 2002) for Raleigh CWA

Probable Maximum Precipitation

Hydrometeorological Reports (HMR) 51 and HMR 53 from NOAA provide data on 6 to 72 hour probable maximum precipitation (PMP). For Raleigh, NC, the PMP for a 10 square mile area are listed in Table 2-8.[2-24,2-25]

From this data, the 48 hour PMP is estimated to be 45 inches. The roof of the reactor building has two 4 inch diameter rain drains with horizontal leaders that feed into vertical leaders. The roof is flat and has an area of approximately 3000 square feet. The PMP has an average rate of approximately 5 inches per hour based on 6 hour PMP and approximately 1 inch per hour based on 48 hour and 72 hour PMP. The two 4 inch diameter rain drains are capable of draining all rainfall up to a rate of 5 inches per hour based on a horizontal slope of 1/8 inch for the horizontal leaders. Therefore, it is concluded that the reactor roof will be effectively drained for any rain event with no accumulation of water on the roof.[2-12]

The roof drains for the reactor building are independent of the rest of the BEL and feed directly to the campus storm water system. The roof is flat with a rubber membrane with no gravel or drain rocks. Obstruction of the roof drains is unlikely but in event that a drain does become clogged, the building does not have a parapet therefore water would cascade off the side thus preventing any potential for overloading the rating of the roof structure. The exterior of the building is maintained by NCSU Facilities Operations who would be notified to inspect and clear the roof of excess snow, ice or roof drain obstructions. Refer to Section 3.2.

Table 2-8 – Hydrometeorological Report Data for Raleigh, NC

HMR53 Month	6 h PMP (in) over 10 mi^2	24 h PMP (in) over 10 mi^2	72 h PMP (in) over 10 mi^2	
Jan	12 to 14	20 to 22	24 to 26	
Feb	12 to 14	20 to 22	24 to 26	
Mar	14	20 to 22	26 to 28	
Apr	16 to 18	22 to 24	28 to 30	
May	20 to 22	26 to 28	32 to 34	
Jun	26 to 28	36 to 38	42 to 44	
Jul	26 to 30	40	46 to 48	
Aug	26 to 30	40	46 to 48	
Sep	28 to 30	40 to 42	46 to 48	
Oct	24 to 26	34 to 36	42 to 44	
Nov	16 to 18	26 to 28	32 to 34	
Dec	14	20 to 22	26 to 28	
HMR 51 All Season PMP over 10 mi^2				
6 h	**12 h**	**24h**	**48 h**	**72h**
29 to 30	35 to 36	40 to 42	44 to 46	46 to 48

Winter Storms

North Carolina winter weather consists of storms that produce snow, sleet, freezing rain or a wintry mix of multiple precipitation types. It is not uncommon to experience every mode of wintry precipitation in any given storm, due to the nature and atmospheric conditions that are commonly found in NC winter storms. Due to the proximity to the Appalachian Mountains, Atlantic Ocean, Gulf Stream, and Gulf of Mexico, various weather patterns can result in winter weather across NC.[2-18]

North Carolina winter weather evolves from complex meteorological patterns that are difficult to forecast, and which lead to considerable variability in weather conditions across the state. As a result, the frequency and intensity of winter weather events are highly variable and dependent upon large (synoptic) and small (mesoscale) scale atmospheric patterns.

The occurrence of freezing rain typically begins in early December and phases out in mid-March. There is no distinctive period of maximum frequency, although the greatest frequencies occur in late December, early February, and early March.

The distribution of snowfall indicates a period of maximum frequency in mid-January, with a shift to early February along the coast. A gradual increase in snowfall frequency is observed throughout the early part of the winter (typically beginning in early November) with a sharper decline occurring in late-January and into March and April.

Annual probabilities for freezing rain, sleet, and snow accumulation from a winter storm for Raleigh, NC from 1948 to 2003 are given in Table 2-9.[2-15,2-22]

Table 2-9 – Annual Winter Precipitation for Raleigh, NC

Freezing Rain Amount (Inches)	Annual %	Sleet (Inches)	Annual %	Snowfall (Inches)	Annual %
Measurable	~ 100	Measurable	77	> 1.0	~ 100
> 0.12	91	> 0.12	27	> 2.5	83
> 0.25	50	> 0.25	14	> 5.0	48
> 0.50	13	> 0.50	5	> 7.5	16
> 0.75	5	> 0.75	4	> 10	7
> 1.00	4	> 1.00	2	>15	2
> 1.50	0	> 1.50	0	> 20	2

The annual recurrence of various snowfall intensities shows great latitudinal variability, with some coastal influence. Snowfall totals of 20 inches and greater are generally confined to the mountains, although a rare winter storm of 22 inches occurred in Raleigh in January, 2000.[2-18]

Raleigh, NC receives brief periods of snow and has no reported snowpack. The maximum snowfall that has occurred is 22 inches. Density of snow varies with water content with reported values from 5% water weight (2.12 lbs/ft^3) to 33% water weight (20.81 lbs/ft^3).[2-17,2-22] For 22 inches on the reactor roof, the estimated weight per unit area would range 5.8 lbs/ft^2 to 38.2 lbs/ft^2.

Ice up to 1 inch is rare and is the extreme case for Raleigh, NC. 1.5 inches of ice or sleet is extremely unlikely and gives a weight per unit area of 7.8 lbs/ft^2.[2-17]

Roof design load is 30 lbs/ft^2. This equates to 115 inches of light snow or 16.2 inches of heavy snow or 5.8 inches of ice, sleet, or water.[2-12]

The severe weather climatology for Raleigh CWA Weather Forecast Offices review of historical data has provided the following notable findings:[2-18]

1. While tornadoes can and do develop at any time of the year, the majority of tornadoes (60%) occur between March and June, peaking during the month of May.
2. A secondary peak for tornadoes occur in the fall associated with the inland effects from tropical cyclones and from the influence of synoptic scale storm systems associated with seasonal migratory upper level jet stream.
3. Nearly 75% of all tornado events were classified as weak tornadoes (F0 or F1) and only 5 tornadoes (2%) of F4 intensity have been reported. There have been no reports of tornadoes rated as F5 intensity.
4. 78% of severe hail events occurred between April and July
5. Almost half (49%) of all severe hail reports are less than one-inch diameter (quarter size).

6. Severe thunderstorm damaging winds are the most frequent severe event (60%) across the Raleigh CWA.
7. Over half (60%) of all thunderstorm damaging wind events occur in the late spring and early summer months of May, June and July, peaking in June.

It is noted that counties with a higher population density are more likely to report more events. Low population density counties are more likely to have events that are not witnessed firsthand and therefore go unreported. Wake County has a higher population density.

Table 2-10 – Annual Totals for Severe Weather for Raleigh CWA

Year	Hail		Wind		Tornado		Total	
	Report	Days	Report	Days	Report	Days	Report	Days
1980	19	8	49	28	17	13	53	35
1981	16	9	34	20	15	7	46	28
1982	36	18	73	27	15	10	66	37
1983	28	15	141	43	19	14	66	52
1984	84	22	179	45	32	11	148	52
1985	183	29	210	38	7	5	197	44
1986	102	32	260	57	18	9	138	63
1987	90	32	184	56	4	3	98	65
1988	151	38	178	48	20	11	191	66
1989	93	31	379	50	28	15	149	60
1990	126	38	225	56	21	15	168	72
1991	33	17	134	43	27	11	87	56
1992	97	31	178	48	24	10	145	59
1993	207	36	270	53	23	9	253	70
1994	107	38	240	65	21	15	149	74
1995	199	46	503	62	39	14	277	74
1996	261	55	442	78	62	23	385	96
1997	186	39	244	58	11	9	208	69
1998	509	55	341	58	61	16	631	79
1999	171	42	331	53	44	11	259	63
2000	438	60	467	65	21	12	480	83
2001	178	38	214	44	16	11	210	55
2002	266	52	447	60	14	5	294	73
2003	386	52	492	80	44	15	474	96
2004	159	42	292	63	112	25	383	88
2005	176	39	301	49	30	12	236	63
2006	688	68	763	70	35	16	758	88
AVG	185	36	280	52	29	12	494	65

2.3.2. Site Meteorology

The purpose of this section is to present data which are directly applicable to the computation of radiological hazards which might be present during or after reactor operation. Several meteorological measurements applicable to the site are available. The National Oceanic and Atmospheric Administration (NOAA) / National Weather Service (NWS), State Climate Office of North Carolina Data, and Harris Nuclear Power Plant have reported weather data applicable to the NCSU reactor site.[2-15,2-22,2-26,2-27,2-28]

Data provided in this section includes:

1. Wind rose (wind speed, wind direction)
2. Atmospheric stability
3. Surface temperatures
4. Atmospheric pressure
5. Humidity
6. Dew Point
7. Precipitation and Snow

Information on weather station location and instrumentation are provided in Table 2-11 through Table 2-16. Average data for multiple consecutive years are used to predict weather conditions at the reactor site.

Wind patterns at 10 m, 12 m, 30 m, and 61 m are available from the referenced weather stations for different years. Two locations have Pasquill-Gifford weather stability classification data available.

Weather Stations:

Station: KRDU - Raleigh-Durham Airport (RDU)

Date of first observation: July 1948 | Station type: ASOS

Latitude: 35.87764° Longitude: -78.78747° | Located 10 km North West

Elevation: 435 feet above sea level | Wind measurement at 10 m

Supported by: NOAA / National Weather Service

Station: REED - Reedy Creek Field Laboratory

Date of first observation: October 1998 | Station type: ECONET

Latitude: 35.80712° Longitude: -78.74412° | Located 2 km West

Elevation: 420 feet above sea level | Wind measurement at 10 m

Supported by: NC Agricultural Research Service

Station: LAKE - Lake Wheeler Rd Field Lab

Date of first observation: May 1982 — Station type: ECONET

Latitude: 35.72816° Longitude: -78.67981° — Located 6 km South

Elevation: 382 feet above sea level — Wind measurement at 10 m

Supported by: NC Agricultural Research Service

Station: Raleigh State University (JC Ralston Arboretum)

Date of first observation: January 1892 — Station type: COOP

Latitude: 35.794° N Longitude: -78.699° W — Located 3 km West

Elevation: 400 feet above sea level — [no wind data]

Supported by: NOAA / National Weather Service

Station: (CAMP) Jordan Hall NC State University

Date of first observation: September 2014 — Station type: COOP

Latitude: 35.7822° N Longitude: -78.67648° W — Located 1 km South West

Elevation: 450 feet above sea level — Wind measurement at 30 m

Supported by: NOAA / National Weather Service

Station: HAR – Shearon Harris Nuclear Power Plant

Date of first observation: March 1973 — Station type: ASOS

Latitude: 35.588° N Longitude: -78.939° W — Located 35 km South West

Elevation: 260 feet above sea level — Wind measurement at 12m and 61m

Supported by: Shearon Harris Nuclear Power Plant (HNP)

Station Durham 11W (Duke Forest)

Date of first observation: March 2000 — Station type: RAWS

Latitude: 35.971° N Longitude: -79.093° W — Located 46 km North West

Elevation: 565 feet above sea level — Wind measurement at 10 m

Supported by: NOAA / US Climate Reference Network

ASOS Station (RDU and HAR):

NWS Automated Surface Observing System: Hourly weather conditions are recorded by these automated sensors. ASOS arrays are often located at airports. Parameters include air temperature, humidity, winds, precipitation, visibility, and pressure. Sensors are maintained by National Weather

Service. ASOS sensors were introduced beginning in 1996. Observations are generally available each hour.

ECONET Stations (REED and LAKE):

North Carolina Environment and Climate Observing Network: The NC ECONet combines sensors from several networks into a single comprehensive database. SCO stations record hourly weather and environmental conditions are recorded by automated sensors. Parameters include air temperature, humidity, winds, precipitation, pressure, solar radiation, soil temperature, and soil moisture. Sensors are maintained by the State Climate Office. Hourly data for these sensors began in 1996. Observations are generally available each hour.

COOP Station (Raleigh State University):

NWS Cooperative Observer: Daily air temperatures and precipitation are recorded by volunteers. Sensors are maintained by National Weather Service. COOP data make up the bulk of the historical climate record - several stations in NC have records for over 100 years. Current observations are available for many stations on the next day. Some stations only have updated observations 3-6 months later.

RAWS Station (Durham 11 W - Duke Forest):

This weather station is part of the US Climate Reference Network (USCRN) developed by the National Oceanic and Atmospheric Administration (NOAA). The USCRN's primary goal is to provide long-term temperature, precipitation, and soil moisture and temperature observations that are of high quality and are taken in stable settings.

Every USCRN observing station is equipped with a standard set of sensors, a data logger, and a satellite communications transmitter. Some of the measured parameters (e.g. temperature, precipitation, and soil conditions) have multiple sensors for redundancy and independent validation. Remote Automated Weather Stations (RAWS) are weather stations set up on tripods.

The data collected from these stations are used in numerous applications, including fire weather, climatology, resource management, flood warning, noxious weed control, all-risk management, and air quality management. The solar-powered units gather important weather information on an hourly basis. RAWS units collect, store, and forward data hourly (via satellite).

Table 2-11 – Parameter Metadata for ASOS Station at RDU

Parameter	Sensor	Range	Accuracy	DSR	DPR	DAP	DRR
Air Temperature	resistive temperature device (RTD)	-80°F to 130°F (-62°C to 54°C)	RMSE: 1.1°F - 7.9°F	1min	1min	5min	60min
Dew Point	chilled mirror	-30°F to 86°F (-34°C to 30°C)	RMSE: 1.1°F - 7.9°F	1min	1min	5min	60min
Cloud Level	laser celiometer	100ft - 12,000 ft	+/- 100 ft or 5% (whichever is greater)	12sec	30sec	30min	60min
Visibility	forward scatterometer	<0.25 - 10mi+	+/- 0.25mi - 1.25mi; up to +/- 2 reportable increments between 4mi and 10mi	1min	1min	10min	60min
Weather	precipitation identifier, vibrating element (FZRA)	Detects Light, Moderate, and Heavy Rain or Snow; or Mixed Precipitation, Freezing rain	Solid: 99% detected, 97% correctly identified; Liquid: 99% detected, 90% correctly identified	1min	1min	10min	60min
Obscuration	Derived from air temp, dew point, & Visibility	only when visibility < 7mi	Reports haze and fog if within range	Deriv ed	Deriv ed	Deriv ed	60min
Pressure	redundant pressure cells	572mb to 1067mb	+/- 0.7mb	1min	1min	1min	60min
Wind Direction	vane	0 - 359 degrees	+/- 5 degrees @ speed > 5kt	5sec	5sec	2min	60min
Wind Speed	cup anemometer	0 - 125 kt	+/- 2kt or 5% (whichever is greater)	5sec	5sec	2min	60min
Wind Gust	cup anemometer	0-125 kt	+/- 2kt or 5% (whichever is greater)	5sec	5sec	5sec	60min
Precipitation	heating tipping bucket	0-10 in/hr	+/- 0.02" or 4% of hourly total (whichever is greater)	1min	1min	60min	60min

Table 2-12 – Parameter Metadata for ECONet Stations

Parameter	Sensor	Range	Accuracy	DSR	DPR	DAP	DRR
Air Temperature	ceramic capacitance	-52°C to 60°C (-60°F to 140°F)	+/- 0.2°C at -40°C, +/- 0.3°C at 20°C, +/- 0.4°C at 40°C	1min	1min	1min	30min
Relative Humidity	film capacitance	0-100%	+/- 3% @ 0%-90%, +/- 5% @ 90%-100%	1min	1min	1min	30min
Pressure	silicon capacitive pressure	600mb-1100mb	+/- 0.5mb @ 0°C-30°C, +/- 1mb @ -52°C- 60°C	1min	1min	1min	30min
Wind Direction	potentiometer vane	360 degree mechanical, 0-355 degree electrical	+/- 3 degrees	1min	1min	1min	30min
Wind Speed	helicoid propeller	sustained 0-116 kt (60m/s)	+/- 0.5 kt	5sec	1min	1min	30min
Wind Gust	helicoid propeller	gusts:0-191 kt (0-100m/s)	+/- 0.5 kt	5sec	1min	1min	30min
Precipitation	tipping bucket rain gage	infinite in increments of tips	within +/-1% @ <1 in/hr, -3% @ 1-2 in/hr, -5% @ 2-3in/hr	instant	instant	1min	30min
Soil Moisture	Theta MLx probe	0 - 0.5 m^3/m^3	within +/- 1%	1min	1min	1min	30min
Soil Temperature	thermistor	-35°C to 50°C	+/- 0.5°C	1min	1min	1min	30min
Solar Radiation	photovoltaic detector	0 - 3000 W/m^2	within +/- 1%	1min	1min	1min	30min
Photosynthetically Active Radiation	silicon photodetector	0 - 10000 micromole/s m^2	within +/- 1%	1min	1min	1min	30min

Table 2-13 – Description of Parameters for Weather Stations

Parameter	**Description**
Sensor	The instrument used to receive and respond to a parameter-specific stimulus.
Range	Signifies the minimum and maximum values that an instrument is capable of measuring.
Accuracy	The amount of possible deviation from the true value when sensor is properly calibrated and working efficiently.
DSR Data Sampling Rate	How often an instrument on an observing network samples its parameter.
DPR Data Processing Rate	The time interval over which the samples are averaged. *Note: Most instruments do not average their samples.
DAP Data Accumulation Period	The time period over which the samples, or averaged samples, are stored before being reported.
DRR Data Reporting Rate	How often the data is incorporated into the database and can be viewed by the public.

Table 2-14 – HNP/HAR Meteorological Monitoring Tower Sensor Elevations

Sensor	**Approximate Elevation Above Tower Base (m)**
Wind Speed and Direction	12 and 61
Dew Point	12
Solar Radiation	1.5
Ambient Temperature	12 and 61
Delta-Temperature[1]	12 and 61
Precipitation	1.5
Barometric Pressure	1.5
[1]Used to measure differential temperature channel between these elevations.	

The equipment at HAR is checked and calibrated on a routine basis and in accordance with NRC guidance. Accumulated system data are routinely analyzed for inconsistent or erratic data, including a comparison with appropriate meteorological data obtained from other local or regional meteorological observation stations.

In order to achieve the required level of system reliability (i.e., annual data recovery targets), the following maintenance program at HNP/HAR is followed:

- Calibrate data logger input channels semiannually.
- Calibrate or replace wind sensors with National Institute for Standards and Technology (NIST)-traceable calibrated sensors semiannually.
- Calibrate precipitation monitoring device (rain gauge) semiannually.
- Calibrate or replace barometric pressure, dew-point temperature, and solar radiation channel sensors with NIST-traceable calibrated sensors annually.
- Check the two ambient/differential temperature channels for deviations. Temperature sensors are thermistors purchased with NIST-traceable calibration documentation. Thermistors are inherently stable (100-month drift less than 0.01°C) and routine sensor calibration or replacement is therefore not necessary. Deviation between the two ambient/differential temperature channels provides an early warning of a problem with one of these channels.
- The guy wires and the tower anchors are inspected prior to instrument maintenance and calibration events on a semiannual basis.

The current monitoring system at HNP/HAR is compliant with the requirements of NRC Regulatory Guide 1.23, Revision 1.[2-29]

Table 2-15 – HNP/HAR Meteorological Monitoring Tower Accuracy of Monitored Parameters

Monitored Parameter	Basis	Accuracy Criteria
Wind Direction 0 – 360 degrees (12 and 61 m)	NRC Regulatory Guide 1.23 Revision 1	±5 degrees. Starting threshold < 1 mph. Resolution to 1.0°
Wind Speed 0 – 90 mph (12 and 61 m)	NRC Regulatory Guide 1.23 Revision 1	±0.45 mph or 5% of observed wind speed. Starting threshold < 1 mph. Resolution to 0.1 mph°
Ambient Temperature -4°F – 104°F (12 and 61 m)	NRC Regulatory Guide 1.23 Revision 1	±0.9°F Resolution to 0.1°F
Differential Temperature -108°F – 108°F (calculated)	NRC Regulatory Guide 1.23 Revision 1	±0.18°F Resolution to 0.01°F
Wet-Bulb Temperature	NRC Regulatory Guide 1.23 Revision 1	±0.9°F Resolution to 0.1°F
Relative Humidity/Dew Point 0 – 100%	NRC Regulatory Guide 1.23 Revision 1	Relative humidity: ±4% Resolution to 0.1% Dew point: ±2.7°F Resolution to 0.1°F
Total Precipitation	NRC Regulatory Guide 1.23 Revision 1	Precipitation (water equivalent). ±10% for a volume equivalent to 0.1 inches of precipitation at a rate of < 2 inches per hour. Resolution 0.01 in.
Solar Radiation[1]	ANSI/ANS 2.5-1984	Consistent with state-of-the-art.
Barometric Pressure 26.06 – 32.58 inHg	ANSI/ANS 2.5-1984	Consistent with state-of-the-art.
Data logger Sampling Rate	NRC Regulatory Guide 1.23 Revision 1	At least once every 5 seconds
Time	NRC Regulatory Guide 1.23 Revision 1	±5 minutes. Resolution to ±1 minute.
[1]There are no accuracies specified in RG 1.23 for these parameters. ANSI/ANS 2.5-1984 guidance reflects industry and regulator-accepted state-of-the-art specifications.		

Durham 11W (Duke Forest) Weather Station:

Off-the-shelf commercial equipment and sensors used are based on performance and durability.

Temperature sensors are placed on a typical 3 meter (10 ft.) instrument tower at 1.5 meters above the ground surface. The instrument system is designed to accommodate additional sensors on the tower without disrupting the physical site.

Hourly and sub-hourly values are stored in a data logger attached to the tower. A geostationary satellite transmitter sends the data to the National Centers for Environmental Information where the data undergo a quality control check and are placed on the internet continuously as the data arrives. Highly accurate measurements and reliable reporting are critical. Instruments are calibrated or verified annually, and maintenance includes routine replacement of aging sensors. The performance of each station's measurements is monitored daily. Problems are addressed quickly, typically within days.

Table 2-16 – Durham 11W (Duke Forest) Weather Station Measurements and Sensors

Parameter	Description
Air Temperature	Three platinum resistance thermometers housed in fan aspirated solar radiation shields
Precipitation	An inlet-heated, wind-shielded weighing rain gauge (configured with three load cell sensors), precipitation (wetness) detector, and an auxiliary tipping bucket gauge.
Wind Speed	A 3-cup anemometer at the same height as the air temperature shield intakes.
Solar Radiation	A silicon pyranometer
Surface (Skin) Temperature	A precision infrared temperature sensor pointed at the ground surface
Relative Humidity	A capacitive thin-film polymer humidity sensor providing accurate and stable measurement even in environments with high humidity
Soil Temperature & Moisture	Moisture sensors with built-in thermistors installed at specific depths: 5, 10 20, 50 and 100 cm.

Data

Average and extreme data on the following parameters is given for various periods as available from the referenced weather stations:[2-15]

1. Surface wind speed
2. Surface wind direction
3. Surface temperature
4. Precipitation (rainfall and snow)
5. Barometric pressure
6. Relative humidity
7. Dew point

Wind rose data were similar for the weather stations reviewed. Most had peak sector frequencies of less than 15% for 16 sectors and average wind speeds of 2 to 4 m/s.[2-15,2-22]

The wind rose data closest to the reactor site was taken from the CAMP weather station (Jordan Hall) at NCSU. CAMP wind data is measured at 30 m, which is the same height as the reactor stack. Data from Sep 2014 to Sep 2016 was used.

Inversion and mixing depth data reported by Harris Nuclear Plant for Greensboro, NC are given in Table 2-18 and Table 2-19.[2-27]

Wind rose data (wind speed and direction) for various consecutive year periods for six locations are shown in Figure 2-22 through Figure 2-33 and Table 2-17.[2-15,2-22]

Inversion and mixing depth data for Greensboro, NC is given in Table 2-18 and Table 2-19, respectively.[2-27]

Weather data is given in Table 2-21 and Table 2-22 and Figure 2-34 for Raleigh, NC.[2-15,2-22]

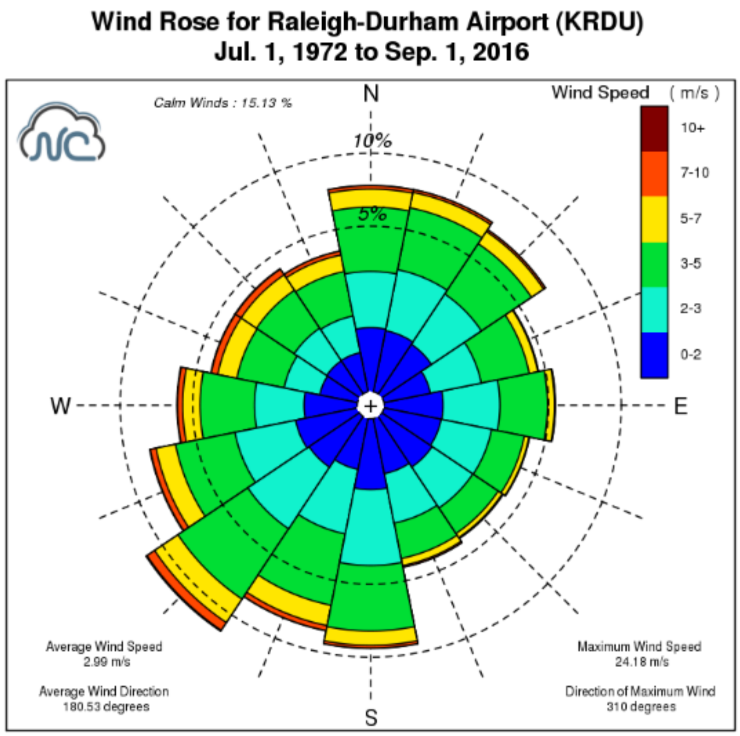


Figure 2-22 – Wind Rose Data for Raleigh-Durham Airport (1972 – 2016)

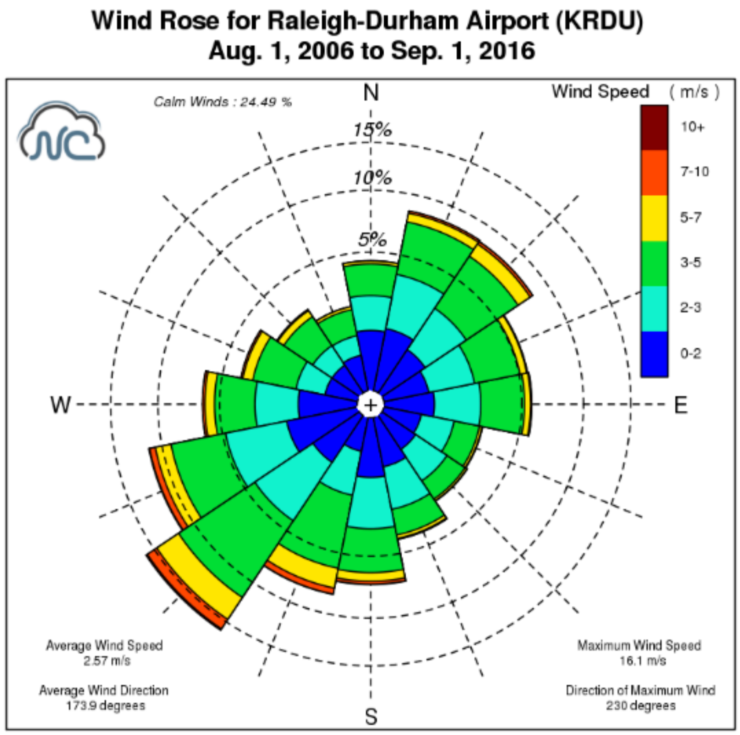


Figure 2-23 – Wind Rose Data for Raleigh-Durham Airport (2006 – 2016)

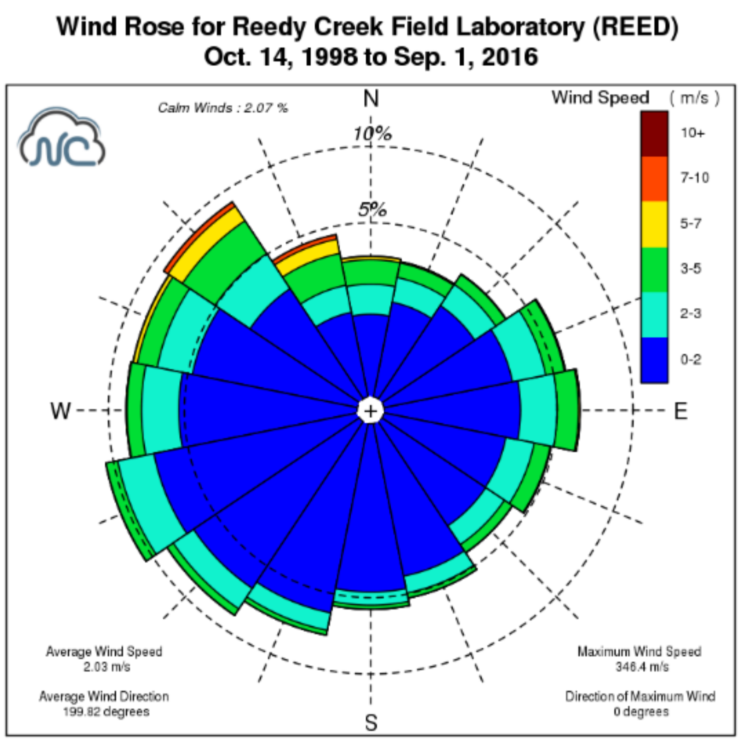


Figure 2-24 – Wind Rose Data for Reedy Creek Field Laboratory (1998 – 2016)

Wind Rose for Reedy Creek Field Laboratory (REED)
Sep. 1, 2006 to Sep. 1, 2016

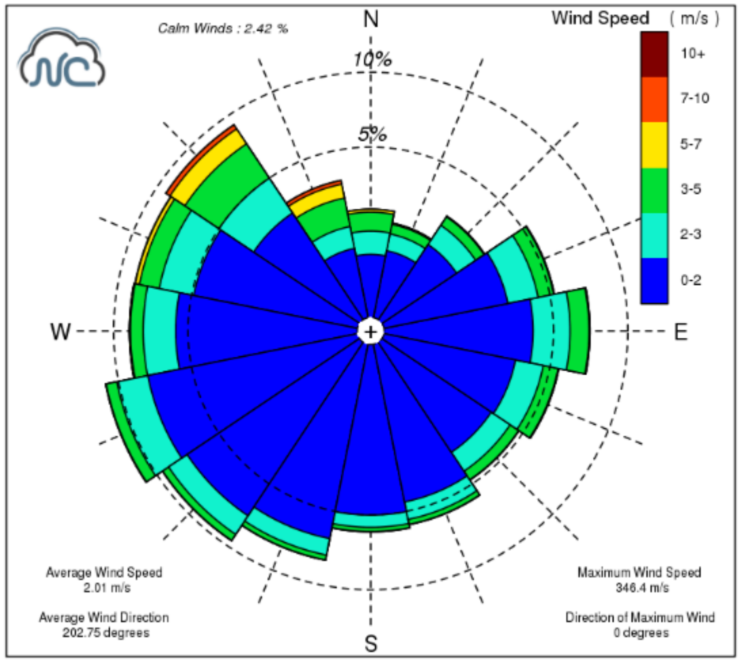


Figure 2-25 – Wind Rose Data for Reedy Creek Field Laboratory (2006 – 2016)

Wind Rose for Lake Wheeler Rd Field Lab (LAKE)
May. 12, 1982 to Sep. 1, 2016

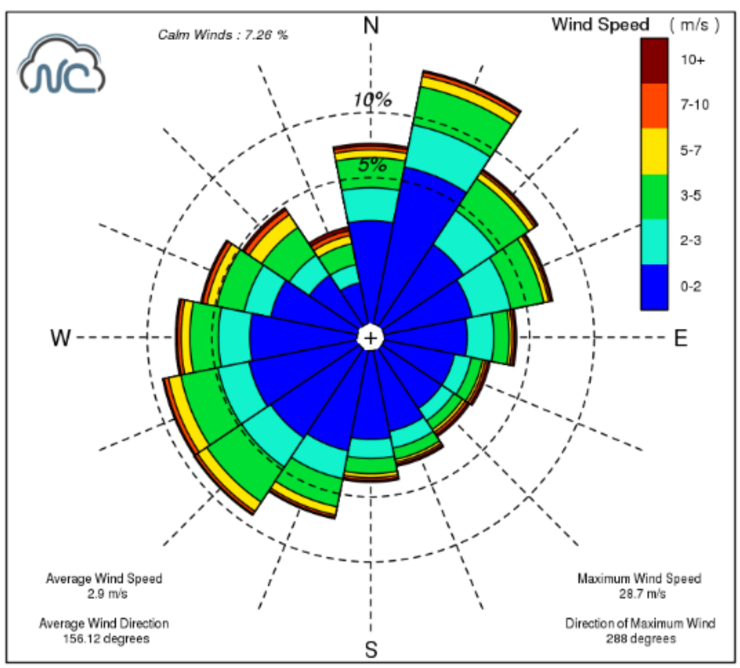


Figure 2-26 – Wind Rose Data for Lake Wheeler Road Field Laboratory (1982 – 2016)

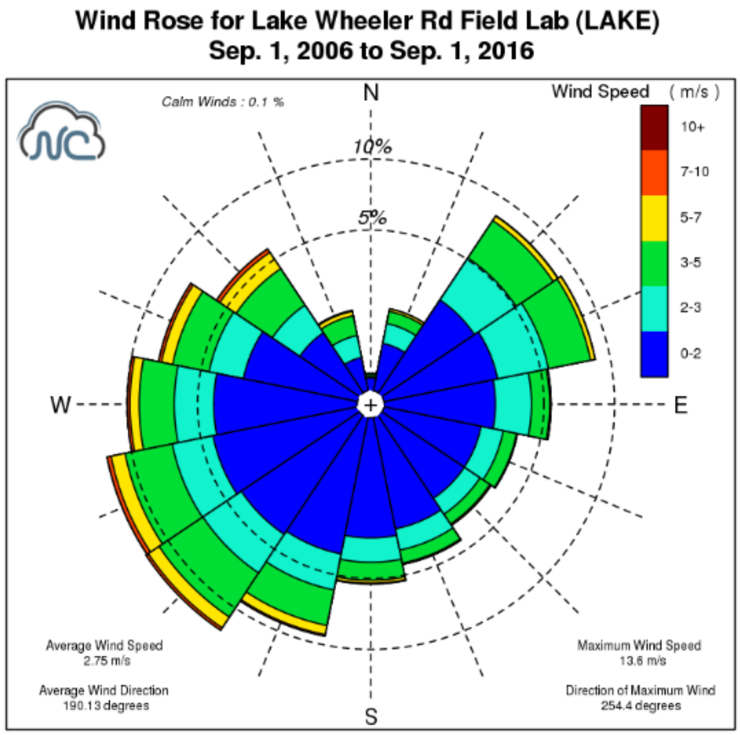


Figure 2-27 – Wind Rose Data for Lake Wheeler Road Field Laboratory (2006 – 2016)

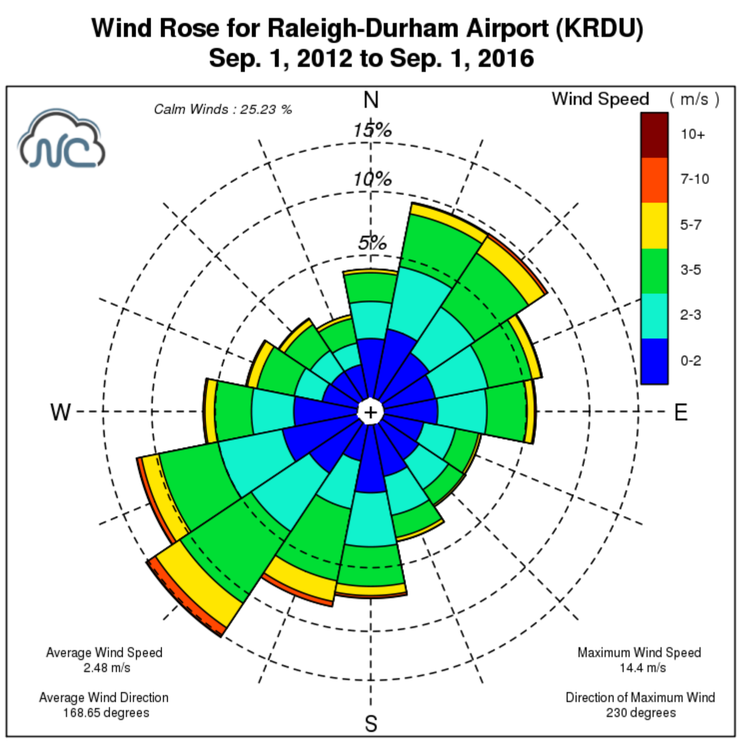


Figure 2-28 – Wind Rose Data for Raleigh-Durham Airport (2012– 2016)

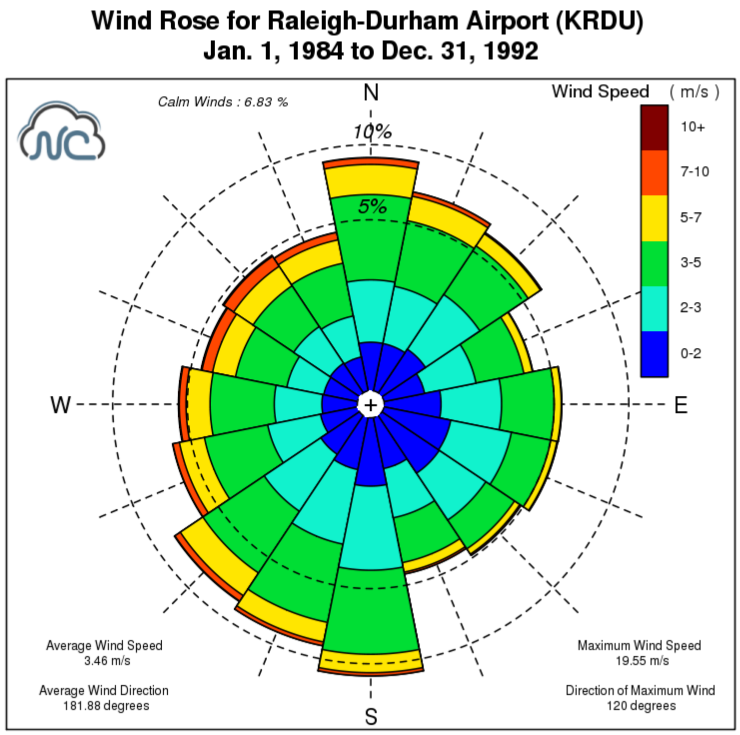


Figure 2-29 – Wind Rose Data for Raleigh-Durham Airport (1984– 1992)

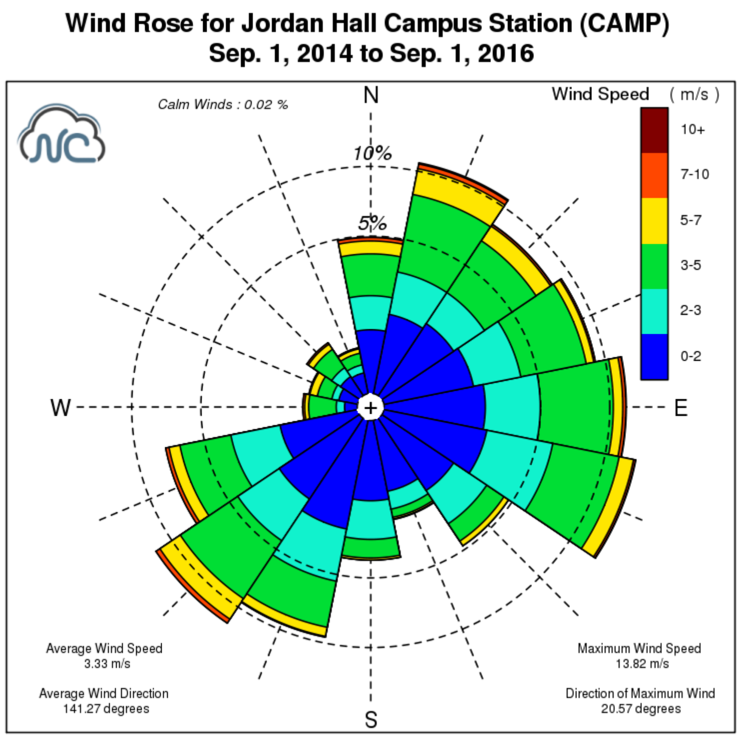


Figure 2-30 – Wind Rose Data for Jordan Hall Campus Station (2014– 2016)

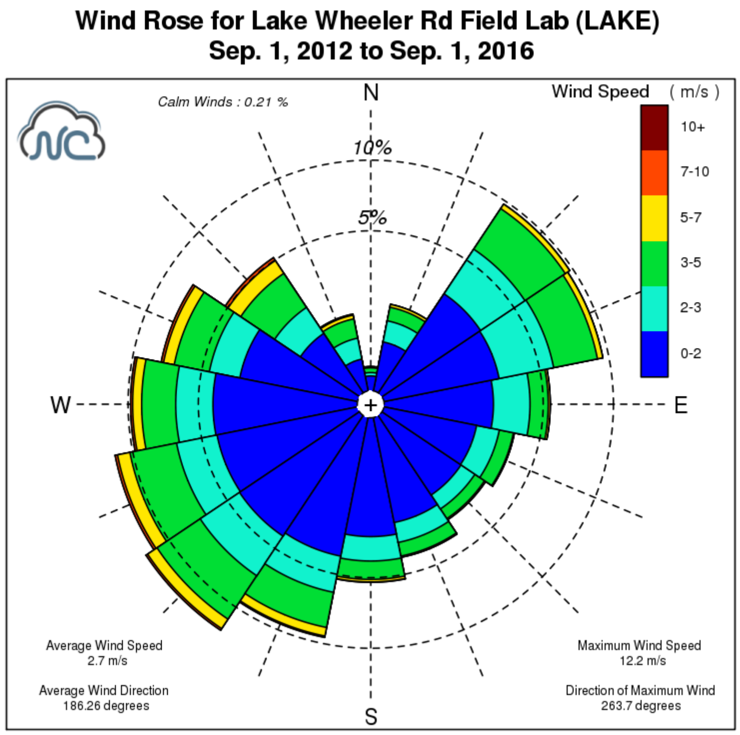


Figure 2-31 – Wind Rose Data for Lake Wheeler Road Filed Laboratory (2012– 2016)

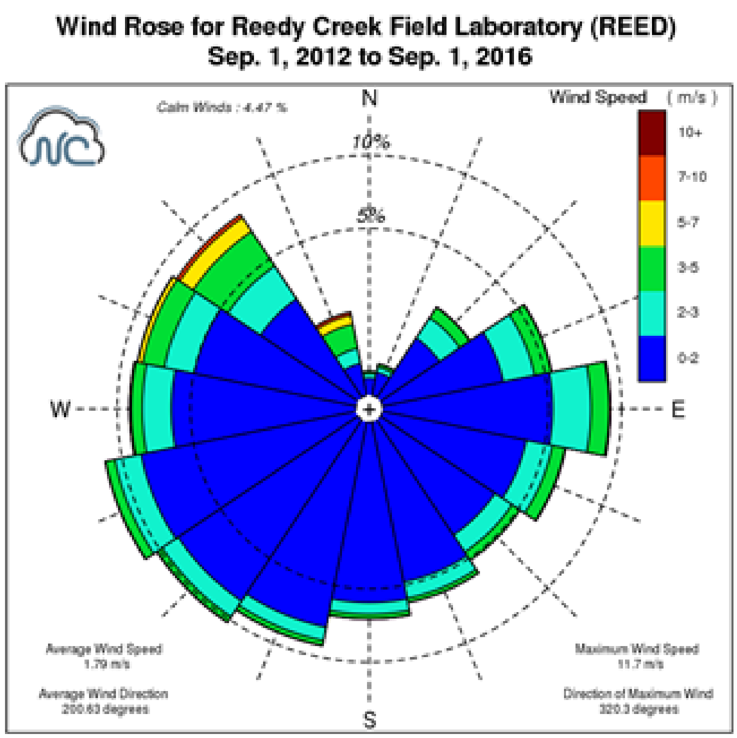


Figure 2-32 – Wind Rose Data for Reedy Creek Filed Laboratory (2012– 2016)

Table 2-17 – Wind Rose Data for HAR Weather Station (1994 – 1999)

12 m Tower 1994 – 1999 Frequency Distribution																	
mph	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
1 to 3	6.43	7.01	4.95	3.02	2.08	1.63	1.51	1.63	2.22	2.86	2.48	2.34	1.93	2.25	6.43	7.01	48.16
4 to 7	4.24	3.34	2.45	2.11	1.25	1.14	1.45	2.12	3.24	3.09	2.95	3.83	1.59	1.6	4.24	3.34	39.59
8 to 12	0.67	0.78	0.5	0.19	0.09	0.09	0.13	0.24	0.65	0.68	0.83	1.38	0.39	0.85	0.67	0.78	9.51
13-18	0.05	0.01	0	0	0.01	0	0.01	0.02	0.01	0.01	0.05	0.12	0.03	0.06	0.05	0.01	0.45
19-24	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0.01
> 24	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
Total	11.39	11.14	7.89	5.32	3.42	2.86	3.09	4.02	6.13	6.63	6.31	7.67	3.95	4.75	5.82	7.32	97.72

61 m Tower 1994 – 1999 Frequency Distribution																	
mph	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW	Total
1 to 3	0.49	0.56	0.42	0.49	0.48	0.49	0.5	0.58	0.59	0.83	0.93	0.7	0.57	0.62	0.71	0.53	9.5
4 to 7	2.1	2.42	2.22	2.13	1.61	1.56	1.65	1.73	2.87	3.29	3.31	3.27	1.82	1.48	2.05	2.04	35.57
8 to 12	3.52	4.49	2.88	2.43	1.61	1.17	1.26	1.61	2.75	3.92	4.39	4.11	1.79	1.73	2.24	2.49	42.4
13-18	1.03	0.81	0.6	0.28	0.11	0.07	0.1	0.23	0.66	1.36	1.37	1.31	0.4	0.71	0.84	0.96	10.84
19-24	0.08	0.12	0.04	0.01	0.01	0	0.01	0.04	0.09	0.14	0.2	0.23	0.07	0.11	0.08	0.08	1.29
> 24	0.03	0	0	0	0	0	0	0	0	0	0.03	0.04	0	0.01	0	0	0.13
Total	7.26	8.4	6.17	5.35	3.81	3.29	3.53	4.2	6.97	9.54	10.22	9.67	4.66	4.65	5.93	6.1	99.73

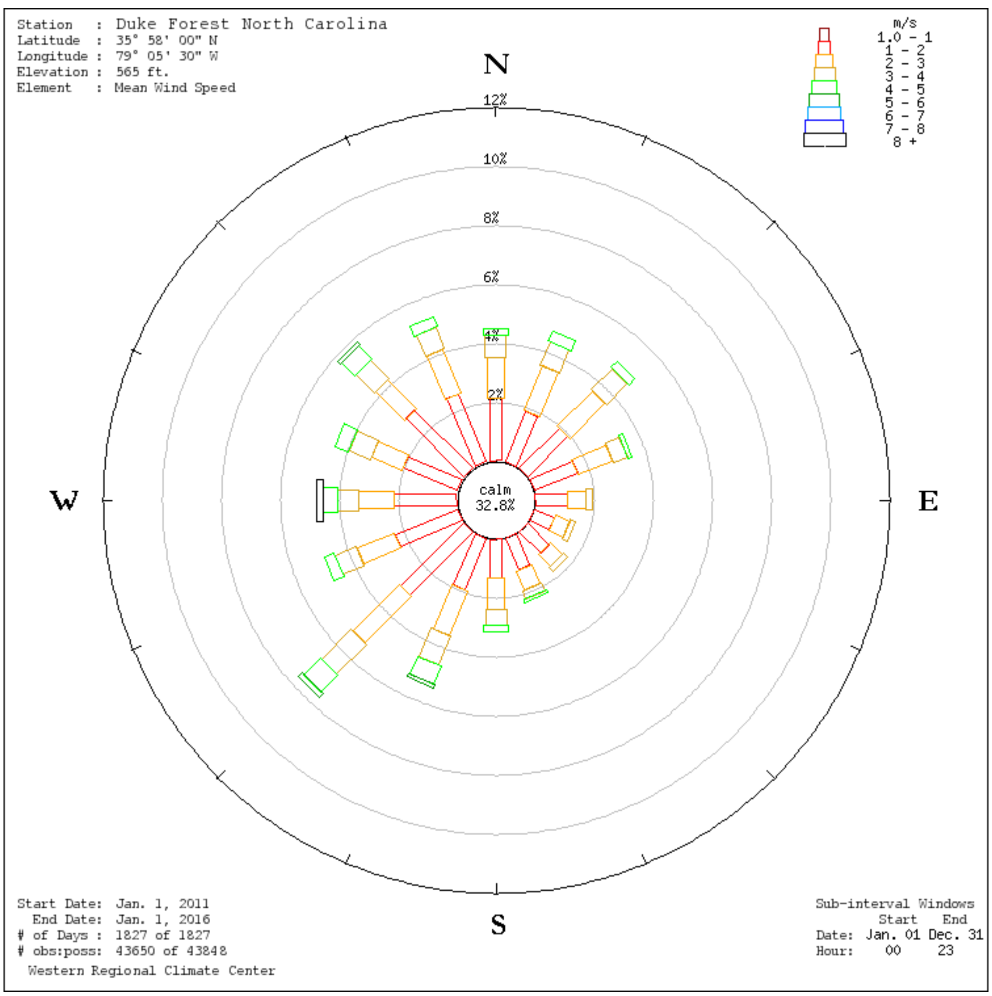


Figure 2-33 – Wind Rose Data for Durham 11W (Duke Forest) (2011– 2015)

Table 2-18 – Seasonal Percent Frequency of Inversion below 152 m (500 ft.) in Greensboro, NC

Season	300 GMT	1500 GMT	0 GMT	1200 GMT	All Times
Winter	73	15	58	72	43
Spring	70	3	13	66	32
Summer	78	1	11	66	33
Fall	74	4	52	74	40

Table 2-19 – Mean Monthly Mixing Depth in Greensboro, NC

Month	Mean Monthly Mixing Depth (m) Greensboro, NC
January	390
February	650
March	1130
April	1180
May	1530
June	1790
July	1490
August	1420
September	1370
October	1020
November	840
December	580

Wind and Precipitation

Upper Air Movement

Although the immediate problem of diffusion in this area does not extend to the air above 5,000 feet, this data is included since upper air movement and surface patterns are not entirely separable.

Monthly averages of the mean geostrophic wind at the 700 mbar constant pressure surface (approximately 10,000 feet) are given in Table 2-20:

Table 2-20 – Mean Geostropic Winds at 700 mb Over Raleigh, North Carolina

Month	Direction	Speed mph
January	west	38
February	west	20
March	west	29
April	west	29
May	west	18
June	west	17
July	northwest	21
August	southwest	18
September	west	19
October	west	24
November	west	24
December	west	26

Table 2-21 – RDU Airport Monthly Weather Data Summary

Month-Year	Average Daily Temp at 2m (°C)	Average Daily Dew Point at 2m (°C)	Average Daily Pressure at 2m (mb)	Average Daily Relative Humidity at 2m (percent)	Sum of Daily Precipitation at 2m (cm)
Sep-12	21.3	16.3	1002.2	75	19.9
Oct-12	15.7	10.5	999.4	74	4.7
Nov-12	8.4	1	1005.8	63	1.4
Dec-12	9.4	3.4	1001.8	69	7.5
Jan-13	6.7	1.1	1006.2	71	7.9
Feb-13	5.6	-1.9	1001	62	10.4
Mar-13	7.3	-2.3	999.7	55	7.5
Apr-13	15.8	8	1003.9	63	10.6
May-13	19	12.9	1002.8	71	9.8
Jun-13	23.8	18.8	999.5	76	25.6
Jul-13	25.5	21	1002.9	78	8.7
Aug-13	24	19.5	1002.4	78	12.4
Sep-13	21.1	15.5	1001.6	73	7.4
Oct-13	16.4	11.7	1002.6	76	3.6
Nov-13	8.9	1.4	1007.5	63	7.6
Dec-13	7.8	2	1005.2	70	15
Jan-14	2.5	-6.2	1002.5	56	5
Feb-14	6	-1.6	1001.7	62	7.6
Mar-14	7.8	-0.8	1000.8	60	12.9
Apr-14	16	6.7	1000.7	58	13.3
May-14	21	13.4	1001.9	64	10.2
Jun-14	24.8	18	1000.6	68	8.4
Jul-14	25.4	18.5	1001.1	68	22.8
Aug-14	23.9	18.9	1001.1	75	17.5
Sep-14	21.8	17.3	1003.2	77	15
Oct-14	17	10.6	999.7	69	5.5
Nov-14	8.2	1.1	1003.9	64	9.6
Dec-14	7	1.6	1005.3	71	12.7
Jan-15	4.3	-3.7	1005.2	60	8.5
Feb-15	1.6	-7.5	1003.3	55	7.3
Mar-15	10.5	2.1	1004.7	61	8.3
Apr-15	16.5	9.1	1000.6	66	13.2
May-15	21.5	14.9	1005.4	68	7.7
Jun-15	26	19.6	1000.2	71	16.4
Jul-15	26.4	20.8	999	73	12.7
Aug-15	25.1	18.5	1000.6	69	12.2
Sep-15	22.6	18.1	1001.6	78	12.2
Oct-15	15.4	10.8	1002.7	77	11.1
Nov-15	12.8	7.8	1006.8	75	18.2
Dec-15	13.1	9.5	1004.7	81	14
Jan-16	3.9	-3.1	1001.9	63	4.4
Feb-16	6.7	-0.9	1002.1	62	11.9
Mar-16	14.6	6.5	1001.9	62	10.3
Apr-16	15.9	6.4	1002.1	57	7.6
May-16	19.5	14.4	999.9	75	15.7
Jun-16	24.7	18	1000	69	15.1
Jul-16	27.2	22	1000.9	75	20.6
Aug-16	27.2	21.5	1002.7	73	7.7
Sep-16	23.1	17.6	1002.7	75	4

Table 2-22 – NC State University Average Monthly Weather Data – 1921 to 2012

Parameter	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec	Annual
Average Max. Temperature (F)	51.0	54.0	62.1	71.8	79.1	86.1	88.9	87.3	81.9	71.9	62.5	53.0	70.8
Average Min. Temperature (F)	31.6	33.0	40.0	48.4	57.2	65.2	69.2	67.9	62.2	50.4	41.4	33.5	50.0
Average Total Precipitation (in.)	3.58	3.43	3.98	3.11	3.82	4.13	4.81	4.61	4.23	3.15	3.08	3.33	45.26
Average Total Snowfall (in.)	1.9	1.3	0.8	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.7	4.7
Average Snow Depth (in.)	0	0	0	0	0	0	0	0	0	0	0	0	0

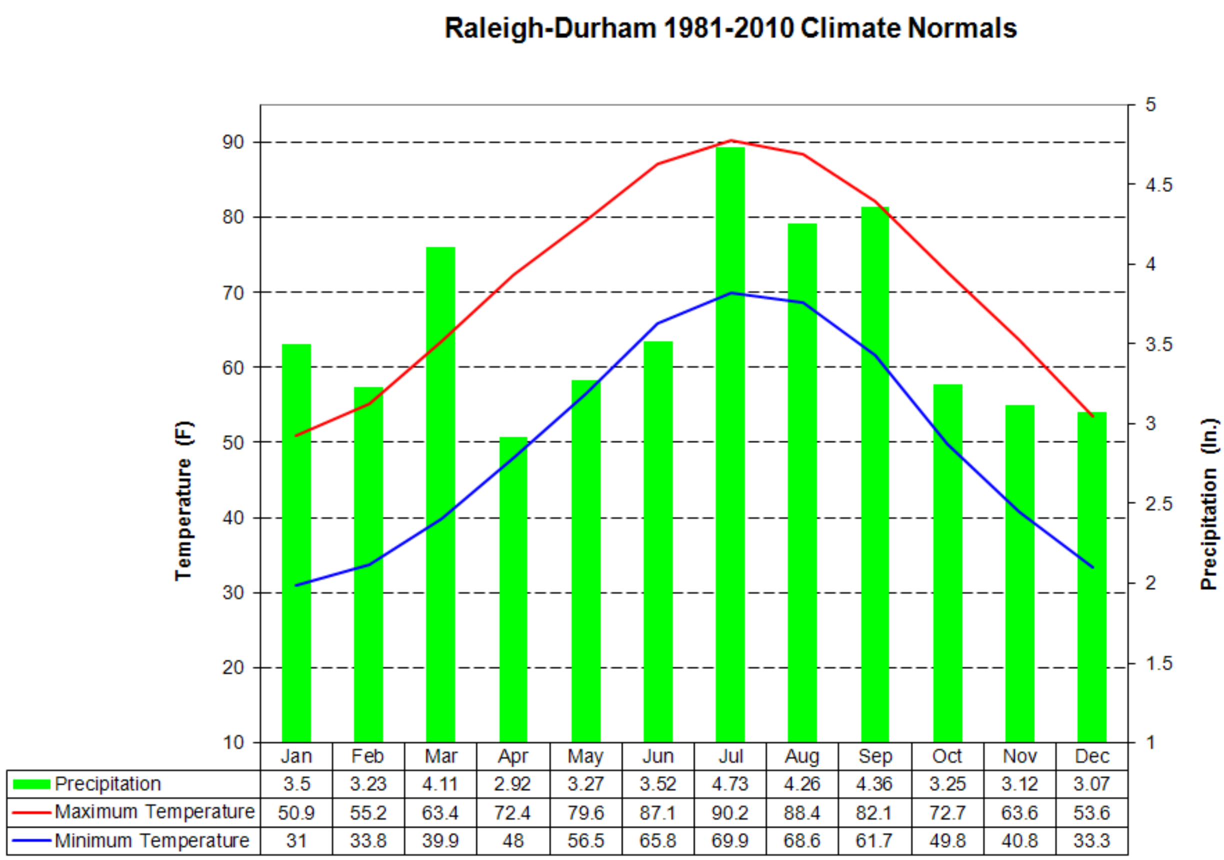


Figure 2-34 – Raleigh-Durham Climate Normals – 1981 - 2010

Atmospheric Stability

Pasquill-Gifford (PG) stability classes A through F are used in the Gaussian plume model for atmospheric releases of radioactivity. PG stability classes are available from HAR and Durham 11W. PG stability class distribution is shown in Table 2-23 through Table 2-25.

Table 2-23 – HAR Weather Station PG Stability Class Data

Year	Class A %	Class B %	Class C %	Class D %	Class E %	Class F* %
1976 – 1978	6.5	4.3	5.4	27.0	24.3	32.5
1994 – 1999	1.5	2.9	5.4	32.0	28.6	29.6
Average	4.0	3.6	5.4	29.5	26.5	31.1
1994 – 1999 wind speed m/s	5.3	4.5	4.2	3.7	4.1	3.3

*NOTE: Data for Class F is the sum of classes F and G

Wind speed was estimated is this analysis as the midpoint of each wind speed range reported up to 24 mph. 24 mph was used as the upper value.

Table 2-24 – HAR Weather Station Wind Speed Data

mph	**Midpoint m/s**
1 to 3	0.89
4 to 7	2.46
8 to 12	4.47
13 to 18	6.93
19 to 24	9.61
24	10.73

Durham 11W (Duke Forest) PG Stability Class Distribution:

PG stability class data is available from the Durham 11W (Duke Forest) weather station using the NOAA / USCRN algorithm based on solar radiation and temperature difference.

Table 2-25 – Durham 11W (Duke Forest) PG Stability Class Distribution Fractions

5 Year Average – 2011 - 2015		
PG Class	**Percent**	**m/s**
A	0.00	
B	0.00	
C	0.00	
D	26.69	3.90
E	11.18	2.20
F	61.67	1.10

RDU Estimated PG Stability Class Distribution:

Hourly meteorological data reported by the US EPA Center for Exposure Assessment Modeling (CEAM) for the RDU airport from 1 Jan 1984 to 31 Dec 1992 were analyzed to determine the PG stability class distribution. The Turner method for determining PG stability categories from data that are routinely collected at National Weather Service (NWS) stations was used.[2-28,2-30,2-45]

The Turner method estimates the effects of net radiation from solar altitude (a function of time of day and time of year), total cloud cover, ceiling height, and wind speed on stability. Solar altitude applies to day time data only. The net radiation index in the day time is related to the insolation value, cloud cover, and ceiling height. At night, the net radiation index is dependent on the cloud cover and ceiling height. The ceiling height was not consistently reported in the CEAM data for RDU. Net radiation index for ceiling heights above and below 7000 feet were averaged using the Turner method Procedure. The PG stability class is determined using the wind speed and net radiation index.[2-30]

Results calculated and are shown in Table 2-26 through Table 2-29 and Figure 2-35 through Figure 2-37.

Table 2-26 – RDU Airport PG Stability Class Data

PG Stability Class		A	B	C	D	E	F
9 Year Average	%	1.41	8.78	12.73	28.51	16.30	32.27
9 Year Average	Wind, m/s	2.0	2.8	4.0	5.2	3.3	1.8

Table 2-27 – RDU Airport Weather Stability Class Distribution for 24 Hours

Year	Value	A	B	C	D	E	F
1984	Total Average	1.16	8.60	12.88	28.64	15.25	33.48
1985	Total Average	1.55	9.78	11.95	27.79	16.54	32.39
1986	Total Average	1.31	9.01	13.07	27.51	16.73	32.38
1987	Total Average	1.16	8.53	13.29	27.63	17.08	32.31
1988	Total Average	1.39	8.17	13.92	24.74	16.91	34.88
1989	Total Average	1.24	7.49	12.85	33.46	17.64	27.32
1990	Total Average	1.72	8.73	12.82	27.95	14.50	34.28
1991	Total Average	1.65	10.11	11.41	27.95	15.67	33.22
1992	Total Average	1.51	8.64	12.39	30.90	16.34	30.22
9 Year	%	1.41	8.78	12.73	28.51	16.30	32.27
9 Year	Wind, m/s	2.0	2.8	4.0	5.2	3.3	1.8

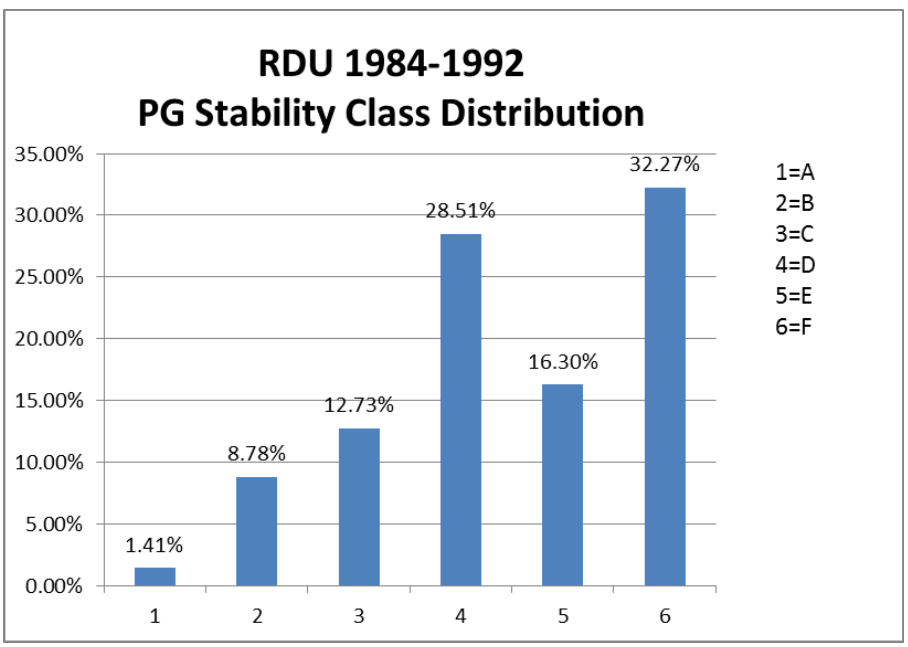


Figure 2-35 – RDU Airport PG Stability Class Distribution – 1984 – 1992

Table 2-28 – RDU Airport Weather Stability Class Distribution for Days

Year	Value	A	B	C	D	E	F
1984	Day Average	3.48	25.79	38.63	32.10	0.00	0.00
1985	Day Average	4.64	29.35	35.86	30.15	0.00	0.00
1986	Day Average	3.92	27.04	29.20	29.85	0.00	0.00
1987	Day Average	3.49	25.60	39.86	31.04	0.00	0.00
1988	Day Average	4.17	24.52	41.75	29.56	0.00	0.00
1989	Day Average	3.72	22.47	38.54	35.27	0.00	0.00
1990	Day Average	5.17	26.20	38.46	30.17	0.00	0.00
1991	Day Average	4.95	30.33	34.23	30.50	0.00	0.00
1992	Day Average	4.54	25.91	37.16	32.39	0.00	0.00
9 Year	%	4.23	26.35	38.19	31.23	0.00	0.00

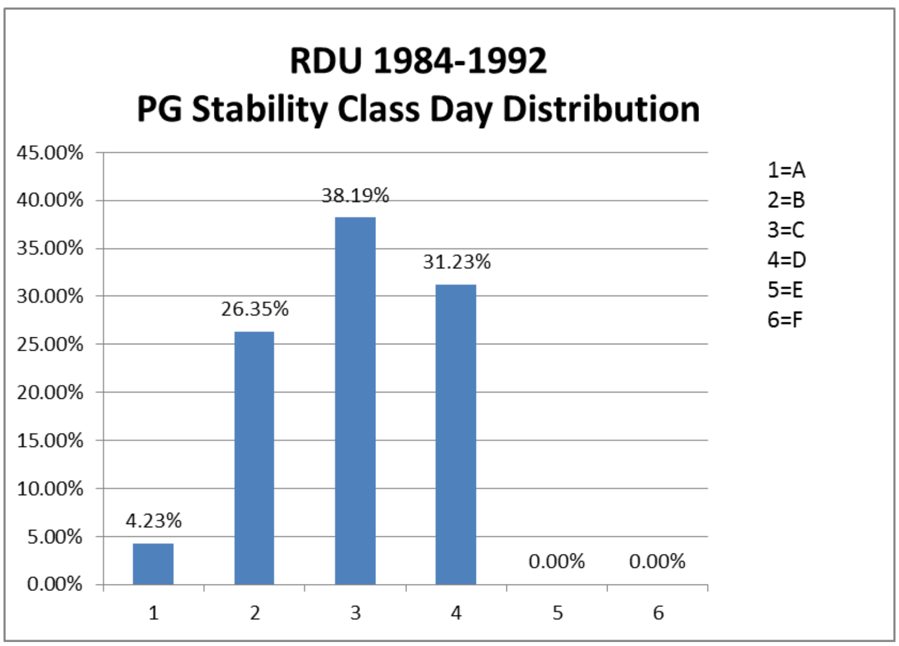


Figure 2-36 – RDU Airport Weather Stability Class Distribution for Days – 1984 – 1992

Table 2-29 – RDU Airport Weather Stability Class Distribution for Nights

Year	Value	A	B	C	D	E	F
1984	Night Average	0.00	0.00	0.00	26.91	22.87	50.21
1985	Night Average	0.00	0.00	0.00	26.60	24.81	48.59
1986	Night Average	0.00	0.00	0.00	26.34	25.09	48.57
1987	Night Average	0.00	0.00	0.00	25.92	25.63	48.46
1988	Night Average	0.00	0.00	0.00	22.33	25.36	52.31
1989	Night Average	0.00	0.00	0.00	32.55	26.46	40.98
1990	Night Average	0.00	0.00	0.00	26.83	21.76	51.41
1991	Night Average	0.00	0.00	0.00	26.67	23.50	49.83
1992	Night Average	0.00	0.00	0.00	30.16	24.51	45.33
9 Year	%	0.00	0.00	0.00	27.14	24.44	48.41

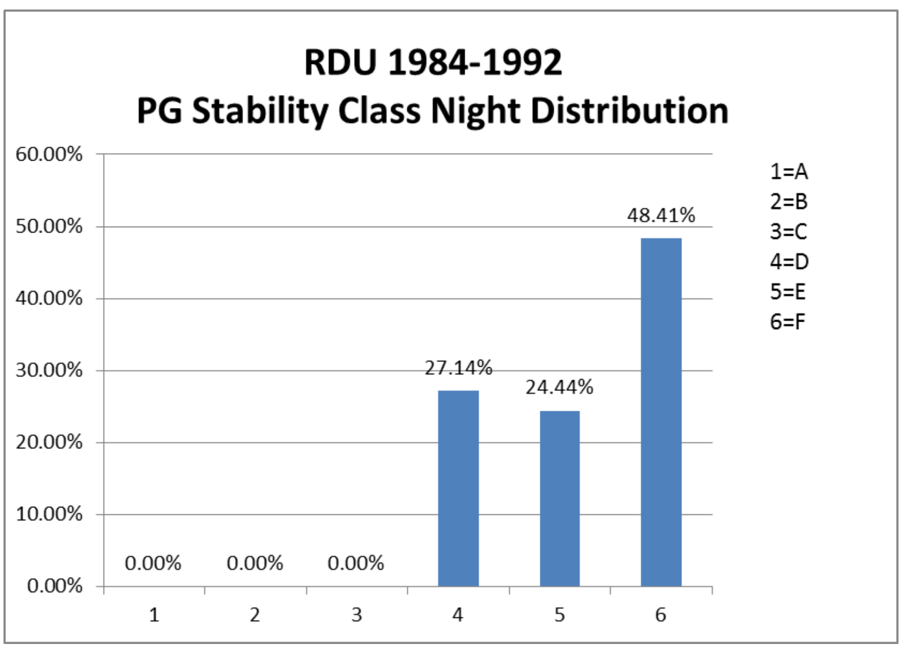


Figure 2-37 – RDU Airport Weather Stability Class Distribution for Nights – 1984 – 1992

PG stability class data for the weather stations used in this report are listed below with average results shown. Default values given in ANSI/ANS 15.7 are also shown for comparison in Table 2-30.

Table 2-30 – Summary of PG Stability Class Data

Station	Parameter	PG Stability Classes					
		A	B	C	D	E	F
HAR	Percent	4.0	3.6	5.4	29.5	26.5	31.1
RDU	Percent	1.4	8.8	12.7	28.5	16.3	32.3
Durham	Percent	0.0	0.0	0.0	26.7	11.2	61.7
Average	Percent	1.8	4.1	6.0	28.2	18.0	41.7
HAR	Wind, m/s	5.3	4.5	4.2	3.7	4.1	3.3
RDU	Wind, m/s	2.0	2.8	4.0	5.2	3.3	1.8
Durham	Wind, m/s	0.0	0.0	0.0	3.9	2.2	1.1
Average	Wind, m/s	2	2	3	4	3	2
ANSI/ANS 15.7	Percent			33.3	33.3		33.3
ANSI/ANS 15.7	Wind, m/s			3	2		2

Since the three regional weather stations reviewed give similar results, the data as analyzed is considered acceptable for the reactor facility. PG stability class data for HAR, Durham 11W (Duke Forest), and RDU airport data are considered to be similar since classes D, E, and F are dominant. RDU airport and HAR data include lower contributions for classes A, B, and C.

For RDU airport, the daytime PG stability was observed to be more unstable than at night as analyzed in this report. Classes B, C, and D are dominant in the day time and classes D, E, and F are dominant at night. It is noted that typically the reactor facility is a day time operation and shut down at night. If the facility operates at night, it typically is also operated during the day.

Using the average PG stability class data for all three weather stations is considered realistic since data from both day and night are included. The average PG stability data is considered to be slightly conservative since the distribution is dominated classes C, D, E, and F.

Based on this analysis, PG stability data for the reactor facility is given in Table 2-31.

Table 2-31 – PG Stability Class Data for the PULSTAR Reactor Facility

Parameter	PG Stability Classes					
	A	B	C	D	E	F
Percent	1.8	4.1	6.0	28.2	18.0	41.7
Wind, m/s	2	2	3	4	3	2

The average PG stability data above are used in evaluating atmospheric dispersion of airborne effluent from routine operations in Section 11.

2.4. Hydrology

Hydrology for the reactor site and surrounding area is discussed in this section. Both surface and ground water sources are used by residents within Wake County, NC. The City of Raleigh water supply uses surface waters. Private and community (subdivision) wells are used throughout Wake County and the surrounding areas.[2-4,2-10,2-31]

2.4.1. Surface Water

Surface water provides the principal water supply to Wake County, in which the reactor site is located and to many surrounding communities. Major local streams and lakes are shown in provided figures.

The natural drainage from the site is to the south into Rocky Branch Creek located in a well-defined valley about 1,700 feet from the reactor. Rocky Branch Creek empties into Walnut Creek at a point about 2.25 miles downstream from Lake Raleigh. Walnut Creek is a tributary of the Neuse River, which empties into the Atlantic Ocean. The Walnut Creek catchment area contains both Lake Raleigh (81 acres) and Lake Johnson (174 acres), which have a combined water storage capacity of 840 million gallons.[2-31]

The only other drainage area in the proximity of the reactor site is the Swift Creek catchment basin, which covers 67 square miles lying south and southwest about 6 to 10 miles from the NCSU campus. Swift Creek flows through Lake Wheeler (650 acres) and Lake Benson (500 acres) to the Neuse River. Lake Benson is located 9 miles south of Raleigh and has a surface area of 490 acres and a storage capacity of 1 billion gallons.[2-31]

The city of Raleigh obtains its drinking water supply from two treatment plants; the E.M. Johnson plant utilizes Falls Lake Reservoir which is on the upper Neuse River and the Dempsey E. Benton facility which utilizes water from Lake Benson and Lake Wheeler which are part of the Swift Creek reservoir system. Lake Raleigh and Lake Johnson serve as recreational facilities. Falls Lake Reservoir is located approximately 12 miles north of the reactor site. The E.M. Johnson plant is located approximately 8 miles north of the reactor site. There is no direct natural drainage from the reactor site to the Falls Lake Reservoir. The E.M. Johnson plant can supply 47 million gallons per day (MGD) with a maximum capacity of 86 MGD. The Dempsey E. Benton facility can supply up to 11.2 MGD and has a maximum capacity of 20 MGD. These two facilities are expected to meet the water needs of the area until 2018. The city of Raleigh plans to expand the E.M. Johnson plant for a maximum capacity of 100 MGD by 2018.[2-31]

Rocky Branch Creek runs more than a mile through the NCSU campus. It drains into Walnut Creek, which is a tributary of the Neuse River. Starting in 2002, restoration and renovations to Rocky Branch Creek were made to stabilize the creek and to improve water quality and aquatic and wildlife habitat. Using natural channel design techniques, the restoration allowed Rocky Branch Creek to meander through a newly created floodplain.[2-32]

Flow rates of the Rocky Branch Creek below Pullen Road in Raleigh, Walnut Creek at Sunnybrook Drive in Raleigh and the Neuse River and Swift Creek near Clayton, NC have been measured by the US Geological Survey. Since the Falls Lake Dam located northwest of Raleigh went into operation in December 1983, the flow of Neuse River has been regulated. Annual data for each are given in Table 2-32.[2-33]

Table 2-32 – Flow Rates for Nearby Creeks and Rivers Near the Reactor Site

	Rocky Branch Pullen Park Raleigh, NC ft^3s^{-1}	Walnut Creek Sunnybrook Dr. Raleigh, NC ft^3s^{-1}	Neuse River Clayton, NC ft^3s^{-1}	Swift Creek Clayton, NC ft^3s^{-1}
Average	2.14	35.0	1018 (1983-2012)	78
Max	3.66	61.2	2052 (1983-2012)	126.1
Min	1.33	19.0	425 (1983-2013)	42.5

NOTE: 1 cubic foot = 7.5 gallons = 28.3 liters

The flow of streams in the Piedmont section of NC can be described as average slow when compared with the rest of the country. The time constant, or surging flow, cannot be applied to stream flow due to widespread variability. A better description would be critical or noncritical. Flow reaches criticality (turbulent rather than laminar) at many times, and often exists as critical.

2.4.2. Ground Water

Regional Ground Water Hydrology

The principal components of the ground water system for a typical area in the Piedmont province of North Carolina are provided in Figure 2-38 through Figure 2-41. Bedrock exposures occur infrequently. Bedrock is typically covered by unconsolidated material that may reach depths of 100 feet. Collectively, this unconsolidated material is composed of saprolite, alluvium, and soil and is referred to as regolith. Saprolite is the clay-rich, residual material derived from in-place weathering of the bedrock.[2-34]

The regolith can be generally divided into three zones; soil zone, saprolite, and a transition zone between the saprolite and unweathered bedrock. The three zones represent stages in the breakdown of bedrock by weathering. The climate in central North Carolina favors the development of a thick and extensive layer of regolith. The soil zone is typically 3 to 8 feet thick and has no resemblance to the parent bedrock. The saprolite zone tends to retain the characteristics to the bedrock from which it is derived. The total thickness of the regolith in the Piedmont averages 52 feet.[2-34]

The transition zone between the unweathered bedrock and overlying saprolite is more permeable than the saprolite. If the transition zone is sufficiently thick, it is the zone through which most of the lateral movement of ground water takes place. The higher permeability of the transition zone to the saprolite is a result of fewer clay size particles. As bedrock starts to weather, hydration of unstable minerals occurs (e.g. feldspars) resulting in expansion of these minerals. This expansion ruptures the bedrock creating minute cracks and voids and increases its permeability. Further weathering of bedrock increases its porosity. An increase in clay and clay sized particles as the bedrock becomes completely weathered to saprolite reduces the interconnections of the openings created in the initial stages of weathering thereby reducing permeability. Permeability of saprolite is on the order of 1×10^{-3} to 1×10^{-5} cm/s. Fractures in the bedrock are characterized by slightly higher permeability.[2-34]

Porosity of the regolith is 35 to 55 percent near land surface but decreases with increasing depth as the degree of weathering decreases. Because of its higher porosity, the regolith acts as a reservoir that slowly feeds water downward into fractures in the bedrock. Fractures are most numerous and the largest openings near the top of bedrock. These fractures are the openings along which water can move. As depth increases, the pressure of overlying material, or lithostatic pressure, holds the fractures closed and the total porosity can be less than 1 percent. The base of the ground water system is indistinct because the fractures decrease in size and number with increasing depth.[2-34]

Under natural conditions, ground water in the bedrock fractures and intergranular pore spaces of the regolith is derived from infiltration of precipitation. Water enters the ground-water system in the recharge areas, which generally include all of the land surface above the lower parts of stream valleys.[2-34]

Following infiltration, the water slowly moves downward through the unsaturated zone to the water table, which is the top of the saturated zone. Water then moves laterally through the saturated zone and discharges naturally as seepage springs on steep slopes and as bank and channel seepage into streams, lakes, or swamps.[2-34]

The depth to the water table varies from place to place and from time to time depending on the topography, climate, growing season, and properties of the water bearing materials. Topography probably has the greatest influence on the depth to the water table in a specific area with the other effects superimposed to cause short-term fluctuations.[2-34]

In stream valleys and areas adjacent to ponds and lakes, the water table may be at or near the land surface. On the upland flats and broad interstream divides, the water table generally ranges from a few feet to a few tens of feet beneath the surface, but on hills and rugged ridge lines, the water table may be at considerably greater depths.[2-34]

In general, the shape of the water table is similar to the topography of the land surface, but the relief of the water table is less than that of the land surface. The water table divides tend to coincide with ridges and hilltops, which are also the surface-water drainage divides.[2-34]

Seasonal changes in water levels can be related to seasonal changes in the use of water by vegetation and the rate of soil moisture evaporation. During the growing season, vegetation intercepts and consumes large amounts of water before it reaches the water table, especially from mid-April through October. During the same period, warmer temperatures contribute to higher rates of soil moisture losses through evaporation. As a result, the water table declines gradually throughout the summer and fall months and is usually lowest in the late fall. It is at this time of year that the ground-water system has the least amount of water in storage. The long steady rains, lower temperatures, and low transpiration losses during the winter and early spring months favor the recharge of ground water. Barring unusual weather conditions, the water table will rise and fall cyclically on an annual basis and at a given time each year will be approximately at the same level.[2-34]

The most favorable path for ground-water flow in foliated metamorphic rocks, excluding fractures, is in the plane of foliation. This hydrologic characteristic is most pronounced in the regolith that is derived from foliated rocks such as schists and gneisses where the permeability in the plane of the relic foliation may be one or two orders of magnitude greater than the permeability normal or at an angle to the foliation.[2-34]

Cracks, burrows, roots, and quartz seams allow ground water to percolate more rapidly. Near the land surface, water movement may also be affected by buried pipes. Drainage patterns shown by streams generally reflect the fracture pattern of the bedrock. Percolation of water into the bedrock should be enhanced by fractures. Thus, areas of high fracture frequency might be expected to have lower drainage densities because less water occurs as surface runoff.[2-35]

Reactor Site Ground Water Hydrology

Ground waters are estimated to move slowly through the soil surrounding the reactor site under the influence of gravity ranging in rate up to a few feet per day as previously reported and from data given in Table 2-33 for a well located approximately 4 km south of the reactor site.[2-36] After percolating downward through the pore space in the soil, the ground water is shunted almost laterally by the bedrock and discharges into Rocky Branch Creek. Wells in the surrounding area provide generally small, sometimes moderate, supplies of water of only a few gallons per minute.[2-35] According to the Wake County Geographic Information Services, there are no major wells that are still in service located between the reactor site and Rocky Branch Creek.[2-10]

A monitoring well has been installed near the corner of Stinson Drive and Broughton Drive southwest of the reactor building on the NCSU campus. This well is located in the general direction of ground water movement from Hillsborough St to Rocky Branch creek. The well is approximately 40 feet deep (elevation [redacted]) from the street elevation ([redacted]). The reactor building lower elevation is at [redacted]. Details on the ground water monitoring well are given in Figure 2-42 through Figure 2-44.[2-11]

Given the topography near the reactor site and flood prone areas being located approximately 100 m or more away from the reactor site boundary, and the top of the reactor pool being approximately [redacted] above street level, damage to the reactor facility from flooding is considered unrealistic and therefore not credible. As a result, no analysis is made.

Airborne releases that may potentially contaminate surface waters and ground water from deposition on soil are expected to be non-detectable. Given the magnitude of a release and the amount of atmospheric dilution and diluting water, public dose is estimated as being negligible, or well below 1 mrem. Refer to Section 13 for analysis.

Water from the primary coolant system that potentially enters the surface waters or ground water would eventually appear in the Rocky Branch Creek and later in the Neuse River. This event is considered to be unlikely, abnormal, and infrequent. However, sampling and analyses may be performed to assess the release and public dose assuming it were to occur. Given the magnitude of the potential release compared to that of diluting waters, the anticipated dose would be well below public limits. Should a leak pathway be identified, repairs would be considered and implemented if practical. Refer to Section 13 for analysis.

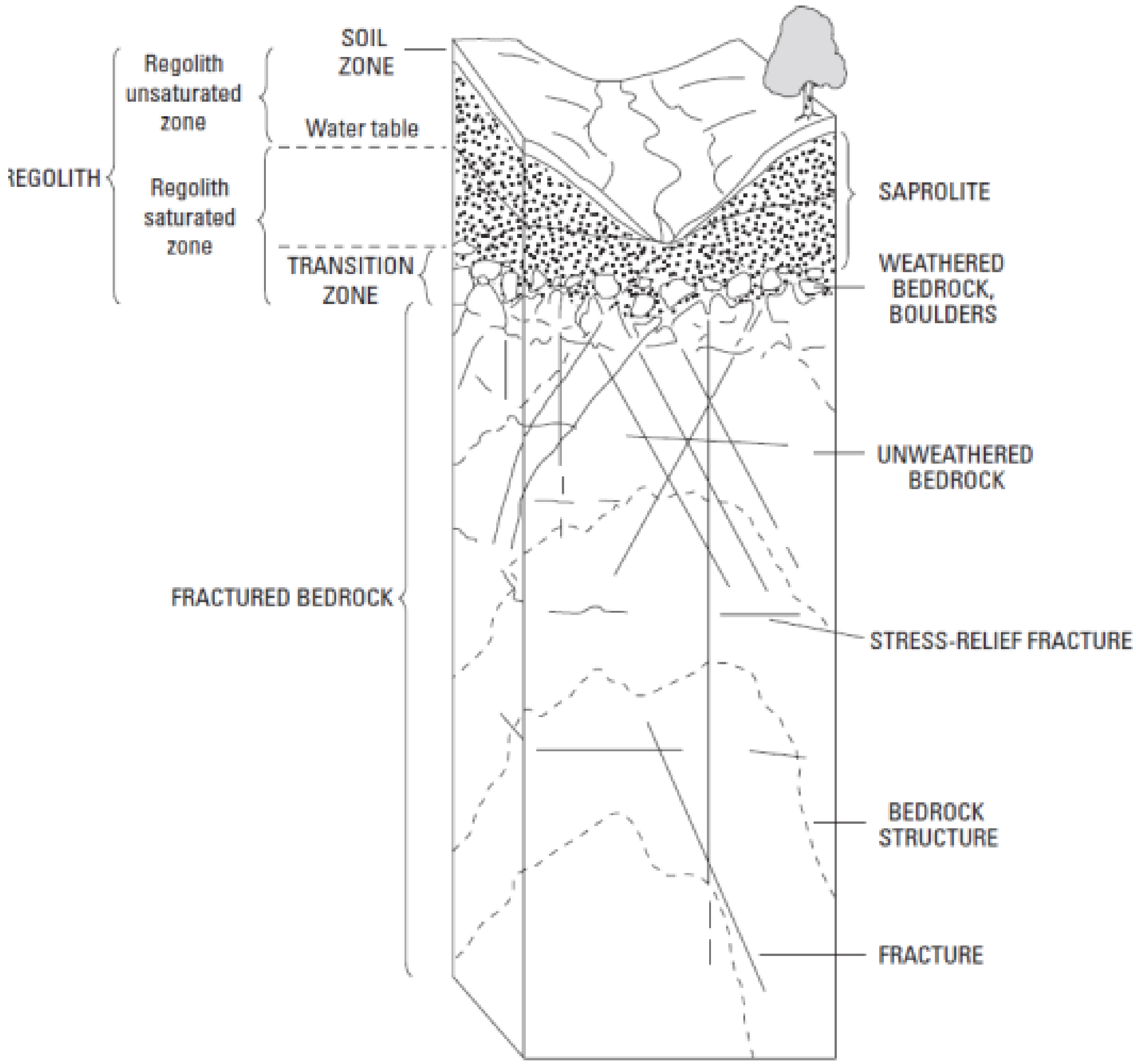


Figure 2-38 – Components of the Piedmont and Mountains Ground Water System in North Carolina

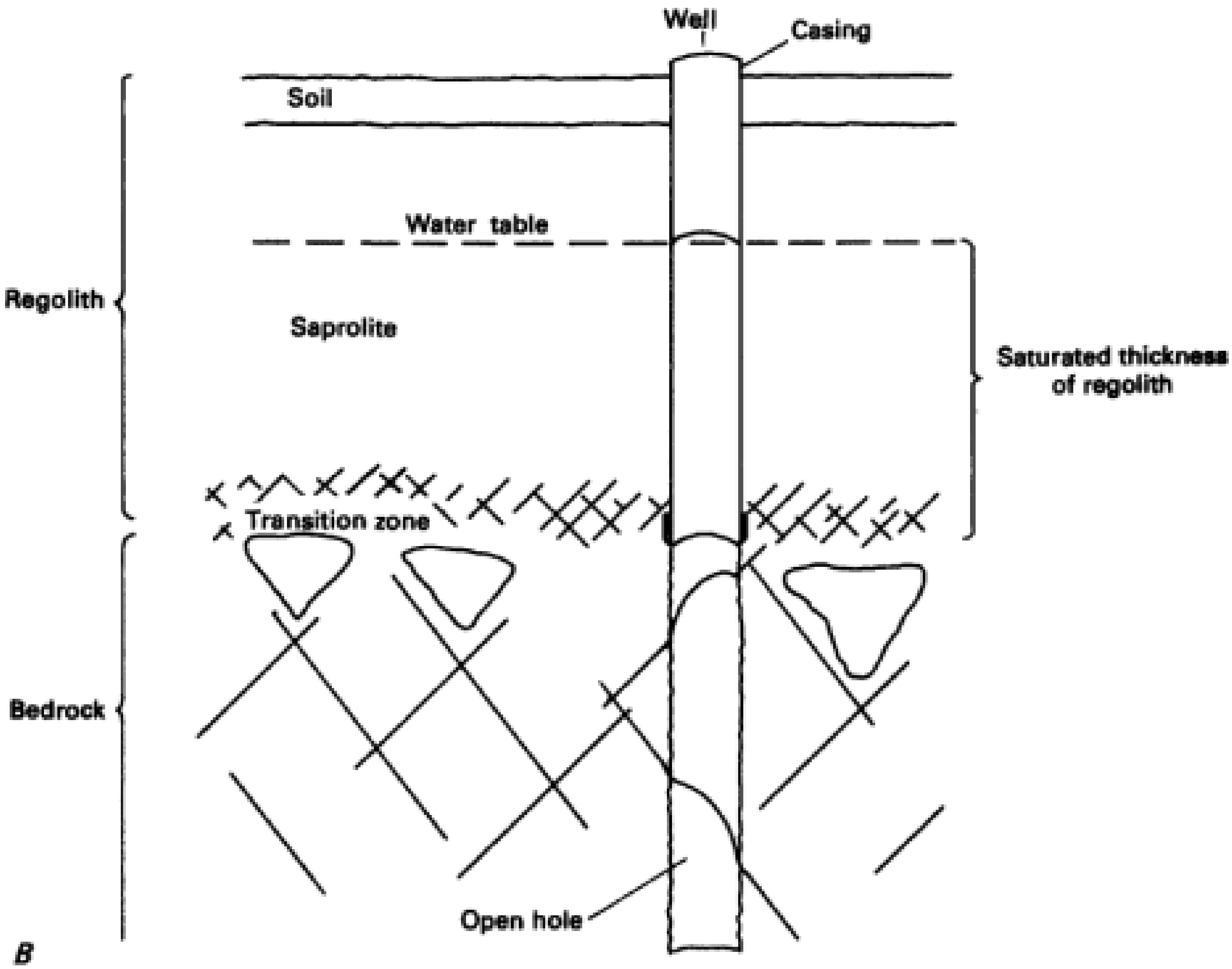


Figure 2-39 – Ground Water System showing Typical Construction of a Drilled Open-holed Well

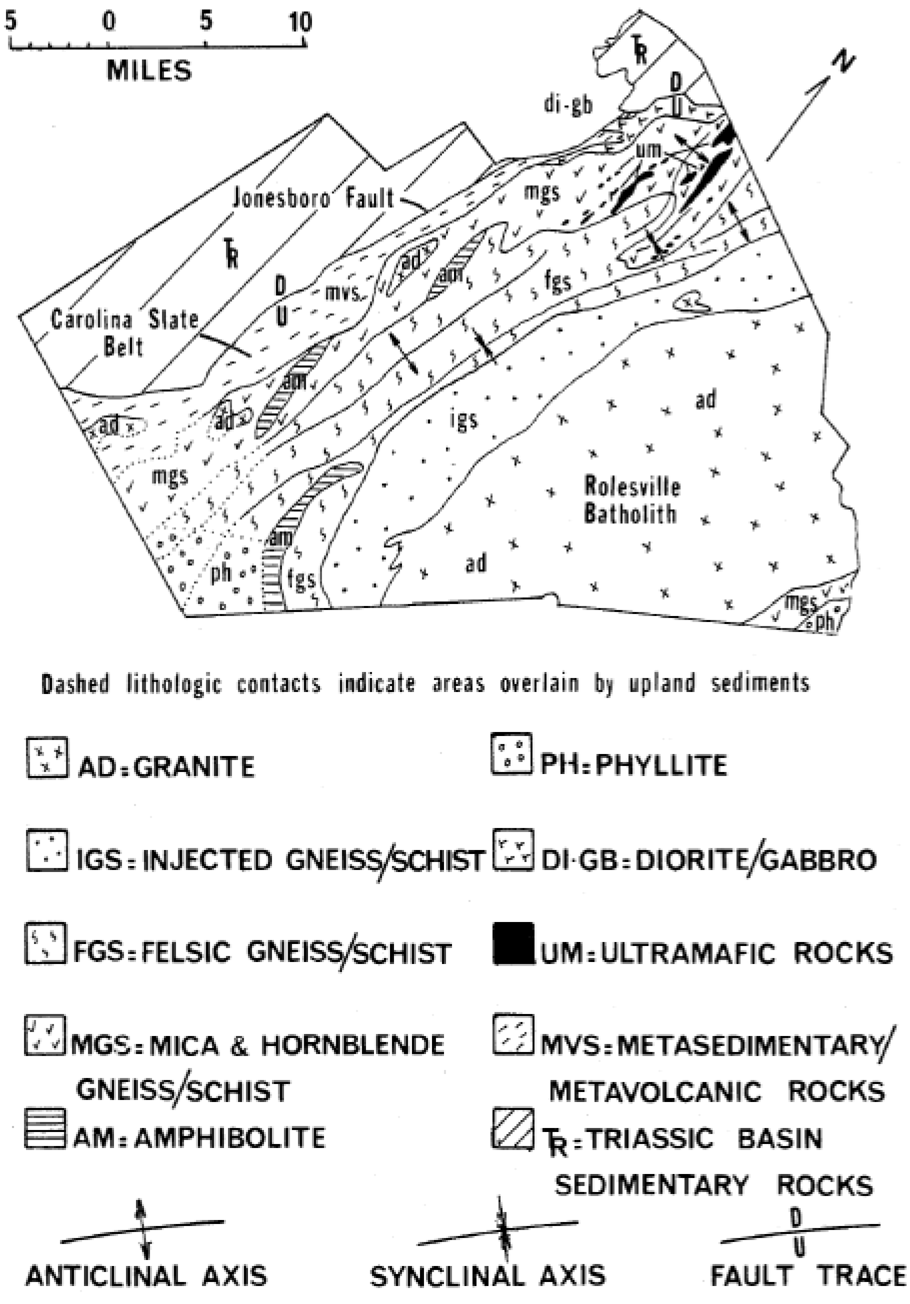


Figure 2-40 – Crystalline Rock Types – Wake County, NC

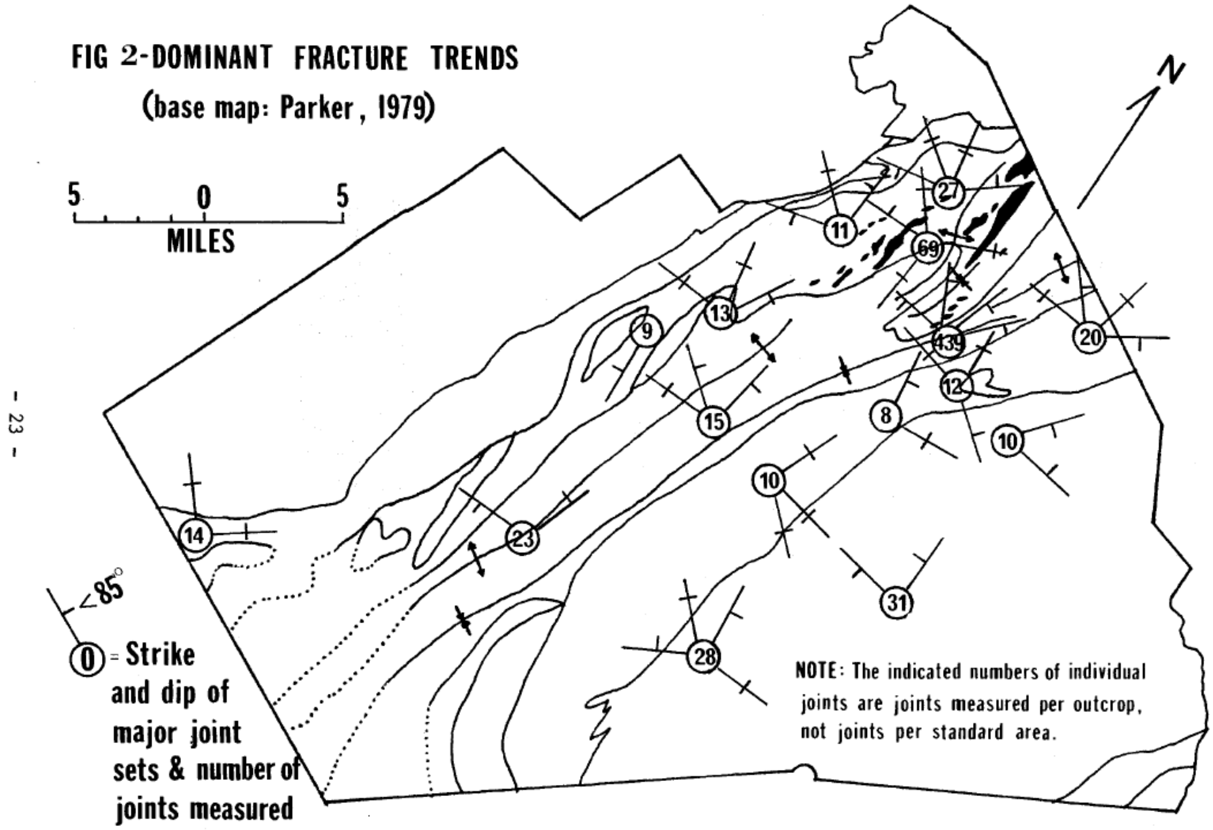


Figure 2-41 – Dominant Fracture Trends – Wake County, NC

Table 2-33 – Analytical Results of Slug Tests in Wells at the Raleigh Hydrogeologic Research Station

Well Number	Screened/open Interval (feet below land surface)	Hydraulic Conductivity (feet per day)
Regolith Wells		
WC-1S	13-28	0.8
WC-2S	13.5-28.5	7
WC-3S	13.6-28.5	6
Transition Zone Wells		
WC-1I	24-39	2
WC-2I	27-42	5
WC-3I	34-49	5
Bedrock Wells		
WC-1CH	21-90	0.6
WC-2D	59-440	10
WC-2CH	70-85	3
WC-3D	40-300	4
WC-3CH	40-125	0.4

[redacted]

Figure 2-42 – Well Sites Location at the PULSTAR Reactor Facility

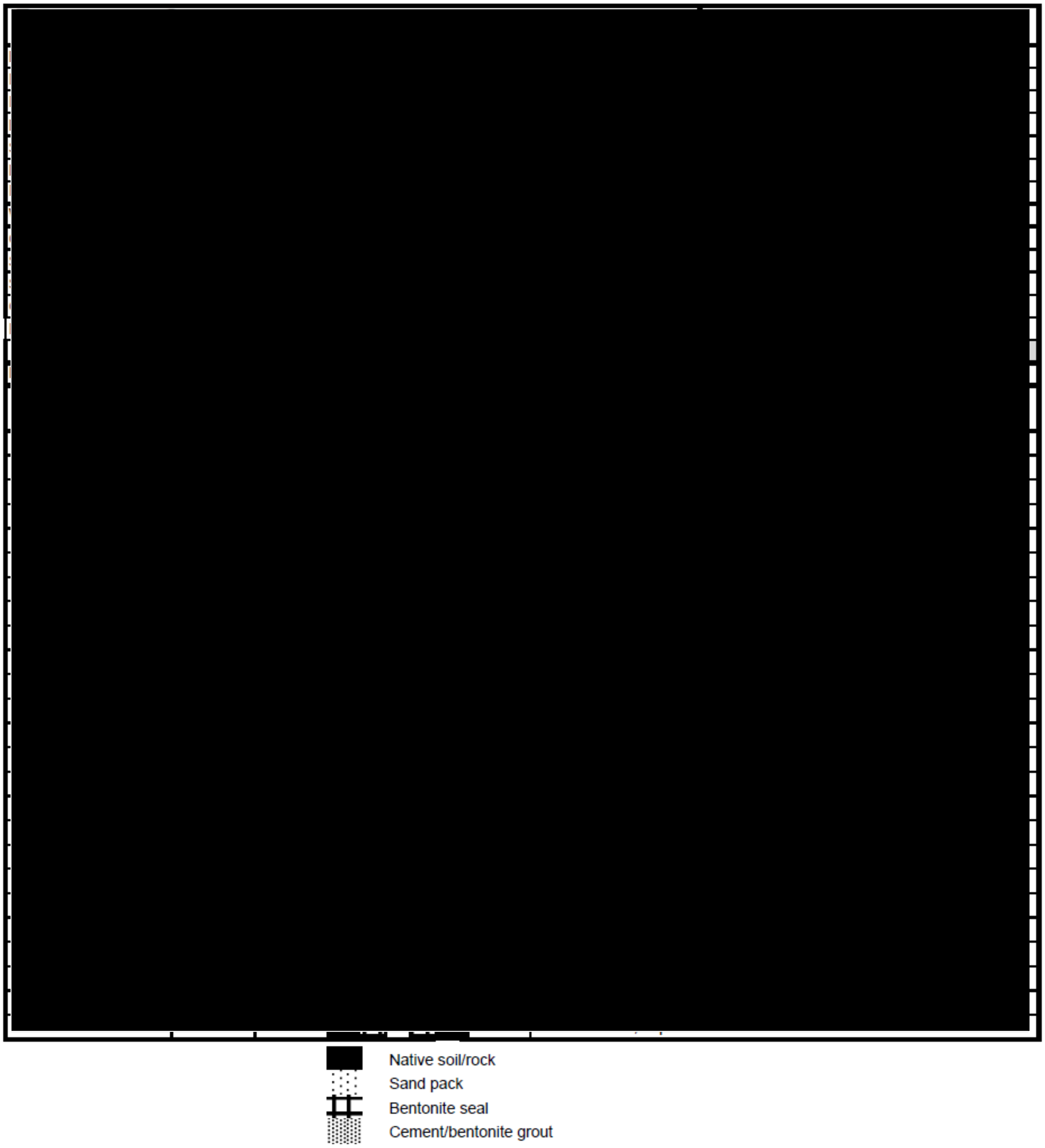


Figure 2-43 – Well Site Data at the PULSTAR Reactor Facility – Sheet 1

[redacted]

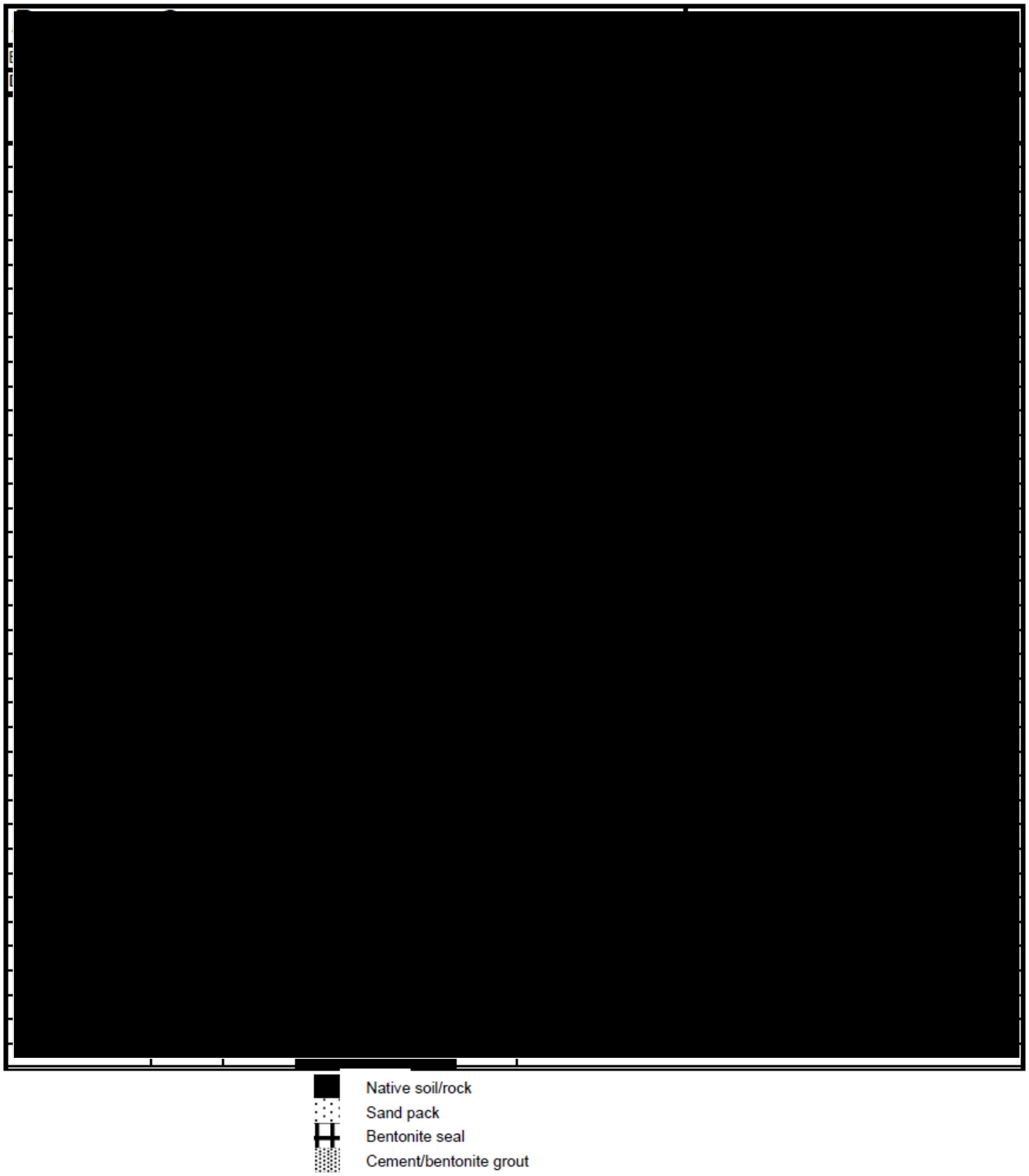


Figure 2-44 – Well Site Data at the PULSTAR Reactor Facility – Sheet 2

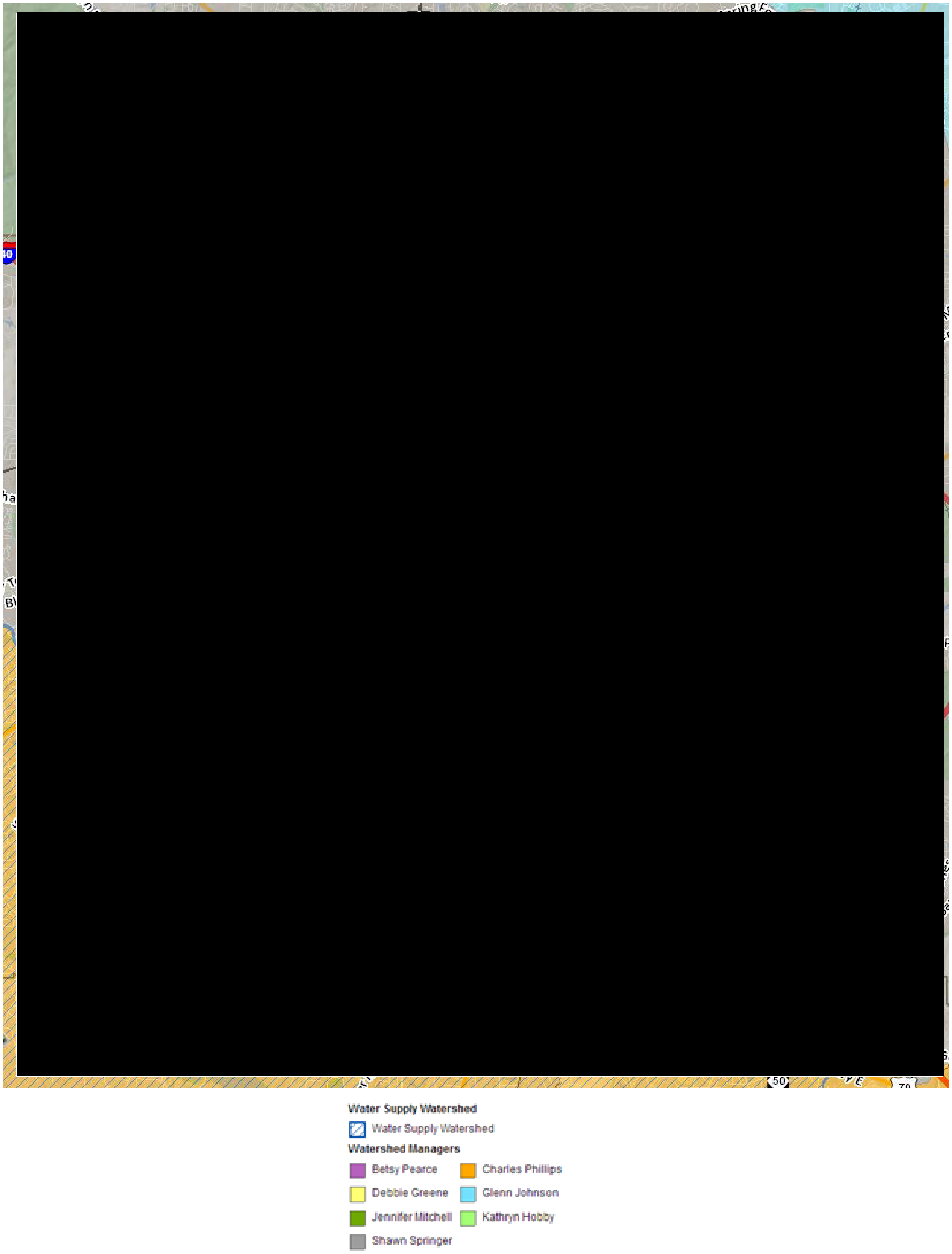

[redacted]

Figure 2-45 – Water Supply and Watersheds within 8 km of the Reactor Site – Raleigh, NC

2.5. Geology, Seismology, and Geotechnical Engineering

Geologic and seismic information regarding reactor site and area are discussed below. Construction of the reactor building and Burlington Engineering Laboratory followed applicable building codes. Overall, the reactor site and area are stable and at a low risk of seismic damage.

2.5.1. Regional Geology

North Carolina is divided into three major physiographic provinces; the Coastal Plain, Piedmont, and the Blue Ridge. Each province is characterized by particular types of landforms. The reactor is located in the Piedmont province of North Carolina.[2-2]

Raleigh and the reactor site are located in the Raleigh Belt of the Piedmont province. The Raleigh Belt contains granite, gneiss, and schist. Within the Raleigh Belt are the Crabtree Terrane and Raleigh Terrane. The Falls leukogneiss is part of the Raleigh Terrane.[2-3]

Adjacent regions include the Carolina Slate Belt, Triassic Basins, Eastern Slate Belt, and Coastal Plain. The Carolina Slate Belt consists of heated and deformed volcanic and sedimentary rocks and was the site of a series of oceanic volcanic islands about 550-600 million years ago. The Triassic Basins are filled with sedimentary rocks that formed 200-190 million years ago. Streams carried mud, silt, sand, and gravel from adjacent highlands into rift valleys. The Eastern Slate Belt contains slightly metamorphosed volcanic and sedimentary rocks similar to those of the Carolina Slate Belt. The rocks are poorly exposed and partially covered by Coastal Plain sediments. The metamorphic rocks are 500-600 million years old and are intruded by younger 300 million year old granitic bodies. The Coastal Plain is a wedge of mostly marine sedimentary rocks that gradually thickens to the east. The most common sediment types in the Coastal Plain are sand and clay with a significant amount of limestone in the southern part of the Coastal Plain.[2-3]

There are several ancient and inactive faults in Wake County and the Raleigh, NC area. These include the Nutbush Creek fault, Leesville fault, Fall Lake fault, and Jonesboro fault. The Nutbush Creek fault is located on the NCSU campus near the reactor site. The Nutbush Creek fault, which separates the Raleigh terrane from the Crabtree terrane, goes through the NCSU campus and passes near the reactor site (Broughton Drive and Morrill Drive). This is a right-lateral strike slip fault. The Nutbush Creek fault has shear zones from the late Paleozoic age. In addition, the Raleigh Anticline is also located on the NCSU campus. The Jonesboro fault is on the boundary between the Triassic Basins and Raleigh Belt. The Jonesboro fault is located near Cary, NC, which is approximately 10 miles west of the NCSU campus. The Jonesboro fault is a normal fault of Mesozoic age. These geologic features are illustrated in provided figures. Diabase dikes of Alleghanian age are also present in Wake County.[2-3]

The Nutbush Creek fault extends from southern Virginia through Wake County (NC) to Lillington, NC and has a reported total length of approximately 180 km. The fault zone in the Raleigh area is diffuse and is reported to range from 1 km to 10 km wide. Near Lillington, NC the fault disappears beneath the Coastal Plain deposits. This fault is recognized by these consistent features; (1) intensely deformed mylonites, phyllonits, and L-tectonites in a north-northeast trending belt averaging 1 km wide, (2) mesoscopic and microscopic structures indicating right-lateral displacement, and (3) different rock suites on opposite sides of the fault zone. Along the west flank of the Raleigh terrane, the Nutbush Creek fault zone juxtaposes different rock units and terranes. From the Tar River north to the Virginia border, high-grade Raleigh gneiss is juxtaposed against rocks of the Carolina Slate Belt. South of the Tar River, the Nutbush Creek fault zone places the Raleigh terrane gneiss against rocks of both the Falls Lake melange and felsic gneiss. A consistent right lateral displacement is determined from S-C mylonites, shear bands, rotation of porphyroclasts, and minor fold-vergence. Displacement was estimated in 1988 to be about 160 km. This estimate is based on the interpretation that the contact between volcanogenic rocks and continental basement rocks, exposed at the southern end of the Rolesville batholith, and truncated to the west by the Nutbush Creek fault, reappears west of the fault in the south-central Virginia Piedmont. Shearing is bracketed at about 300 Ma by whole-rock Rb-Sr dates on sheared and unsheared granite plutons. The 314 Ma Buggs Island pluton is deformed and the 285 Ma Wilton pluton is not. The Nutbush Creek fault zone is one of many right lateral Alleghanian faults that affected the Piedmont. These include the Eastern Piedmont fault system and other major faults as far west as the Brevard zone.[2-3]

The geological evolution of the Wake County area mainly involves three distinct periods of geological time, during which major tectonic events occurred.

Neoproterozoic (620 - 540 million years ago)

This period of time includes the very last of the Precambrian and the very beginning of the Paleozoic. Rocks of this age range are the oldest in the region. These rocks are related in some way to volcanic arc activity of the Carolina slate belt or other volcanic arcs and originally formed far away from ancient North America. Subduction of parts of the oceanic crust was responsible for the formation of magmas that resulted in the volcanism. In between volcanic eruptions, sediment was deposited on the flanks of the volcanoes, on land and under water between volcanic islands. Some of the magmas never reached the surface, but crystallized beneath the volcanoes as plutons. Later, during the Alleghanian, all of these rocks were metamorphosed and deformed by folding or faulting. As a result, these rocks are metamorphic rocks with variable metamorphic intensity. For example, the original nature of some of the rocks is still apparent; these may be referred to as metavolcanic, metaplutonic, or metasedimentary.[2-37]

Late Paleozoic (a.k.a. Alleghanian; 320-280 million years ago)

This is the time of the Alleghanian orogeny, a major mountain-building episode. This is when the ancient North American continent collided with the ancient counterparts of Europe and Africa. This is the huge collision that formed the Appalachian mountain belt (compare to India crashing into Asia today). There was a tremendous amount of compression that resulted in folding and faulting (including the Blue Ridge thrust). The huge amount of crustal thickening that occurred caused rocks to be buried deep in the earth, resulting in intense regional metamorphism. Also, heat generated during the collision caused melting, and granite magmas formed and moved upward, finally stopping and cooling in the middle levels of the crust, to form plutons such as the Rolesville batholith. About the same time as the granite bodies were forming, the big collision of continents slowed and instead evolved to a sideways motion (compression changing to shear). This sideways motion resulted in a number of right-lateral strike-slip faults, such as the Nutbush Creek fault.[2-37]

Early Mesozoic (210-190 million years ago)

This segment of time records the breakup of the supercontinent Pangaea that occurred during the Triassic period and the early Jurassic. As the continent began to stretch, normal faults developed (tensional forces) throughout the region. An example of such a fault is the Jonesboro fault. Some of these continued to move so that long rift valleys formed. These rift valleys caused changes in the pattern of rivers and streams at the time, so that streams flowed down out of the mountainous areas on the upthrown side of the fault (foot wall) and into the valleys (hanging wall). They deposited sediment in the valleys. Later this sediment was lithified, and is now represented by clastic sedimentary rocks. One of these rift valleys is continuing to open today; it is the Atlantic Ocean. When the Atlantic began to open, mantle rock (peridotite) began moving upward and underwent partial melting. This resulted in the formation of basaltic magma, and the magma intruded into many of the faults and fractures. They are now diabase dikes.

Some of the magma intruded along bedding planes of the sedimentary rocks making diabase sills; and farther north, some of it was erupted as lava flows. This process of manufacturing basalt magma continues today, along the Mid-Atlantic Ridge. The entire Atlantic Ocean crust was formed in this way, beginning in the early Mesozoic.[2-37]

Geologic maps of the region are provided in Figure 2-46 through Figure 2-50.[2-38,2-39,2-40]

Earthquakes

Earthquakes everywhere occur on faults within bedrock, usually miles deep. Most bedrock beneath the inland Carolinas was assembled as continents collided to form a supercontinent about 500-300 million years ago, raising the Appalachian Mountains. Most of the rest of the bedrock formed when the supercontinent rifted apart about 200 million years ago to form what are now the northeastern USA, the Atlantic Ocean, and Europe.[2-3]

The inland Carolinas are located on the North American tectonic plate and are far from the nearest plate boundaries located in the center of the Atlantic Ocean, Caribbean Sea, west coast of the USA and Canada, and eastern Siberia. The region is laced with known faults and numerous smaller or deeply buried faults that remain undetected. Even the known faults are poorly located at earthquake depths. Few, if any, earthquakes in the inland Carolinas can be linked to named faults. Faults identified to date in North Carolina are ancient and inactive. As described previously, there are several ancient and inactive faults in Wake County and the Raleigh, NC area.[2-3]

Earthquakes in the central and eastern USA, although less frequent than in the western USA, are typically felt over a much broader region. East of the Rockies, an earthquake can be felt over an area as much as ten times larger than a similar magnitude earthquake on the west coast. A magnitude 4.0 eastern USA earthquake typically can be felt at many places as far as 100 km (60 mi) from where it occurred, and it infrequently causes damage near its source. A magnitude 5.5 eastern USA earthquake usually can be felt as far as 500 km (300 mi) from where it occurred, and sometimes causes damage as far away as 40 km (25 mi).[2-41]

Earthquake data before 1886 are sparse. Seismic instruments were installed in the region in the late 1920's. Prior to that time earthquake data are based on historical records. The distribution of seismograph stations did not allow for location of earthquakes with magnitudes <4 until 1962-1963. Micro-earthquake sensing networks began operating in the region in the mid-1970s.[2-41]

Since at least 1735, people living inland in North and South Carolina, and in adjacent parts of Georgia and Tennessee, have felt small earthquakes and suffered damage from infrequent larger ones. The largest earthquake in the area (magnitude 5.1) occurred in 1916. Moderately damaging earthquakes strike the inland Carolinas every few decades, and smaller earthquakes are felt about once each year or two.[2-41]

The USGS reports 3350 earthquakes from 1700 to 2006 in the Central and Eastern North America (CENA) with a magnitude greater than or equal to 3.0. These include earthquakes with epicenters outside the USA that have affected areas within the USA.[2-41]

Landslides

Landslides are possible in areas with steep slopes. In the Piedmont region, landslides may be caused by large rainstorms, hurricanes, freeze-thaw processes and human activities. Landslides made of different types of material travel at different speeds. Some landslides only consist of soil, called an earthslide. Some are a mixture of soil, rock trees and mud, called a debris flow. Other landslides contain only rock, called a rockfall or rockslide. If the land is slowly pulling apart from the hillside, tension cracks may appear. With time, the ground on one side of the tension crack may slide downhill forming a scarp. If the ground moves far enough, it will leave a mark called a scar. A fresh scar will usually have a lighter color and no vegetation compared to the surrounding slopes. Debris flow and rockslides are faster moving and can leave a trail of destruction along its path and a pile of debris or rocks at the end of its path. The reactor site is not located on a steep slope and is well drained.[2-42]

The Piedmont region is characterized by a dissected rolling plain formed on residual soil from deeply weathered metamorphic rocks bordered on the east by a dissected terraced plain on thick deposits of sand, gravel, and clay. Most of the region is free of landslides. High and rounded hills in the interior of the Carolinas are covered with thick residual soil and colluvium overlying igneous and metamorphic rocks. The weathered metamorphic rocks, especially micaschist and micagneiss, are susceptible to earth flows, slumps, and rockslides. The coastal plains, which are composed of sand, clay, and limestone, are generally free of landslides, although a few slumps occur along river valleys.[2-42]

For Wake County NC, the USGS Landslide Hazard Map rates the landslide incidence as low affecting less than 1.5% of the area. The weathered bedrock at the reactor site does contain micaschist and micagneiss, but there are no steep slopes. Excavation and other human activities are reviewed by NCSU and others to prevent damage to surrounding buildings. No landslides or evidence of landslides have been reported near the reactor site.[2-41]

Sinkholes

Sinkholes are commonly associated with areas with limestone bedrock. Rain water may percolate down through the soil to openings in the limestone bedrock, gradually dissolving the rock matrix. Void spaces eventually form in the subsurface form, ranging from microscopic to cavern size.[2-42]

In most areas of the southeastern United States, the limestone bedrock is not directly exposed at the surface, but is covered by a variable thickness of sand, silt and clay. This overburden may bridge subsurface cavities for long periods of time. Eventually a catastrophic collapse of the overburden into the subsurface cavity may occur, and a sinkhole is formed. This type of sinkhole is known as a cover collapse sinkhole.[2-42]

A cover collapse sinkhole is just one end of the sinkhole spectrum. At the opposite end of the spectrum is the cover subsidence sinkhole, formed where overburden is relatively thin (a few feet to tens of feet). In this setting, as subsurface solution occurs, the land surface gradually subsides into the void space below, since it lacks the cohesiveness to form a significant "bridge" across the void. Cover-subsidence sinkholes are often mistaken for other land subsidence features, since they do not form in as spectacular a manner as the cover-collapse sinkhole. One common indicator of this type of sinkhole is the formation of cracks in nearby buildings or in roads.[2-42]

Under natural conditions, sinkholes usually form rather slowly, over the course of many years. However, some human activities can trigger abrupt sinkhole formation, or accelerate processes that have been going on for a long time. Activities such as dredging, diversion of surface drainage systems, or pumping of ground water can accelerate the natural growth of sinkholes.[2-42]

In North Carolina, sinkholes are common features of the outer coastal plain. Most NC sinkholes become flooded and appear as small to medium sized circular lakes. They can be distinguished from non-sinkhole lakes by the absence of any outflow drainage and lack of relationship to surface drainage systems.[2-41]

Areas in NC affected by sinkholes are greater than 50 miles from the reactor site. The bedrock near the reactor site is metamorphic (Raleigh Belt) and does not contain limestone. No sinkholes or evidence of sinkholes have been reported near the reactor site.[2-41]

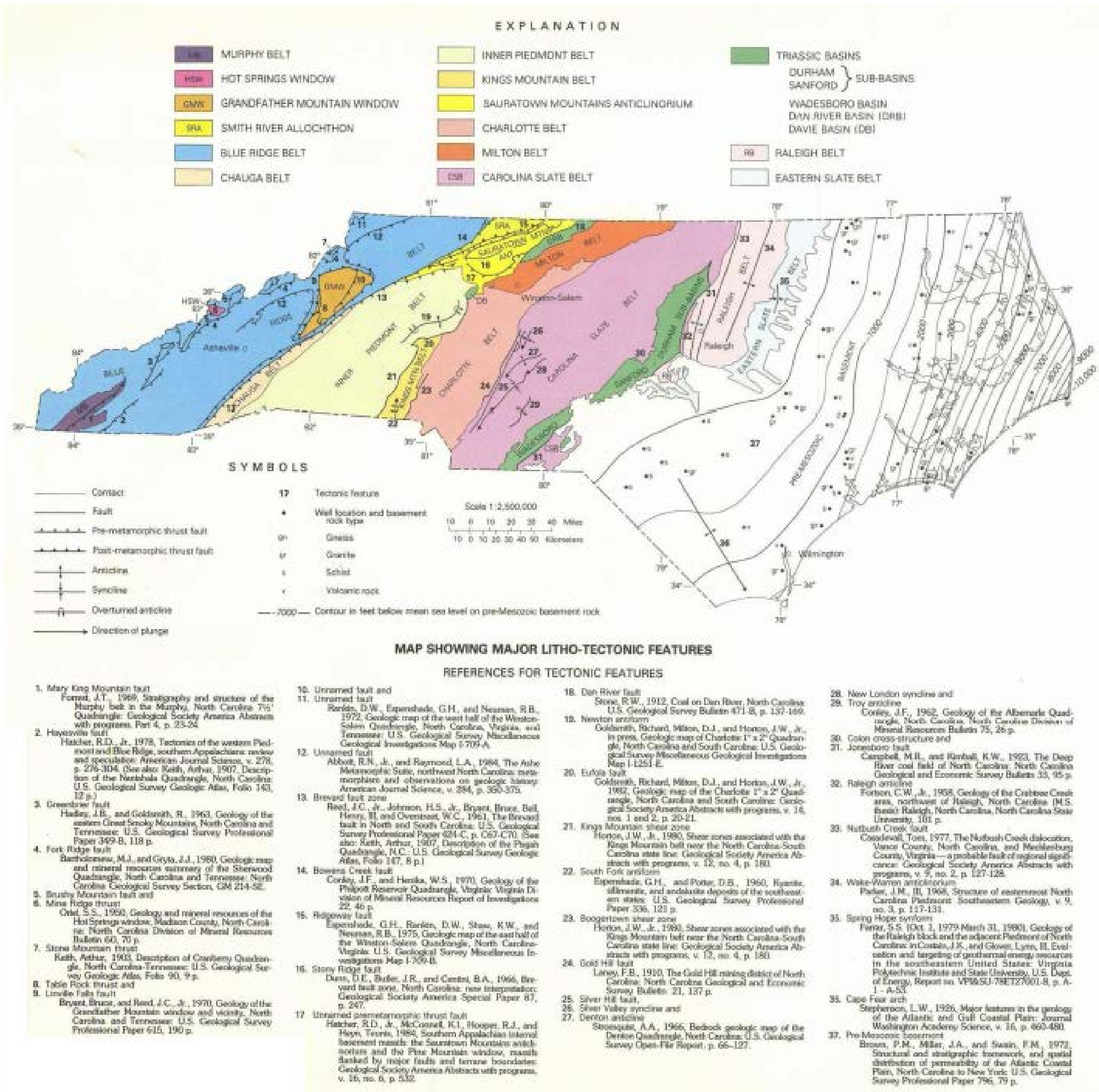


Figure 2-46 – Map of Major Litho-Tectonic Features

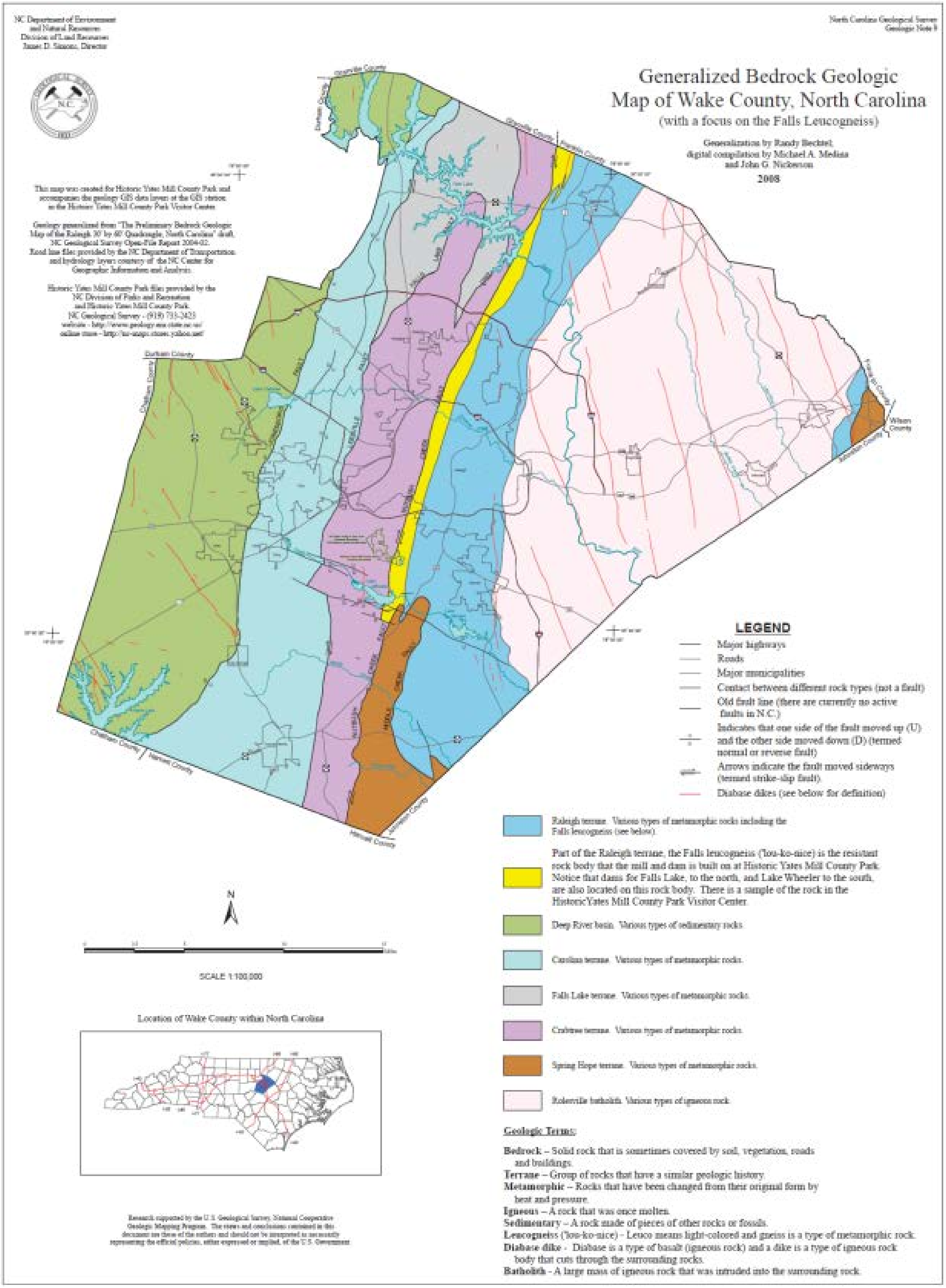


Figure 2-47 – Generalized Bedrock Geologic Map of Wake County, North Carolina

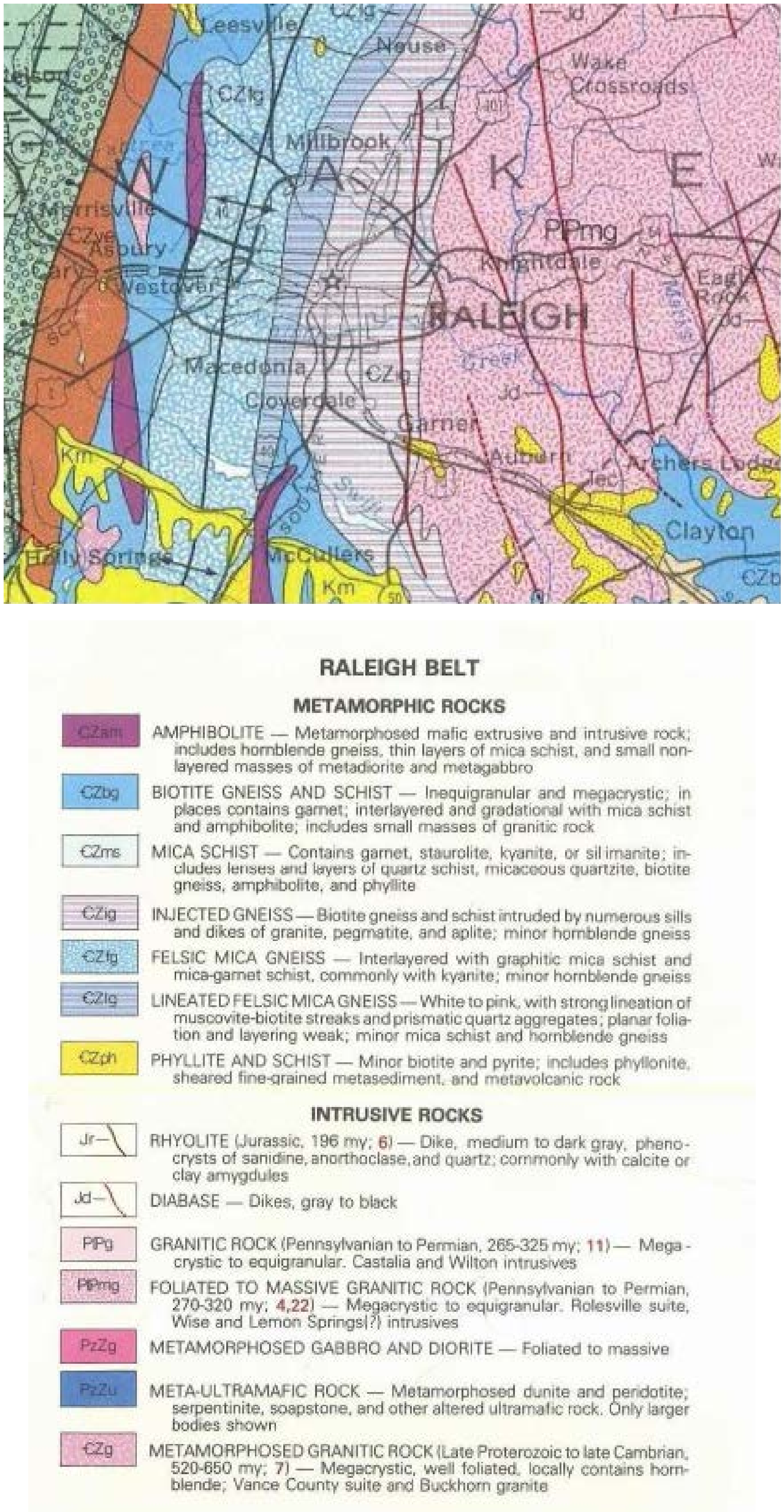


Figure 2-48 – Generalized Bedrock Geology Map of Wake County, NC

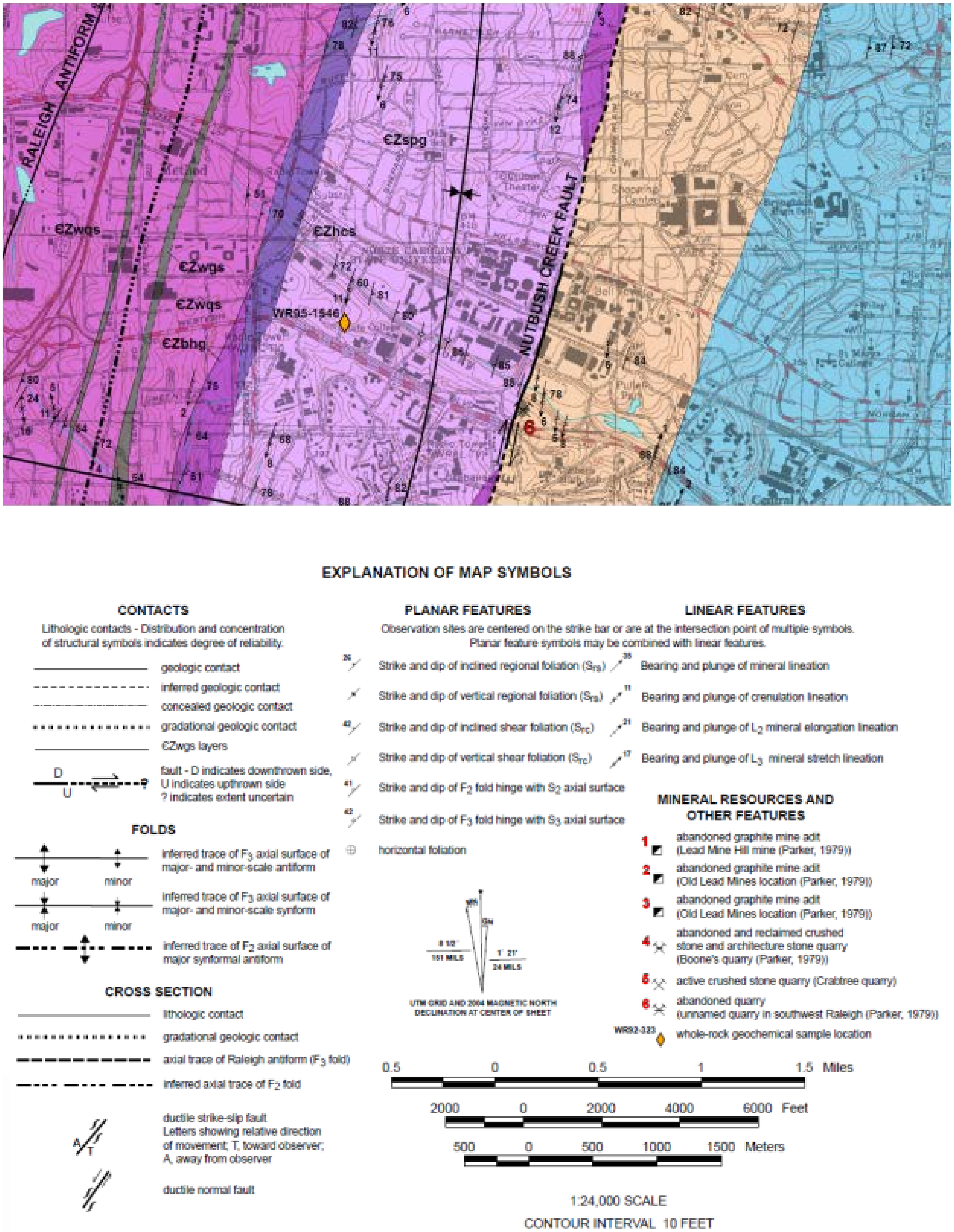
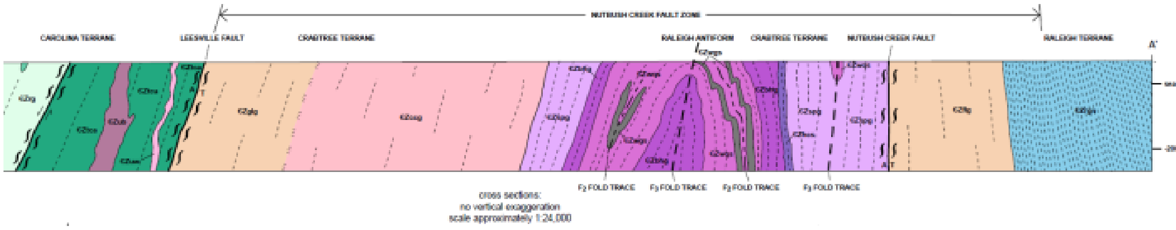


Figure 2-49 – Geologic Map of the Raleigh West 7.5 Minute Quadrangle – Wake County, NC

Cross Section of Nutbush Creek Fault Zone



Metasedimentary and Metavolcanic (?) Units

Crabtree Metamorphic Suite

ЄZrcs	ЄZrcs – Richland Creek schist: Silver-gray, white-gray, or tan-white, fine- to medium-grained, compositionally layered, and well-foliated. Mixed metasedimentary pelitic and metavolcanic (?) felsic lithodemic layers of schist, phyllonite, and fine-grained gneiss. White mica, biotite, quartz, and feldspar define a lepidioblastic and granoblastic matrix for local garnet, staurolite, and tourmaline porphyroblasts, generally less than 1 cm in size. Contains white mica garnet graphite schist layers (CZwgs) that are less than 10 meters thick, and generally less than 1 meter thick. Graphite schist layers have variable lengths from 1 meter up to several km and their distribution is inferred to represent laterally discontinuous metasedimentary horizons (see Stop 4 in Blake et al., 2001).
ЄZspg	ЄZspg – Southwest Prong gneiss: Leucocratic (CI<5-10), orange-gray and tan-white, dominantly very fine- to fine- and locally medium-grained, poorly to well-foliated depending on mica content. Lithodemic compositional layers vary in biotite content chiefly in a granoblastic microcline, plagioclase, quartz, and white mica matrix. Locally, horizons are magnetite-bearing. Mylonitic and ultramylonitic fabric element overprint obscures primary protolith textures and contact relationships, which may be volcanic or plutonic based upon its felsic bulk-rock composition (see Stop 3 in Blake et al., 2001).
ЄZwqs	ЄZwqs – White mica quartzitic schist: Gold-gray and silver-gray, very fine- to fine-grained, compositionally layered, and well-foliated. Mixed metasedimentary pelitic and quartzitic, and metavolcanic (?) felsic layers of schist, phyllonite, and fine-grained gneiss. White mica, biotite, minor chlorite, quartz, and feldspar define a lepidioblastic and granoblastic matrix for local garnet, staurolite, and kyanite porphyroblasts, generally less than 1 cm in length. Contains three prominent white mica garnet graphite schist layers (ЄZwgs) that are greater than 10 meters thick, but continuously extend along the length of the unit and are locally known as the Raleigh Graphite.
ЄZbhg	ЄZbhg – Biotite ± hornblende gneiss: Mesocratic (CI=30-50), white-gray to black-gray, very fine-grained to fine-grained, compositionally layered and well foliated. Variations in biotite, hornblende, plagioclase, and lesser white mica and microcline content produce the granoblastic or lepidioblastic lithodemic layering that is variable in thickness. Mylonitic and ultramylonitic fabric elements overprint primary protolith textures and contact relationships, which may have been volcanic or plutonic based upon its felsic bulk rock composition.
ЄZhcs	ЄZhcs – Horse Creek schist: Silver-gray to tan-gray-white, medium- to coarse-grained, and well foliated, lineated, and compositionally layered pelitic metasedimentary rock. White mica, biotite, quartz, and feldspar define a lepidioblastic and granoblastic matrix for conspicuous porphyroblasts of garnet, kyanite, and minor staurolite. Variations in mica content versus felsic minerals define the schistose foliation that may be phyllonitic.
ЄZwgs	ЄZwgs – White mica graphite schist: Silver-gray to black-gray, fine- to medium-grained, well foliated, and locally crenulated. White mica, graphite, biotite, and quartz form a schistose lepidioblastic matrix for conspicuous garnet porphyroblasts up to 5 mm in diameter that are commonly replaced by iron- and manganese-oxide minerals. Locally contains mm-scale kyanite and staurolite porphyroblasts. Three prominent layers in ЄZwqs are known as the Raleigh Graphite. (see Stop 3 in Stoddard and Blake, 1994, and Stop 4 in Blake et al., 2001). Three historic adits shown in ЄZwgs.

Raleigh Terrane

Metasedimentary and Metavolcanic (?) Units

ЄZflg	ЄZflg – Falls Leucogneiss: Leucocratic (CI<5), orange-tan, white, pink-white or pink-gray, fine- to medium-grained, weakly to moderately foliated and penetratively lineated. Microcline, plagioclase, and quartz defines a granoblastic matrix for aggregates of biotite and magnetite that form a mineral lineation in the mylonitic to ultramylonitic L>S tectonite fabric. Commonly forms rapids, massive pavement, waterfall outcrops, and dam sites along prominent drainages including Crabtree and Walnut Creeks. Boulders along Centennial Boulevard indicate that the leucogneiss crosscuts the Raleigh Gneiss as an A-type granitoid intrusion (see Stop 2 in Blake et al., 2001; Farrar, 1985a,b; Farrar and Owens, 2001).
ЄZrgn	ЄZrgn – Raleigh Gneiss: Leucocratic to melanocratic (variable CI), white-tan-pink to gray-black to black, fine- to coarse-grained, compositionally layered, foliated, and locally lineated. Variations in biotite, hornblende, and granoblastic plagioclase, microcline, and quartz content and crystal size define the complex cm- to meter-scale, discontinuous compositional layering that displays a migmatitic aspect and protomylonitic or mylonitic fabric elements. Clinopyroxene, scapolite, calcite, apatite, titanite, garnet, white mica, and epidote occur locally. Gneissic layers display mutually concordant or discordant relationships and indicate a complex, multi-intrusive history involving magmas ranging from gabbroic to granitic. Moderately to weakly foliated biotite, microcline, plagioclase, and quartz granitic gneiss, pegmatitic granite, and fine-grained aplite commonly cross-cut mesocratic and melanocratic biotite gneiss, biotite hornblende gneiss, and amphibolite (see Stop 1 in Blake et al., 2001).

Figure 2-50 – Geologic Map of the Raleigh West 7.5 Minute Quadrangle – Wake County, NC

2.5.2. Site Geology

In the immediate NCSU campus area, the soil is mostly moderately impervious Cecil clay loam. The mineralogy of the clay fraction is approximately 50 percent kaolinite, with 40 percent subsidiary vermiculite and 10 percent free iron oxides. The cation exchange capacity of the surface soil is about 5 milliequivalents per 100 grams of soil, and increase to only 8 to 10 milliequivalents per 100 grams of subsoil. Thus, the soil would generally have a low capacity for decontaminating water by cation exchange. The coarse fraction which makes up about 25 percent to 30 percent of the soil is almost entirely quartz. The soil varies from permeable to virtually impermeable, i.e., from loose sand to tight alloy. The average depth of the soil on the NCSU Campus is about 20 feet with a range of from about

5 to 50 feet to bedrock which is predominately micagneiss with some micaschist.

Eight deep and three shallow test borings have been made at the reactor site. In general, the data indicates a dark brown stratum of sandy silt beneath the topsoil overlaying variable color strata of coarse to fine sand with a trace of silt. Sandy strata of micaceous disintegrated rock weathered in place were encountered at the bottom of all the test borings before reaching the bedrock. Typical test borings data are shown in Table 2-33.

The USGS Natural Resources Conservation Service soil survey for Wake County, NC indicates various types of Appling loam, Cecil loam, and Creedmoor loam soils make up about half of the soil and that several other soil types are less than 4% each.[2-43]

A groundwater monitoring well was installed to a depth of approximately 40 feet near the corner of Stinson Drive and Broughton Drive just off the site boundary in 2013. Data from MW1 is provided in Figure 2-43 and Figure 2-44.[2-11]

2.5.3. Seismicity

North Carolina is considered to be a non-seismic state. The United States Geological Survey has listed the following three significant earthquakes affecting North Carolina:[2-41]

1. February 21, 1916 – felt over 200,000 square miles
2. October 20, 1924 – felt over 56,000 square miles
3. November 20, 1928 – felt over 40,000 square miles

These three earthquakes had a maximum intensity of at least VI by the modified Mercalli scale (MMI).

Fourteen earthquakes have been felt in Raleigh, NC between 1811 and 2016. These include those in neighboring states (VA, SC, TN) and the New Madrid, MO earthquake of 1811. The most recent occurred on August 23, 2011. These earthquakes ranged from a MMI of II to VIII. Ten of those earthquakes had a MMI ≤ IV and two earthquakes had a MMI of V to VII. Only the Charleston SC earthquake of 1886 and the Mineral VA earthquake of 2011 had a MMI ≥ VIII. Earthquake intensities with a MMI greater than VIII have not been recorded in Raleigh, NC.

USGS regional seismicity map from 1973 to 2012 is provided in Figure 2-51. It is to be noted that no epicenters have been recorded in the immediate vicinity of the North Carolina State University reactor site. No additional earthquakes of intensity comparable to the three above have been recorded to date. Areas of recent seismic activity have occurred in central Virginia, eastern Tennessee, and South Carolina (especially near Charleston SC).

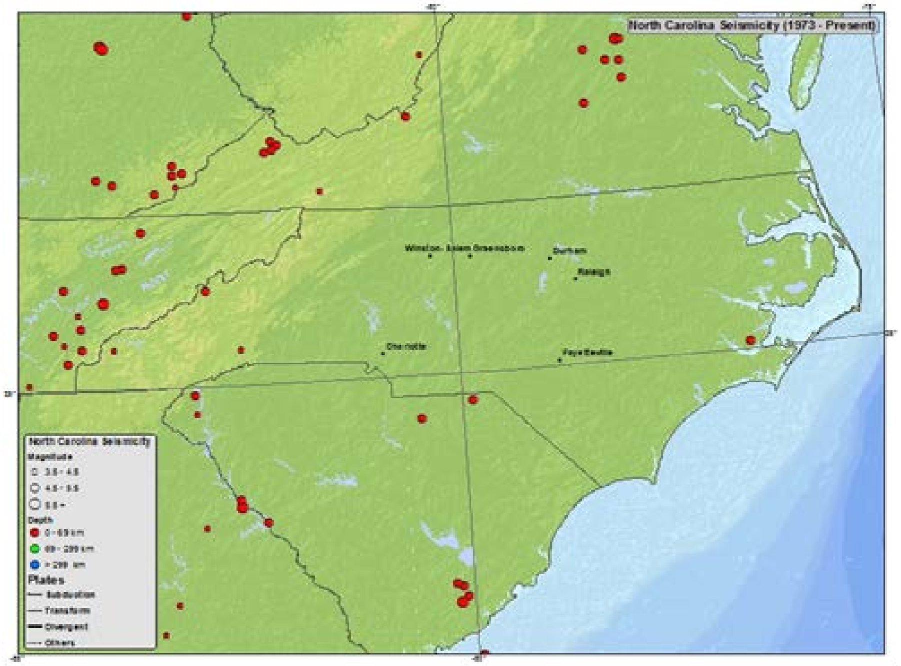


Figure 2-51 – Regional Seismicity Map – 1973 to 2012

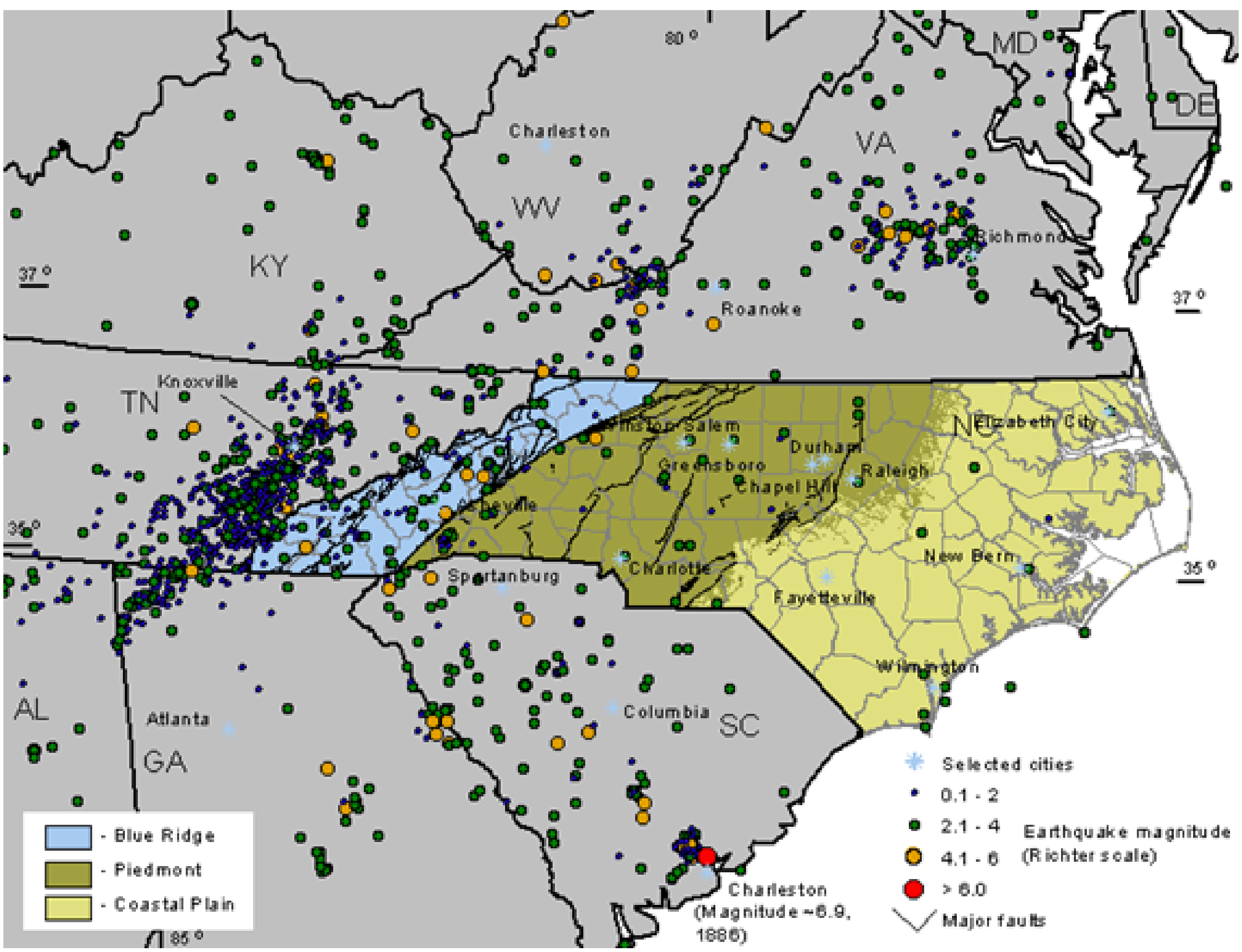


Figure 2-52 – Regional Map of Earthquake Epicenters – 1698 to 1997

Figure 2-52 indicates earthquake epicenters recorded in North Carolina and portions of adjacent states between 1698 and 1997. Major geologic provinces and known major faults exposed at the surface are shown for North Carolina. Faults identified to date in North Carolina are ancient and inactive. The lack of correspondence between the locations of earthquake epicenters and these faults indicates they are not responsible for earthquakes in North Carolina within historical times. The faults beneath the surface that generate earthquakes have yet to be positively identified.

Earthquake data before 1886 are sparse. Seismic instruments were installed in the region in the late 1920's. Prior to that time earthquake data are based on historical records. The distribution of seismograph stations did not allow for location of earthquakes with magnitudes less than 4 until 1962 – 1963. Micro-earthquake networks began operating in the region in the mid-1970s.

Within North Carolina, the largest recorded magnitude earthquake occurred on February 21, 1916 near Waynesville, NC with a magnitude of 5.2 (intensity VII). The isoseismal map for this earthquake is shown in Figure 2-53 and indicates an intensity of II to III for Raleigh, NC.[2-41]

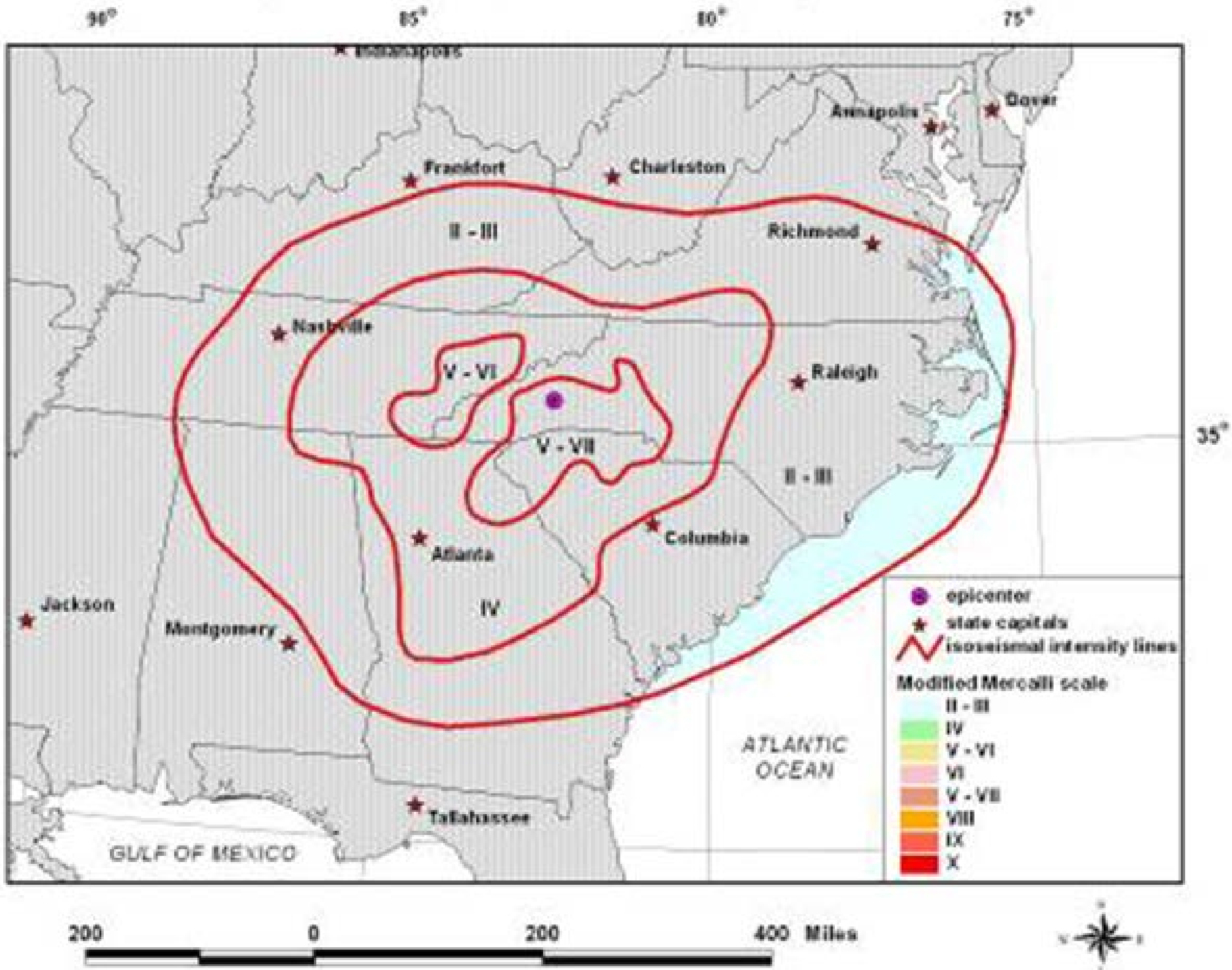


Figure 2-53 – Isoseismal Map for the Skyland, North Carolina, Earthquake of February 21, 1916

Seismic Activity Effects

The 1886 earthquake with an epicenter in Charleston SC was one of the most damaging earthquakes to occur in the Southeast United States and one of the largest historic shocks in Eastern North America. The meizoseismal area of MM intensity X effected an elliptical area, roughly 35 by 50 kilometers, trending northeast between Charleston SC and Jedburg SC and including Summerville SC. Seismic activity that still continues today may be a continuation of the 1886 aftershock series. The intraplate epicenter of this major shock is not unique for large earthquakes in the Eastern and Central United States. Earthquakes occurring along boundaries of plates (e.g., San Francisco, 1906) are well understood in terms of plate tectonics, but those occurring within plates are not similarly understood. This problem still is being studied more than 100 years after the earthquake. This earthquake was reported from distant places such as Boston, MA; Milwaukee, WI, Chicago, IL; Cuba and Bermuda. The isoseismal map for this earthquake is shown in Figure 2-54 and indicates an intensity of V in Raleigh, NC.[2-41]

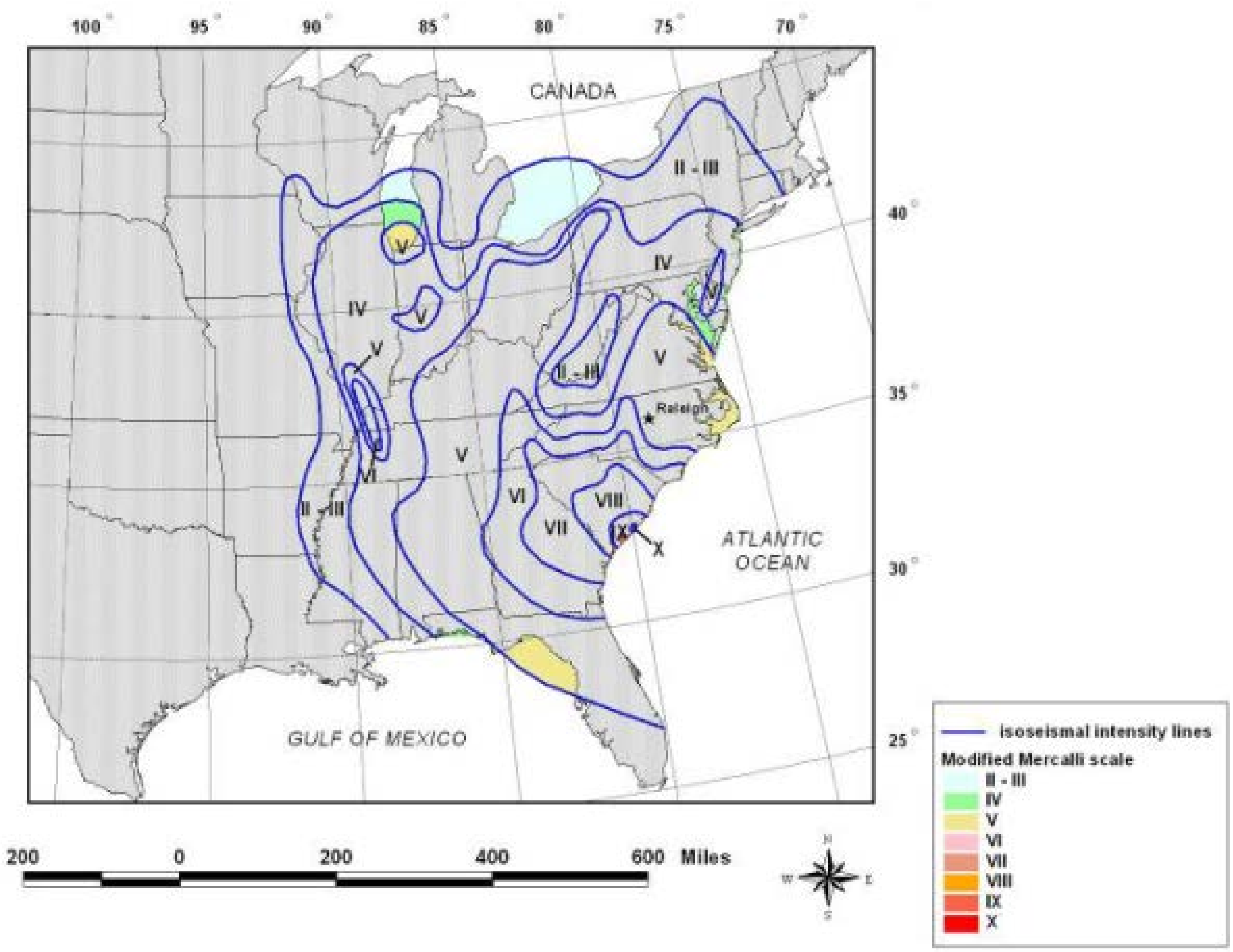


Figure 2-54 – Isoseismal Map for the Charleston, South Carolina, Earthquake of September 1, 1886

On August 23, 2011, a magnitude 5.8 earthquake occurred in Mineral VA (Louisa County) and was felt in Raleigh NC. Figure 2-55 provides the MMI map for this earthquake. Moderately heavy damage (MMI of VIII) occurred in a rural region of Louisa County southwest of Mineral. Widespread light to moderate damage occurred from central Virginia to southern Maryland including the Washington DC area. Minor damage was reported in parts of Delaware, southeastern Pennsylvania and southern New Jersey. The earthquake was very strongly felt (VII) in VA at Boston, Bumpass, Kent Store, Louisa, Mineral, Rhoadsville and Summerduck. Strongly felt in much of central Virginia and southern Maryland. Felt throughout the eastern US from central Georgia to central Maine and west to Detroit MI and Chicago IL. Felt in many parts of southeastern Canada from Montreal to Windsor. Felt weakly to lightly in Raleigh NC (III-IV).[2-41]

The Virginia earthquake of August 23, 2011 occurred as reverse faulting on a north or northeast-striking plane within a previously recognized seismic zone, the "Central Virginia Seismic Zone." The Central Virginia Seismic Zone has produced small and moderate earthquakes since at least the 18th century. The previous largest historical shock from the Central Virginia Seismic Zone occurred in 1875. The 1875 shock occurred before the invention of effective seismographs, but the felt area of the shock suggests that it had a magnitude of about 4.8. A magnitude 4.5 earthquake on December 9, 2003 also produced minor damage.

Previous seismicity in the Central Virginia Seismic Zone has not been causally associated with mapped geologic faults. Previous, smaller, instrumentally recorded earthquakes from the Central Virginia Seismic Zone have had shallow focal depths (average depth about 8 km). They have had diverse focal mechanisms and have occurred over an area with length and width of about 120 km, rather than being aligned in a pattern that might suggest that they occurred on a single causative fault. Individual earthquakes within the Central Virginia Seismic Zone occur as the result of slip on faults that are much smaller than the overall dimensions of the zone. It is estimated that there were about 450 aftershocks greater than M1.0 from August 24, 2011 to May 2, 2012. A couple hundred of the aftershocks greater than about M1.7 were felt locally. Many more aftershocks smaller than M1.0 (and unlikely to have been felt) have likely occurred in the epicentral area during this time.[2-41]

Earthquakes in the central and eastern US, although less frequent than in the western US, are typically felt over a much broader region. East of the Rockies, an earthquake can be felt over an area as much as ten times larger than a similar magnitude earthquake on the west coast.[2-41]

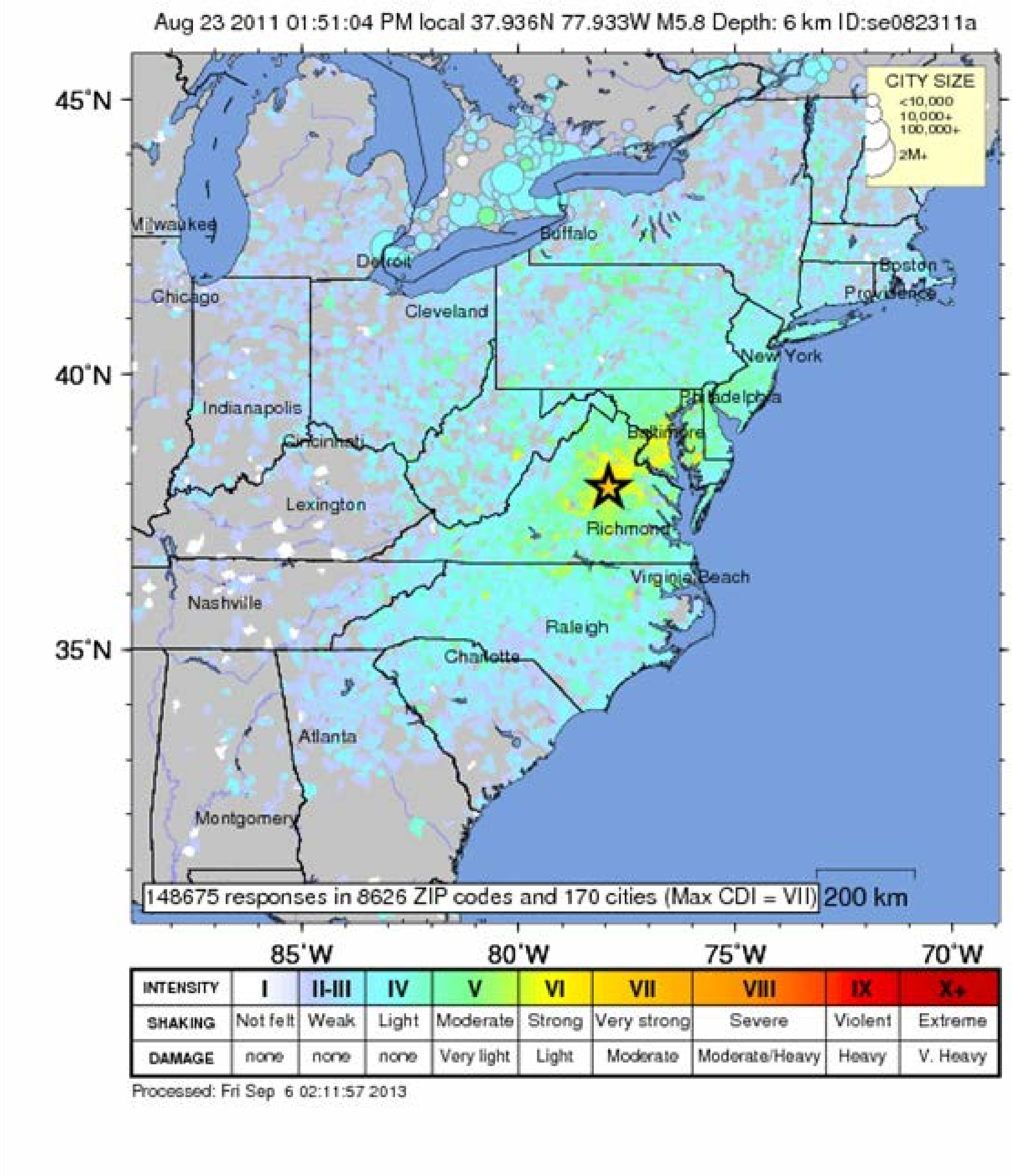


Figure 2-55 – Intensity Map for the Virginia Earthquake of 2011

2.5.4. Maximum Earthquake Potential

The maximum earthquake magnitude for the reactor site and region is estimated by the USGS to have a magnitude of 7.0(+0.2/-0.4) for a craton earthquake and 7.5(+0.2/-0.4) for a margin earthquake in the Central and Eastern United States (CEUS) as shown in Figure 2-56 and Figure 2-57. This estimate is based on based on tectonic and geologic principles rather than the seismic history catalog. The catalog of seismic events spans a time period that is a fraction of the recurrence times of the largest modeled events.[2-41]

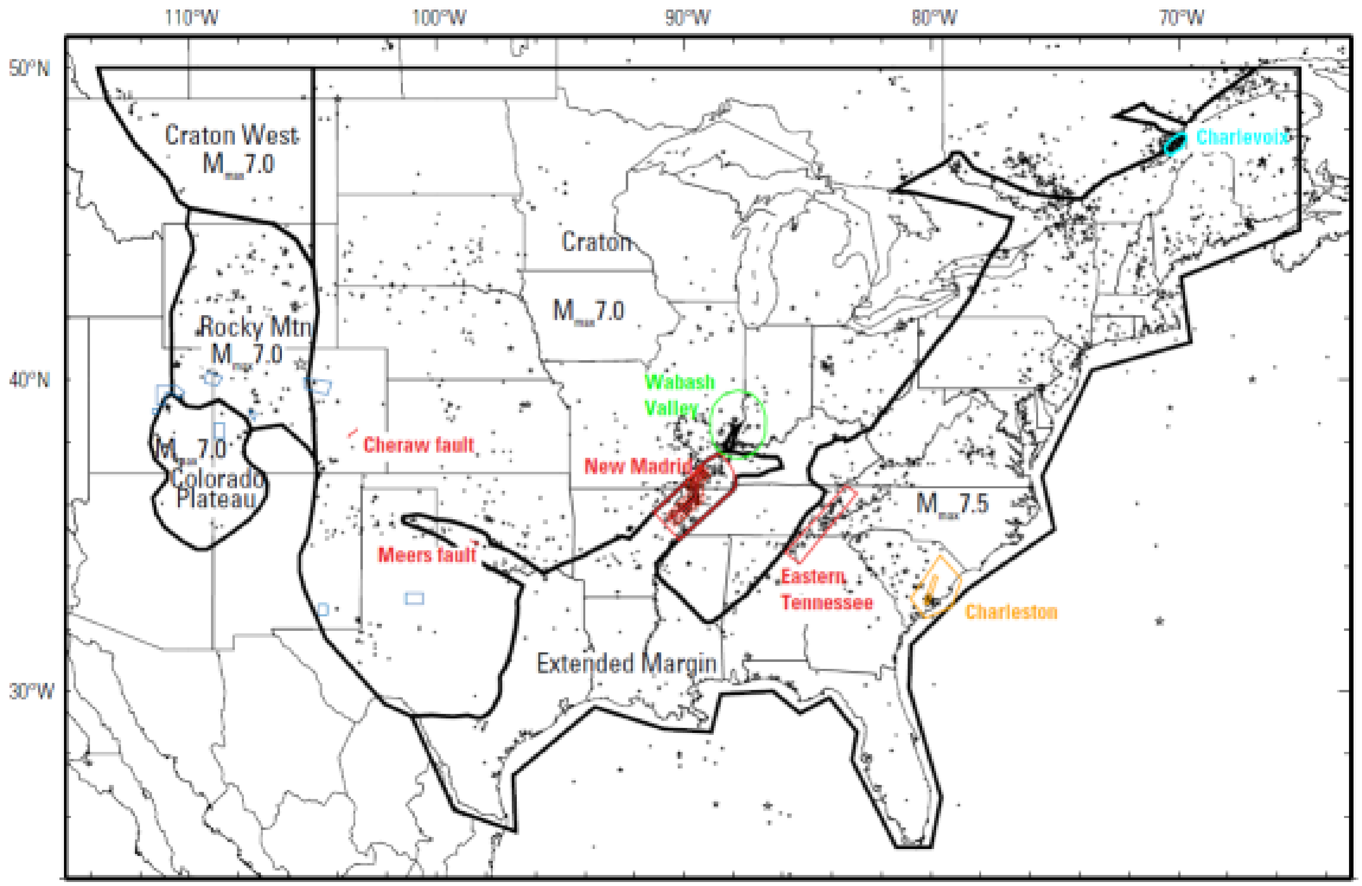


Figure 2-56 – Map of Special Zones, Faults, and Region M_{max} Zones in the Central and Eastern US

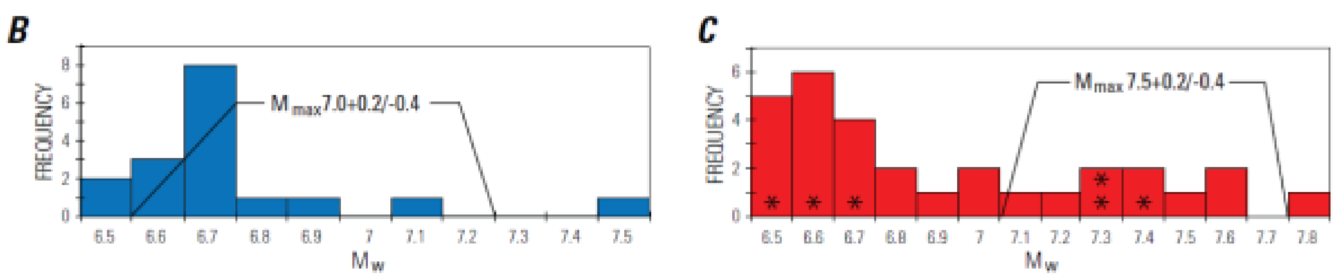


Figure 2-57 – Graph of the Magnitude of the Craton Earthquake (B) and the Margin Earthquake (C) for the CEUS

Earthquake probabilities were computed from the source model of the 2008 USGS National Seismic Hazard Mapping Project (NSHMP). The region of model validity includes the conterminous (lower 48 states) USA.[2-41]

Probability of an earthquake with a magnitude of 5 or greater occurring with 50 km of Raleigh, NC is estimated to be 0% to 1% in 170 years as shown in Figure 2-58.[2-41]

Figure 2-58 – Probability of Earthquake with M > 5.0 within 170 years and 50 km for Raleigh, NC

Probabilities of an earthquake occurring with 50 km of Charleston, SC estimated by the USGS National Seismic Hazard Mapping Project are as follows:[2-41]

1. 8% to 10% for magnitude of 5 or greater in 30 y
2. 4% to 6% for magnitude of 6 or greater in 30 y
3. 3% to 4% for magnitude of 7 or greater in 30 y
4. 0% to 1% for magnitude of 8 or greater in 30 y

The historical record indicates that the reactor site is safe from damage of major consequence from earthquakes.

2.5.5. Vibratory Ground Motion

In an earthquake, damage to buildings and infrastructure is related more closely to ground motion, rather than the magnitude of the earthquake. In severe earthquakes, damage is more often correlated with peak ground velocity. Vibratory ground motion is assessed by the parameters of peak ground acceleration (PGA) and spectral acceleration (SA).[2-41]

Peak ground acceleration (PGA) is a measure of the maximum force experienced by a small mass located at the surface of the ground during an earthquake. PGA is measured by instruments and generally correlates well with the Mercalli scale. For moderate earthquakes, PGA is an index of hazard for short stiff structures and the best determinate of damage for shorter buildings, up to 7 stories.[2-41]

Spectral acceleration (SA) is a measure of the maximum force experienced by a mass on top of a rod having a particular natural vibration period. Short buildings, less than 7 stories, have short natural periods, 0.2 to 0.6 s.[2-41]

SA is a unit measured in g (the acceleration due to Earth's gravity, equivalent to g-force) that describes the maximum acceleration in an earthquake on an object. The forces caused by shaking from an earthquake are measured as a percentage of gravity, or percent g. Specifically SA is measured as a damped, harmonic oscillator moving in one physical dimension. This can be measured at different oscillation frequencies and with different degrees of damping, although 5% damping is commonly applied.[2-41]

SA at a value related to the natural frequency of vibration of the building gives a closer approximation to the motion of a building or other structure in an earthquake than the PGA value, although there is normally a correlation between short period SA and PGA.[2-41]

Ground motion is typically quantified in terms of a median value and a probability density function of the PGA (horizontal) or SA. SA for 0.1 s, 0.2 s, 0.3 s, 0.5 s, 1.0 s, and 2.0 s have been reported by the USGS. Ss and S1 are used for the 0.2s SA (short period) and 1.0s SA, respectively.[2-41]

Probabilistic ground motion maps depict the earthquake hazard by showing (by contour values) the earthquake ground motions of a particular frequency that have a common given probability of being exceeded in a particular period of time, typically 50 years. The ground motions considered at a given location are those from all future possible earthquake magnitudes at all possible distances from that location. Probabilistic ground motion maps are based on modeling of future earthquakes, attenuation relations, and geologic site conditions. These maps relate the source characteristics of the earthquake and propagation path of the seismic waves to the ground motion at a site. Future earthquakes are modeled by the USGS using information about historical earthquakes, quaternary faults, and present crustal deformation (geodetic data). Attenuation relations provide ground motion change as a function earthquake magnitude and distance. Ground motion is affected by the site geology, especially the amount and type of material between the bedrock and the surface (i.e. alluvium or soil column).[2-41]

A building natural period indicates what spectral part of an earthquake ground-motion time history has the capacity to put energy into the building. Periods much shorter than the natural period of the building or much longer than the natural period do not have much capability of damaging the building. Thus, a map of a probabilistic spectral value at a particular period thus becomes an index to the relative damage hazard to buildings of that period as a function of geographic location.[2-41]

The USGS National Seismic Hazard Maps display earthquake ground motions for various probability levels across the United States. The maps incorporate findings on earthquake ground shaking, faults, seismicity, and geodesy. The resulting maps are derived from seismic hazard curves calculated on a grid of sites across the United States that describe the frequency of exceeding a set of ground motions.[2-41]

A probability of exceedance (PE) that a certain amount of ground motion and shaking will occur over a specified period is depicted in the USGS seismic hazard maps in the units of g or %g for PGA and SA. The maps are not actually probability maps, but rather ground motion hazard maps at a given level of probability. PE at 2%, 5%, and 10% for a period of 50 years are reported by the USGS.[2-41]

USGS seismic hazard maps for Raleigh NC are listed in Table 2-35 and shown in Figure 2-59 through

Figure 2-61. The reference site condition for the seismic hazard maps is the boundary between classes B and C (or firm rock) with an average shear wave velocity of 760 m/s in the top 30 m. Hard rock seismic hazard maps are also available in which the shear wave velocity is 3.0 km/s at the surface.[2-41]

Annual probability of exceedance, r, and PE are related. The relationship is given by the following equation:[2-41]

$$r = \frac{\ln(1 - [\%PE/100])}{Y}$$

Equation 2-1

$$r \sim \frac{[\%PE/100]}{Y}$$

Equation 2-2

Where Y is period in years, e.g 50 y.

The inverse of the annual probability of exceedance (1/r) is known as the "return period," which is the average number of years for an exceedance. Note that this exceedance is regarding the reported ground motion (g) for a given location. Data for Raleigh, NC is given in Table 2-34.

Table 2-34 – Ground Motion Probably of Exceedance for Raleigh, NC

%PE in 50 y	Annual PE r	Return Period 1/r, years	Estimated Annual PE, r	Return Period 1/r estimated
2	4.04E-04	2474.92	4.00E-04	2500
5	1.03E-03	974.79	1.00E-03	1000
10	2.11E-03	474.56	2.00E-03	500

Using the USGS seismic hazard data for Raleigh, NC for firm rock conditions and a ground motion of > 0.16 g may be associated with a moderate earthquake intensity (MMI ≥ IV or 4). This occurs with at 2% PE in 50y for SA at 3.33 Hz, 5 Hz, and 10 Hz. The associated return period is approximately 2500 years for 2% PE in 50 y.[2-41]

0.04 g to 0.08 g is estimated to cause weak shaking of buildings and occurs at 10% PE in 50y for SA of 3.33 Hz, 5Hz, and 10 Hz. 10% PE in 50y has an associated return period of approximately 500 years.[2-41]

Table 2-35 – 2008 USGA Seismic Hazard Data for Raleigh, North Carolina

Seismic Parameter	Hz	Period s	%PE in 50 y	Class B/C Firm Rock g	Conterminous US Firm Rock g	Class A Hard Rock g	Maximum g
PGA			2	0.07 - 0.09	0.06 - 0.08	0.05 - 0.06	0.09
			5		0.02 - 0.04		0.04
			10	0.02 - 0.03	0.02 - 0.03		0.03
SA	0.5	2	2		0.02 - 0.04		0.04
	0.5	2	5		0.02 - 0.04		0.04
	0.5	2	10		0.01 - 0.02		0.02
	1	1	2	0.06 - 0.07	0.06 - 0.08	0.05 - 0.06	0.08
	1	1	5	0.04 - 0.06	0.04 - 0.06		0.06
	1	1	10	0.02 - 0.03	0.02 - 0.03		0.03
	2	0.5	2		0.08 - 0.10		0.10
	2	0.5	5		0.04 - 0.06		0.06
	2	0.5	10		0.03 - 0.04		0.04
	3.33	0.3	2	0.14 - 0.19	0.12 - 0.16		0.19
	3.33	0.3	5		0.08 - 0.12		0.12
	3.33	0.3	10	0.04 - 0.06	0.04 - 0.06		0.06
	5	0.2	2	0.16 - 0.20	0.12 - 0.16	0.08 - 0.10	0.20
	5	0.2	5		0.08 - 0.12		0.12
	5	0.2	10	0.06 - 0.07	0.04 - 0.06		0.07
	10	0.1	2		0.12 - 0.16		0.16
	10	0.1	5		0.04 - 0.08		0.08
	10	0.1	10		0.04 - 0.05		0.05

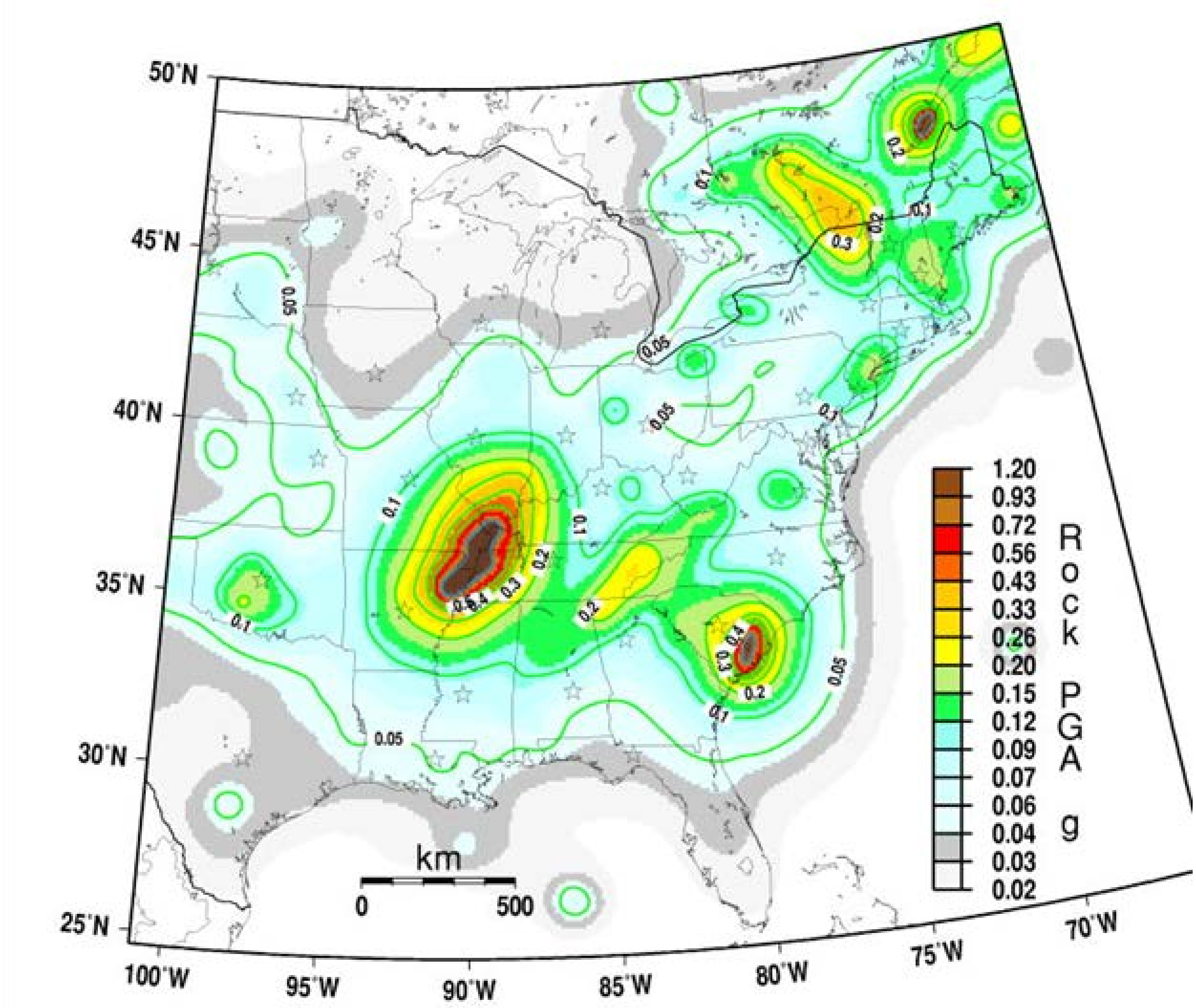


Figure 2-59 – CEUS PGA with 2%/50 Years PE, 2008

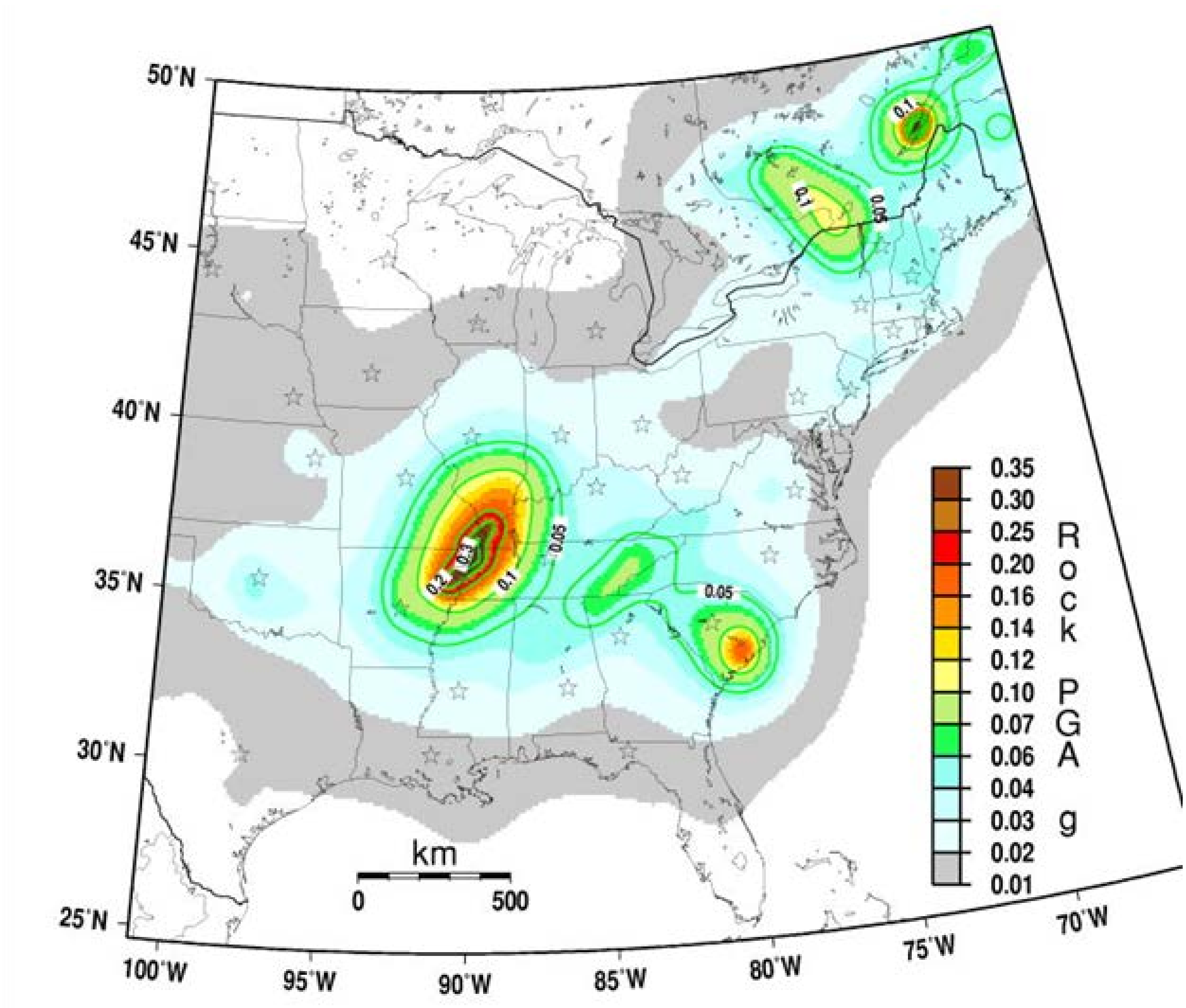


Figure 2-60 – CEUS PGA with 10%/50 Years PE, 2008

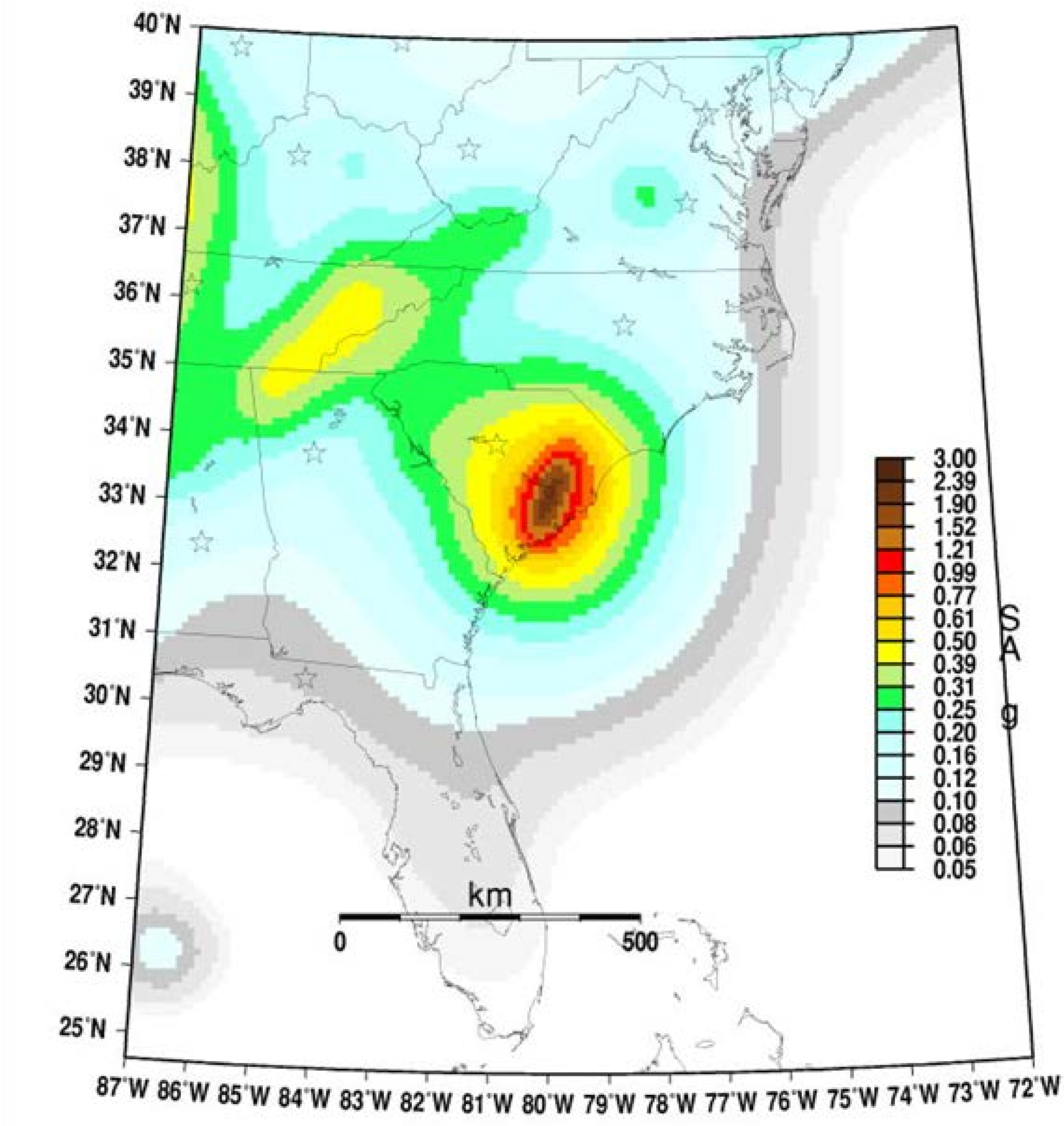


Figure 2-61 – South Carolina Region 0.2-s SA with 2%/50 Year

2.5.6. Surface Faulting

The reactor site is located in the Central and Eastern United States seismic region designated by the USGS. Within the CEUS there are four finite (magnitude 6 or greater) fault sources; New Madrid MO (and adjacent states), Charleston SC, Meers OK, and Cheraw CO. These four sources are the only ones in the CEUS that have paleoseismic data to constrain large earthquake recurrence rates.[2-41]

There are several ancient and inactive faults in Wake County and the Raleigh, NC area. These include the Nutbush Creek fault, Leesville fault, Fall Lake fault, and Jonesboro fault. The Nutbush Creek fault, which separates the Raleigh terrane from the Crabtree terrane, passes through the NCSU campus near the reactor site (Broughton Drive and Morrill Drive). This is a right lateral strike slip fault. The Nutbush Creek fault has shear zones from the late Paleozoic age. The Nutbush Creek fault extends from southern Virginia through Wake County (NC) to Lillington, NC and has a reported total length of approximately 180 km. The fault zone in the Raleigh area is diffuse and is reported to range from 1 km to 10 km wide. This fault is recognized by various geologic features described in Section 2.5.1 previously. The Nutbush Creek fault zone is one of many right lateral Alleghanian faults that affected the Piedmont.[2-3,2-41]

The Jonesboro fault is on the boundary between the Triassic Basins and Raleigh Belt. The Jonesboro fault is located near Cary, NC, which is approximately 10 miles west of the NCSU campus. The Jonesboro fault is a normal fault of Mesozoic age. These geologic features are illustrated.[2-3,2-41]

Another area with faults is the Deep River Coal Field located 45 miles southwest of Raleigh, NC. The faults in this basin were formed in the latter Triassic times, estimated to be from 155 to 165 million years ago.[2-3]

The immediate area of the state has had no displacements from faults as evidenced by the geologic record since Triassic times.

2.5.7. Liquefaction Potential

Liquefaction is a phenomenon where saturated sand and silt take on the characteristics of a liquid during the intense shaking of an earthquake. Areas of historical liquefaction that have occurred in the coastal plain near Wilmington, NC, coastal South Carolina, and central Virginia are listed in Table 2-36. There is no liquefaction noted historically in Raleigh, NC. Soil near the reactor site does not become saturated and does not have sand as a major constituent. Liquefaction potential is therefore not credible or historically observed at the reactor site.[2-41,2-44]

Table 2-36 – Areas of Quaternary and Historical Liquefaction for Coastal North and South Carolina

Number	Name
2657	Charleston liquefaction features
2658	Bluffton liquefaction features
2659	Georgetown liquefaction features
2652	Pembroke faults
2653	Central Virginia seismic zone

2.6. Conclusions

Potential accidents reviewed and a summary of the analysis is given in Section 2.2.3. From the analysis made it is concluded that the reactor facility is not adversely affected by site characteristics from land use and activities in the surrounding community, geology, hydrology, severe weather, or seismic activity.

Monitoring performed at the reactor facility is sufficient to detect environmental impact on the surrounding area and community. Radioactive releases and associated public dose are low and within regulatory limits.

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3. DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

This chapter describes the architectural and engineering design criteria used for various structures, systems, and components (SSCs) of the North Carolina State University (NCSU) PULSTAR reactor. Only those SSCs considered important for ensuring the safe operation of the facility and for the protection of personnel and the public from an exposure to a radiological release are included here. Detailed descriptions of most SSCs and their functions are detailed in the relevant sections of this SAR.

3.1. Design Criteria

The structures important to safe operation of the NCSU PULSTAR reactor include the reactor building and confinement system. All of these structures are part of the Burlington Engineering Laboratory building located on the North campus of North Carolina State University. Access to the reactor building is controlled by authorized personnel, with escorted access provided as necessary for other NCSU personnel and members of the public.

For the structures, systems and components of the PULSTAR reactor which have a vital role in the prevention or mitigation of the consequences of accidents which can cause undue risk to the health and safety of the public, one must assure that:

1. Agreed upon requirements for the components and systems, upon which bases the AEC construction permit was issued, are adequately and correctly delineated by specifications, drawing, procedures and instructions.
2. Purchased material and components fabricated in vendor shops conform to applicable specifications, drawings, procedures and instructions.
3. Components and systems are assembled, constructed, and tested in accordance with applicable specifications, drawings, procedures and instructions.
4. Succeeding phases of operations, fueling, maintenance and modifications of the PULSTAR reactor are conducted using quality assurance practices consistent with those employed during design and construction.

3.1.1. Quality Assurance Program

North Carolina State University implemented a Quality Assurance Program (QAP)[3-1] for the design and construction of the PULSTAR reactor in order to provide the required degree of assurance and permanent documentation that critical systems and structures were manufactured and installed in accordance with the applicable drawings and specifications. The technical requirements were specified in the appropriate detailed plans and specifications, along with the method of assuring that these requirements were adhered to and properly documented. Specified procedures were included in the assurance and quality control sections of the plan.

To assure that these objectives were met, North Carolina State University, the Architect and their design engineers (serving as site quality control engineers (SQCE)) and AMF jointly planned a program for all phases of equipment procurement, fabrication, and facility construction. The specific emphasis was directed to quality and conformance to applicable codes and specifications on those systems which affected nuclear safety. In addition, the program was implemented to assure the quality of other systems which directly affected the reliability of the reactor and experimental facilities.

The purpose of this detailed plan was to ensure that the engineering, material, equipment and workmanship employed in the construction of the PULSTAR reactor met the safety, operability, and objectives established by University, Architect, design engineers, and government and regulatory agencies.

The objectives of the QAP were:

1. To ensure the highest feasible degree of functional integrity, safety, and reliability of vital safety related materials, structures, components, and systems of the PULSTAR reactor.
2. To ensure the highest practical degree of performance of the PULSTAR reactor.
3. To ensure post operational surveillance and inspection for the PULSTAR reactor.

The program was characterized as a three level quality program. The responsibilities and actions of the three levels are summarized below.

3.1.1.1. Quality Control Inspection

Component manufacturers and construction contractors were responsible for providing appropriate quality control procedures, systems, and inspection personnel for assuring and demonstrating that the end product had the specified degree of quality as defined in the appropriate specifications drawing and/or purchase documents. The actual quality control efforts were executed by the equipment manufacturer or construction contractors. This was the first level of the assurance program.

3.1.1.2. Quality Assurance Surveillance

The Architect, through his design and site quality control engineers and inspection coordinator; and, the University, along with AMF, had surveillance responsibility for the quality control systems, procedures and efforts of component manufacturers and construction contractors.

During the actual fabrication and construction, the quality control procedures and programs of the component manufacturers and construction contractors were reviewed and approved as meeting the requirements of the specification and/or procurement documents. Physical surveillance actions were performed to ensure that the quality requirements were in fact met. This was the second level of the assurance program.

3.1.1.3. Quality Assurance Audit

To ensure that the NCSU QAP functioned as planned, periodic audits were performed by NCSU through the quality assurance coordinator, and/or authorized consultants. Specifications and other document requirements furnished by the design contractor were reviewed for the necessary quality requirements. In addition, on a spot check basis, the quality assurance surveillance actions of second level contractors and the quality control and inspection actions of manufacturers and constructors were audited to ascertain if they actually functioned as required. This was the third level of the assurance program.

3.1.2. Reactor Fuel

A PULSTAR fuel assembly consists of 25 rod type fuel pins arranged in a 5 × 5 array rectangular array and contained in a zircaloy box equipped with aluminum end fittings. The fuel pins are made of slightly enriched sintered UO_2 pellets hermetically sealed in a helium atmosphere in zircaloy sheaths with welded zircaloy end caps. The fuel pins are axially symmetric so that the vertical position can be reversed. The fuel pin end caps fit into holes and slots in the aluminum grid plates of the end fittings. Pin-to-pin spacing and pin-to-box spacing are maintained by zircaloy spacer pads brazed to the fuel pin sheaths.

The upper end fitting has an aluminum lifting handle. This end fitting can be removed from the outer box to permit removal and subsequent replacement of individual fuel pins. The lower end fitting has an aluminum nosepiece compatible with the reactor grid plate. The centerline of the nosepiece is offset slightly from the centerline of the fuel assembly so that rotation of adjacent assemblies by 180° will provide space to accommodate the control rods when installed in the core grid plate.

The fuel assemblies are designed to operate in light water at a nominal pressure of 15 psig and a nominal temperature of 100°F. The fuel pins are capable of a steady-state power level equivalent to a maximum rating of 300 watts/cm of fuel pin length and are rated for a burnup of 20,000 MWd/MTU.

The fuel assemblies are manufactured in accordance with the design dimensions in Specification for PULSTAR Fuel Assemblies APR-1, Revision 5 and drawing APR Fuel Element 89-113-60022 Revision D.[3-2,3-3] The material specifications are summarized in Table 3-1.

Inspection and laboratory evaluation procedures are carried out to assure that the quality of the PULSTAR fuel assemblies meet all specifications. All materials to be used in the manufacture of the fuel assemblies shall conform to the requirements of the ASTM specifications.

Table 3-1 – Material Specifications for PULSTAR Fuel Assemblies

Description	Material	Specification[3-2,3-3]
Fuel Pellet	UO_2 Enriched Sintered	CWAPD SPEC.F10 Issue 1[3-4]
Nosepiece	Aluminum	ASTM-B26-58T Alloy SG70A-T6[3-5]
Cladding	Zircaloy-2	ASTM-B353-64T[3-6] Grade RA-1 Temper H-12
Assembly Box	Zircaloy 2 or 4	ASTM-B351-64T[3-7] or ASTM-B352-64T[3-8] Grade RA-1
End Fittings	Aluminum	ASTM-B209-69[3-9] or ASTM-B211-69[3-10] or ASTM-B221-69 Alloy 6061-T6[3-11]
Fasteners	Stainless Steel 300 series	

3.1.3. Reactor Building

Concrete for the reactor building has a minimum compressive strength of 4000 psi for beams, joists, structural slabs, columns and walls while footings and floor slabs on grade have a minimum compressive strength of 3000 psi.[3-12,3-13] Reinforcing bars were installed according to ASTM A432[3-14] and welded wire fabric according to ASTM A185[3-15] and ACI building code.[3-16]

The structural steel of the building was constructed according to AISC specifications using steel meeting ASTM A36 specifications,[3-17] with bars meeting ASTM A15 and A305 specifications.[3-18, 19] All welds were made by individuals certified by the standard qualification procedure of the American Welding Society for the type of weld required.

During construction, the concrete and steel was inspected and tested according to the Quality Assurance Instructions Manual.[3-1] The structural SQCE inspected the quantity and placement of all reinforcing steel in the foundations, walls, floors, beams, and columns. The general contractor was not permitted to pour concrete until such inspection had been made and a report of compliance with the contract documents had been submitted.

3.1.4. Biological Shield

The site quality control engineer for the construction of the reactor building coordinated with the general contractor and the appropriate approved testing laboratory to develop the proper mix for the heavy concrete for the biological shield to meet the strength and density requirements of the specifications. The SQCE determined the suitability of the proposed heavy aggregate from the certified analysis, test, trial mixes, samples, etc. All concrete aggregates (barytes) were visually inspected for color, shape and general appearance.

The design of the forms and supports for the barytes concrete was submitted to the architect by the contractor for approval to ensure that the shielding capacity of the concrete was not reduced. The SQCE analyzed the design and determined acceptability. The Architect and SQCE inspected the embedded items prior to pouring the concrete for the biological shield, assuring that all items were substantially supported, braced and aligned to prevent them from moving out of position.

The soil conditions were inspected prior to pouring the foundation to verify the original soil test reports based on the subsurface investigation.

3.1.5. Pool Liner and Primary Piping

Liquid penetrant examinations were required for the following items:

1. Continuous welds in the pool liner.
2. Weld joints in stainless, carbon steel and aluminum piping.

Radiographic inspection was required for the following items:

1. Pool liner test welds.
2. All single welded joints in the aluminum liner.
3. Stainless, carbon steel and aluminum pipe test welds.
4. All single welded joints in the primary coolant system buried piping.

Leak tests were required for the aluminum pool liner and pressure leak tests were required for plumbing and equipment in accordance with the requirements outlined in the specification. The primary coolant was subjected to a hydrostatic leak test. Proof of acceptability or code stamping was required for finished installation, and pressure and/or leak rate tests were witnessed. Equipment and piping buried or otherwise inaccessible after installation were inspected and pressure tested before it was buried or made inaccessible.

All steel piping assembly welds were hammer tested with a 3 pound hammer in accordance with ASA A37.1[3-20] and visually inspected for leaks while at test pressure. The primary system was tested to 200 psi for 2 hours and embedded piping was tested to 250 psi for 2 hours.

All welder and welding process met the qualification tests in accordance with the ASA and/or ASME codes as follows:

1. Aluminum pool liner – ASME boiler and pressure vessel code section 1X, Welding Qualifications. Aluminum alloy 6061-T6 was used for the qualification test plates.[3-21]
2. Carbon Steel piping – Paragraph UW-28 and UW-29 of the ASME Unfired Pressure Vessel Code.[3-22]
3. Stainless steel piping – Paragraphs UW, UG, and either UNF or UHA of Section VIII and SECTION 1X of the ASME code as applicable.[3-21]

3.2. Meteorological Damage

Section 2 of this report summarizes the meteorological history of the NCSU PULSTAR reactor site. The principal characteristic is an absence of extreme conditions. The reactor building provides more than adequate protection against weather-related phenomena.

3.2.1. Wind Loading

The NCSU PULSTAR core is protected from damage by wind (i.e. high winds from storms, tornados and hurricanes) by virtue of the steel reinforced concrete biological shield which in turn is located inside the steel reinforced poured concrete reactor building. The reactor building has been designed to withstand a loading of 30 pounds per square feet (psf).[3-12,3-13] The wind speed charts from ACSE 7-05 Figure 6-1B[3-23] show that Raleigh, NC is located between the 90 mph and 100 mph zones with an extrapolated value of approximately 92 mph. The 100 year mean recurrence interval is 96 mph.

Using

$$p = 0.00256 \times v^2 \qquad \textit{Equation 3-1}$$

Results in an uncorrected wind speed of 108 mph.

$$v = \sqrt{\frac{30}{0.00256}} = 108\, mph \qquad \textit{Equation 3-2}$$

The result does not take credit for surrounding obstructions and is therefore a conservative value. Section 2 discusses these weather conditions in detail.

3.2.2. Snow and Ice Loading

The live load rating of the reactor building roof structure is 30 psf.[3-12,3-13] The weight of 1 foot of fresh snow ranges from 3 psf for light, dry snow to 21 psf for wet, heavy snow.[3-24] Therefore the roof can support nearly 18 inches of heavy wet snow. The annual snow fall for the Raleigh area is approximately 4 inches.[3-25] The local building code requires that for commercial buildings the snow load rating of no less than 15 psf.[3-23]

One inch of ice weighs a little less than 5 psf, and 1 foot of ice weighs approximately 57 psf. Ice storms usually in conjunction with snow do occur in Raleigh but the total ice accumulation is typical less than ½ inch per occurrence.

3.3. Water Damage

As discussed in Section 2, flooding is not expected at the NCSU PULSTAR. The reactor building is located on relatively higher ground compared to other local buildings and features. In the event of storm sewer malfunctions, storm water from localized heavy rains could potential backup into the reactor building spaces. Equipment, such as instrumentation and control, control rods, radiation monitors, etc. would be unaffected therefore the safety and security of the reactor would not be jeopardized.

3.4. Seismic Damage

The USGS lists the probability of an earthquake resulting in accelerations larger than 0.08 g is less than 2% in 50 years.[3-26] Refer to Section 2.5.5 and Figure 2-59. Seismic intensity at the reactor site is historically moderate or less (Modified Mercalli Intensity Scale (MMI) < V). USGS has estimated the probability of an earthquake having a MMI > V within 50 km of Raleigh as 0 to 1% in 170 years (refer to Sections 2.5.3 – 2.5.7). Slight damage is expected for ordinary structures and negligible damage in structures of good design for MMI of VII or less. Slight damage is expected in structures of good design for MMI of VIII and foundation damage occurs at MMI > VIII. The Reactor Building and associated reactor components were designed and constructed in accordance to all applicable building codes at the time, therefore are of sound design and construction. In the unlikely seismic event with a MMI > V, minimal damage would be expected to the Reactor Building, reactor pool liner and primary coolant system. Furthermore, failure of the reactor pool liner and/or primary coolant system causing a loss of pool water leading to uncovering of the core would not result in fuel failure. Refer to Section 13 for details of the analysis. All other systems, if damaged, would not affect the safe and secure shutdown of the reactor.

3.5. Systems and Components

The NCSU PULSTAR structures, systems, and components whose integrity is important to preventing the release of radioactive material, preventing core damage, and controlling reactivity are designed to facilitate inspections, testing, and maintenance. Some examples include:

1. Acceptance of fuel elements.
2. Visual inspection of the material condition of all in-core components.
3. Confinement filter train testing.
4. Verification of control rod drop times.
5. Channel checks and calibrations of the nuclear safety systems.

There are approved procedures for conducting required surveillances, inspections and tests of all systems.

3.5.1. Reactor Fuel and Fuel Storage

The specification for fuel is described in Section 3.1.2. To maintain the integrity of the fuel cladding total burnup is limited to 20,000 MWd/MTU.

To ensure sub-criticality, reactor fuel shall be stored in a geometrical configuration where k_{eff} is not greater than 0.9 for all conditions of moderation and reflection using light water. The k_{eff} for each storage rack in the reactor pool has been measured and verified to be less than 0.9.

3.5.2. Control Rods and Drive Mechanisms

Control of the reactor is attained through manipulation of four control rods. Each is actuated by means of an in-line drive mechanism mounted on the bridge over the core. The drive package consists of a motor, a gear reduction system, and an acme screw type drive. The limits of the stroke are set by adjustable, cam operated switches mounted on the inside of the control rod drive mechanisms (CRDM). The standard stroke for the rod drive is twenty-four inches. The CRDM can be positioned in a variety of locations in the core grid plate.

The drive speed is a function of the reduction gear system and acme thread pitch and is fixed at 7.5 inches per minute. The drive speed was chosen to limit the rate of reactivity insertion when moving the control rods. The technical specification limit is 200 pcm/sec when in the critical region. Each control rod is connected to a CRDM by an electromagnet which is de-energized following a scram signal. During a scram, the technical specification required time for a control rod to go from its full-out position to the seated position is less than 1.0 second. The drive speed and rod drop times are verified annually but not to exceed fifteen (15) months and after a control rod assembly is moved to a new position or after maintenance is performed on the CRDM. Control rods are visually inspected biennially but at intervals not to exceed thirty (30) months.

3.5.3. Flapper Mechanism

During normal operation of the reactor at power levels above 100 kW, core cooling is maintained with a 1000 gpm primary coolant flow rate. In the event that the primary flow is interrupted by loss of the pump or other causes, a flapper valve on the plenum, which is located directly under the grid plate, will open and provide a path for natural convection cooling to be established within the reactor pool. The flapper valve is held in a closed position by the differential pressure created by the coolant flowing through the core and the static head of the pool water at the plenum level. The normal operating position of the flapper valve during forced convection flow cooling is closed.

In the event that the coolant flowrate drops below the safety system setpoint of 950 gpm and the power level is above the safety system setpoint of 150 kW, a scram signal will occur for low primary flow. Calculations show that fuel and cladding temperatures remain well below the safety limits during the flow reversal and transition to natural circulation cooling, even in the event that the flapper fails to open. Refer to Section 13.2 for more details.[3-27]

The proper operation of the flapper and associated scram circuits are tested prior to operation, and prior to startup the flapper is verified to be open or closed depending on the mode of operation.

3.5.4. Reactor Safety Systems

The instrumentation for the NCSU PULSTAR reactor includes both nuclear and non-nuclear channels using electronic signals. Also included is the scram logic unit and associated trip circuits that make up the Reactor Safety System. A combination of alarms, interlocks, drive inhibits and reverse drive functions are provided for the safe and efficient operation of the reactor. Trips are referred to frequently as fail-safe. This means that upon loss of electrical power to an instrumentation channel, all trip circuits associated with the channel will act to limit reactor power or initiate reactor shutdown.

The NCSU PULSTAR Reactor Safety System (RSS) features predominately automatic shutdown mechanisms. The RSS is defined as that specified combination of instrumentation channels and associated circuitry which either provides the automatic protective action or provides the alarm which requires that manual protective action be taken. Specifically, the RSS consists of the scram logic unit with the magnet current circuits and the protective instrumentation channels and the associated circuitry.

A reliable reactor protection system functions to ensure that all modes of reactor operation are safe; therefore, the protective actions of the system are designed to automatically terminate operations should safe operating conditions cease to exist.

The reactor instrumentation and safety system is verified to be capable of performing its design function by the regular performance of the following tests and calibrations:

1. A channel check of each measuring channel of the RSS shall be performed daily when the reactor is in operation.
2. A channel test of each channel in the RSS shall be performed prior to operation each day, or prior to each operation extending more than one day.
3. A channel calibration of the ^{16}N Channel shall be made semi-annually, but at intervals not to exceed seven and one-half (7½) months. A calorimetric measurement shall be performed to determine the ^{16}N detector current associated with full power operation.
4. A channel calibration of the following channels shall be made semi-annually but at intervals not to exceed seven and one-half (7½) months:
    i. Pool Water Temperature.
    ii. Primary Flow Measuring Channel.
    iii. Flow Monitoring Channel (Flapper Mechanism).
    iv. Primary Heat Exchanger Inlet and Outlet Temperature.
    v. Linear and Safety Power Channels.
    vi. LogN Power Channel.
    vii. Source Range Channel.

3.5.5. Ventilation System

Several potential sources of radioactive particulates and gases exist at the PULSTAR reactor. These vary from the production of ^{41}Ar gas in the beamtubes or similar facilities to a failed experiment or a ruptured fuel pin. The handling of potentially radioactive effluent during both normal and confinement conditions requires an adequate ventilation system capable of minimizing uncontrolled releases to the environment and providing the basic requirement of ventilation for personnel and equipment. The size of the ventilation system is based on the magnitude of the release of potential sources and is discussed in Section 13.

3.5.6. Confinement System

The confinement as opposed to the containment concept was shown to mitigate the radiological consequences of the most severe credible reactor accidents. The confinement system was designed to provide a controlled release path for radioactive particulates and gases so that they can be filtered and passed up the stack for subsequent atmospheric dispersion.

The confinement system is verified to be capable of performing its design function by the regular performance of the following tests and calibrations:

1. The confinement and evacuation system shall be verified to be operable within seven (7) days prior to reactor operation.
2. Operability of the confinement system on auxiliary power will be checked monthly but at intervals not to exceed six (6) weeks.
3. A visual inspection of the door seals and closures, dampers and gaskets of the confinement and ventilation systems shall be performed semi-annually but at intervals not to exceed seven and one-half (7 ½) months to verify they are operable.
4. The control room differential pressure gauges shall be calibrated annually but at intervals not to exceed fifteen months.
5. The confinement filter train shall be tested biennially but at intervals not to exceed thirty (30) months and prior to reactor operation following confinement HEPA or carbon adsorber replacement. This testing shall include iodine adsorption, particulate removal efficiency and leak testing of the filter housing.
6. Air flow rate in the confinement stack exhaust duct shall be determined annually but at intervals not to exceed fifteen (15) months. The air flow shall be not less than 600 CFM.

There are approved procedures for conducting required inspections and tests of all systems.

3.5.7. Liquid Radioactive Drain System

All liquid waste in the reactor building and select adjoining laboratories is treated as potentially contaminated, therefore the drains join into one system that empties into a sump located in the reactor building. As this sump fills, the wastewater is pumped into secondary holding tanks where it is processed, sampled and, upon meeting all regulatory requirements, is disposed of via the sanitary sewer system.

Two roof drains also penetrate the reactor building. However, they are not part of the building drainage system since they simply pass through it and out again into the storm sewer system. They are completely sealed from the interior of the reactor building so that they do not breach the confinement system.

3.6. References

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3-3 American Machine and Foundry Co., *AMF Drawing 89-113-60022-NCSU PULSTAR Reactor APR Fuel Element*, April 1970.

3-4 Canadian Westinghouse Atomic Power Division (CWAPD), Spec F10 Issue 1.

3-5 ASTM B26-58T-*Standard Specification for Aluminum-Alloy Sand Castings.*

3-6 ASTM B353-64T-*Standard Specification for Wrought Zirconium and Zirconium Alloy Seamless and Welded Tubes for Nuclear Service (Except Nuclear Fuel Cladding).*

3-7 ASTM B351-64T-*Standard Specification for Hot-Rolled and Cold-Finished Zirconium and Zirconium Alloy Bars, Rod, and Wire for Nuclear Application.*

3-8 ASTM-B352-64T-*Standard Specification for Zirconium and Zirconium Alloy Sheet, Strip, and Plate for Nuclear Application.*

3-9 ASTM B209-69-*Standard Specification for Aluminum and Aluminum-Alloy Sheet and Plate.*

3-10 ASTM B211-69-*Standard Specification for Aluminum and Aluminum-Alloy Bar, Rod, and Wire.*

3-11 ASTM B221-69-*Standard Specification for Aluminum and Aluminum-Alloy Extruded Bars, Rods, Wire, Profiles, and Tubes.*

3-12 Wheatley, Whisnant and Associates, *Specifications of Material and Labor for the Erection and Completion of the Nuclear Science and Engineering Research Center-General Contract*, March 1968.

3-13 Wheatley, Whisnant and Associates, North Carolina State University-Nuclear Science and Engineering Research Center, *Drawing S9-General Notes and Structural Details*, March 1968.

3-14 ASTM A432-*Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Point.*

3-15 ASTM A185-*Standard Specification for Steel Welded Wire Reinforcement, Plain, for Concrete.*

3-16 American Concrete Institute (ACI) Building Code.

3-17 ASTM A36-*Standard Specification for Carbon Structural Steel.*

3-18 ASTM A15-*Specifications for Billet Steel Bars for Concrete Reinforcement.*

3-19 ASTM A305-*Specifications for Minimum Requirements for the Deformations of Deformed Steel bars for concrete Reinforcement.*

3-20 ASA A37.1

3-21 ASME Boiler Code and Pressure Code.

3-22 ASME Unfired Pressure Vessel Code.

3-23 ACSE 7-05-*Minimum Design Loads for Buildings and Other Structures.*

3-24 FEMA, P-958-*Snow Load Safety Guide*, January 2013.

3-25 State Climate Office of North Carolina, Data Archives.

3-26 US Geological Survey (USGS), Earthquake Hazards Program.

3-27 North Carolina State University PULSTAR Reactor, *Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor*, February 2019.

4. REACTOR DESCRIPTION

4.1. Summary Description

The North Carolina State University PULSTAR Reactor was manufactured by the American Machine and Foundry Company (AMF) and its design, fabrication, and installation are based on the proven prototype that was located at the Buffalo Materials Research Center (BMRC) at the State University of New York at Buffalo.[4-1] The NCSU PULSTAR is a thermal light water moderated and cooled reactor that is UO_2 fueled (with low enrichment (4% and 6%) in ^{235}U). It has a heterogeneous core with pin-type fuel. Its initial criticality was achieved in September 1972. It is operated by the Department of Nuclear Engineering within the College of Engineering. The PULSTAR reactor operates at steady-state power levels of up to two megawatts.

The NCSU PULSTAR core provides a source of neutrons and gamma-rays for research purposes. The core is immersed under twenty feet of water in an open pool, surrounded on the sides and bottom by concrete shielding. Experimental access to core neutrons is provided by five horizontal beam tubes, one through (tangential) tube, and a thermal column. Core access is available by direct insertion of experiments through the pool water into flooded or dry exposure tubes on the core periphery. Refer to Figure 4-7 and Figure 4-8 for plan and elevation views of the PULSTAR reactor.

Control of the reactor power level is maintained by the variable positioning of neutron absorbing rods within the core. The control rods, as well as the various neutron detecting chambers used for power level determination, are suspended from a bridge which spans the top of reactor pool. The reactor is operated from a console located in a control room adjacent to the reactor bay.

Core heat is removed by the forced convection of pool water through the core. Before being returned to the pool, this water is held up for ^{16}N decay and cooled by a plate type heat exchanger. A 20 gallon per minute (gpm) bypass flow of primary water is continuously demineralized and filtered before being returned to the main primary coolant piping.

The reactor fuel was selected and designed to take advantage of proven characteristics of power reactor type fuel, specifically Doppler broadening, high retention of fission products by the matrix, and low thermal conductivity. In particular, the low enrichment and low thermal conductivity of UO_2 fuel results in a substantial negative temperature reactivity feedback effect due to Doppler broadening of ^{238}U neutron resonances. This inherent reactivity feedback effect serves to mitigate uncontrolled reactivity excursions. Furthermore, the low thermal diffusivity of UO_2 leads to a long thermal time constant for the fuel (approximately 4 seconds). The long time constant prevents the explosive formation of steam that has previously been experienced in plate-type metallic reactors undergoing severe reactor transients. The heat capacity of UO_2 is quite large permitting a large release of energy in the core under transient conditions without exceeding the melting point of the fuel or cladding.

The operation of the NCSU PULSTAR reactor since 1972 has provided sufficient experience to permit reliable prediction of core performance.

4.2. Reactor Core

The reactor core for the NCSU PULSTAR consists of up to 25 fuel assemblies in a rectangular aluminum grid plate that is mounted on a plenum that serves as the transition to the 10-inch primary coolant outlet pipe. The core is reflected by a combination of graphite and/or beryllium reflectors located on the periphery. Four identical control rods are installed in the core grid plate. The locations of the control rods can be varied to accommodate different core configurations.

Surrounding the core, located in the thermal neutron reflector peak, are the experimental facilities. These facilities include beamtubes, rotating exposure ports, dry exposure ports, a thermal column, and the pneumatic sample terminus. The experimental facilities are discussed in detail in Section 10.

[redacted]

Figure 4-1 – Core Plan View for Reflected Core No.11 detailing locations of fuel assembly, beryllium reflectors, control rods, nuclear instrumentation, and experimental facilities.

4.2.1. Reactor Fuel

Each of the 25 fuel assemblies which comprise the reactor are composed of 25 fuel pins, as shown in Figure 4-2. The pins are fastened mechanically into bundles and are placed in a zircaloy-2 box open at the top and bottom with an external cross section measuring 2.74 inches by 3.15 inches. The upper end fitting and the lower end fitting (nosepiece) are attached, bringing the overall length of the assembly to 38 inches. A bail is inserted between side plates at the top of the assembly to serve as a handle for moving the assembly. There are also two alignment holes in the shoulder of the lower end fitting which mate with pins on the grid plate to prevent misalignment of the assemblies in the core. Openings are provided at the sides of each fuel assembly box to allow coolant flow should the top of the fuel assembly become blocked by a foreign object. The dimensions of the PULSTAR fuel assemblies are in accordance with American Machine and Foundry (AMF) Specification APR-1 Revision 5.[4-2]

[redacted]

Figure 4-2 – Exploded View of a NCSU PULSTAR Fuel Assembly

Table 4-1 – Fuel Assembly Dimensions

[redacted]

~~WITHHOLD FROM PUBLIC DISCLOSURE UNDER 10 CFR 2.390~~

Each fuel pin, as shown in Figure 4-3, consists of a zircaloy-2 tube with a nominal wall thickness of 0.022 inches, filled with sintered UO_2 pellets and sealed at the top and bottom. The uranium is enriched to four or six percent by weight in the ^{235}U isotope. OEM specifications for the fuel pellets and pins are given in Table 4-2 and

Table 4-3. The finished fuel pin is [redacted]. Approximately [redacted] of ^{235}U are contained in each 4% fuel pin, while [redacted] of ^{235}U are contained in each 6% fuel pin.

[redacted]

Figure 4-3 – NCSU PULSTAR Fuel Pin

Considerable experience and information has been gathered on the characteristics of zircaloy clad UO_2 fuel under irradiation conditions. The chemical and radiation stability of UO_2 is known to be excellent as well as its ability to retain fission products. The high melting temperature and corrosion resistance of zircaloy make it ideal for use as cladding of reactor fuel.

Table 4-2 – Fuel Pellet and Pin Dimensions

[redacted]

Table 4-3 – Fuel Pellet Material Properties

Parameter	OEM Specifications[4-2]	
	4% enriched ^{235}U	6% enriched ^{235}U
Enrichment	4% ± 0.056%	6% ± 0.056%
Density	10.4-10.76 g/cm^3	10.2-10.7 g/cm^3
[redacted]		

One of the strongest arguments for the inherent safety of the PULSTAR fuel is its similarity to the SPERT oxide core, which has undergone extensive testing.[4-4] Transients releasing as much as 100 MW·sec of energy have been initiated in the SPERT core without causing damage. The use of sintered pellets instead of powdered fuel is an added safety feature. The only serious defect discovered in SPERT tests was the presence of a double peaked pulse caused by coherent bowing of the fuel pins. The coherent bowing was eliminated when the pins were supported in a fashion that provided an 18 inch unsupported length of pin instead of the unsupported length of 6 feet. The PULSTAR fuel is designed to provide an unsupported length of only eight inches so that coherent bowing is not a problem. These supports also serve as wear surfaces to prevent damage to the fuel pin cladding.

A PULSTAR reactor was operated at a power of 2 MW at the BMRC in Buffalo, New York, from 1964 to 1995 with UO_2 fuel enriched to 6% in ^{235}U and reaching burnup limits that exceed 15,000 MWD/MTU. Based on the BMRC experience,[4-1,4-5] the operational experience of the NCSU PULSTAR to date, and the supporting analysis,[4-6,4-7] the performance of the NCSU PULSTAR core under mixed (4% and 6%) enrichment conditions is shown to meet the licensing limits as set forth in this SAR and technical specifications.

4.2.2. Control Rods

Control of the reactor is maintained through the manipulation of four control rods. Each is actuated by means of an in-line drive mechanism mounted on the bridge over the core. A control rod assembly, including the control rod drive mechanism (CRDM), is shown in Figure 4-4. OEM specifications for certain control rod components are provided in Table 4-4. The drive package consists of a motor, a gear reduction system, and an acme screw type drive. The limits of the stroke are set by adjustable cam operated switches mounted on the inside of the CRDM. The standard stroke for the PULSTAR rod drive is twenty-four inches. The CRDM can be positioned in a variety of locations on the core grid plate. Each control rod is connected to a CRDM by an electromagnet which can be de-energized in less than 50 milliseconds following a scram demand.

Table 4-4 – Control Rod Component OEM Specifications

Parameter	OEM Specifications[4-8]
Guide Area	0.438” × 6.34” (±0.031”)
Absorber Area	0.180” × 4.85” (±0.015”) x 29” (±0.0.63)”
Clearance, Absorber to Guide	0.025” × 0.066” (±0.017”)

One of the control rod assemblies has been selected as the regulating rod, while the remaining three serve as safety rods. The regulating rod is identical to the other three control rods with the addition that it may be positioned automatically in response to a power demand setting of the Automatic Power Control System, provided the rod is above a specified height determined by the positioning of a limit switch.

All rods are suspended from the bridge directly over the core and extend downward into the core where they are fitted with flat absorber blades of comprised of a silver-indium-cadmium alloy (80%-15%-5%) measuring approximately 0.18 inches thick by 24 inches long. Each absorber blade has a thin protective tin-nickel coating. Out-of-water type scram magnets and armatures are provided at the upper end of each control rod extension. A drawing of the control rod drive package is shown in Figure 4-4.

The control rods are manually actuated from the control console either individually or as a gang. Lights mounted on the control console illuminate as each drive is coupled to the gang switch, while rod contact indication is provided by a switch in the magnet which also provides a status displayed on the console. Vertical control rod position indication is generated by an absolute encoder and is displayed on the console by individual meters. Linear scales are attached to the control rod drive mechanism housings to provide a backup means of verifying control rod positions. The design drive speed for the control rods is 7.5 inches/minute. The maximum reactivity insertion rate for the control rods is limited to 200 pcm/second in the critical region. A startup accident resulting from a continuous gang rod withdrawal is analyzed in Section 13.

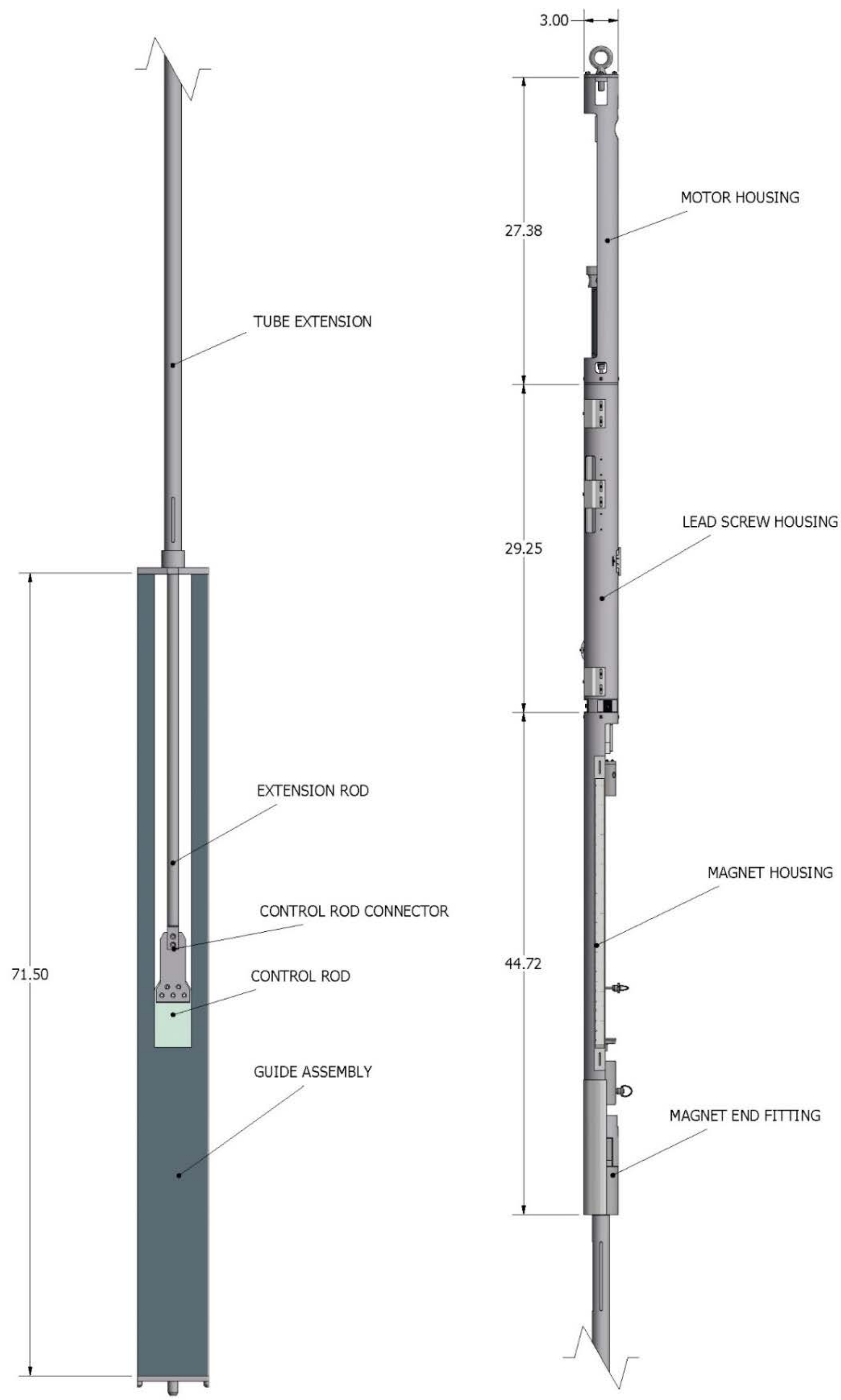

Figure 4-4 – Control rod and control rod drive assembly detailing neutron absorbing blade, rod drive motor and electromagnet.

4.2.3. Neutron Moderator and Reflector

The reactor core is light water moderated with enhanced reflection provided by reflector assemblies comprised of either graphite or beryllium, located on the core periphery. With the exception of the reflecting material, the graphite and beryllium reflectors are identical in construction. See Figure 4-5.

Starting with Reflected Core No.8 and continuing to the current Reflected Core No.11, a total of 10 beryllium reflectors are located on the periphery of the reactor core.

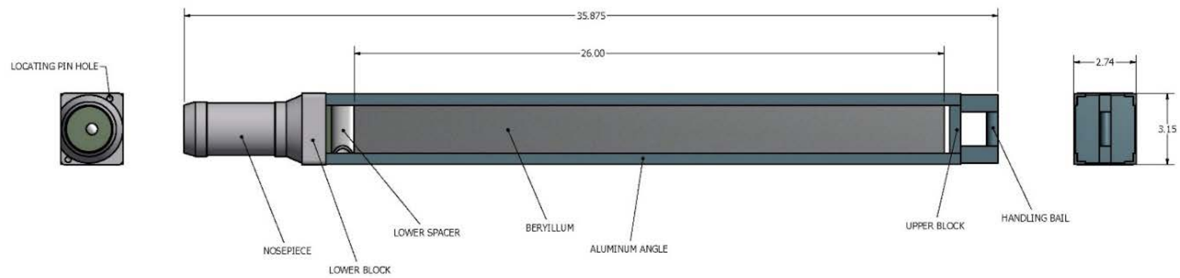

Figure 4-5 – Beryllium Reflector

4.2.4. Neutron Startup Source

A 5 Ci plutonium-beryllium (PuBe) startup source is available for use as needed. Due to the strong photo-neutron source term from the beryllium reflectors, the PuBe startup source is rarely needed during startup and is typically only utilized as a check source for the testing of the nuclear instruments during the reactor startup checklist procedure. When not in use, the startup source is stored in a holder located in the pool.

4.2.5. Core Support Structure

Fuel and reflector assemblies are positioned in a machined aluminum core grid plate. The holes are bored on a 6 by 6 rectangular pitch array, see Figure 4-6. The grid plate and fuel are mounted on top of the plenum chamber which channels the coolant flow from the fuel assemblies into the outlet coolant pipe located near the center of the pool floor. This pipe serves as a support for the entire plenum and core structure. The core support structure also provides a positioning rack for the neutron detectors. An aluminum frame support structure, resting on the pool floor and bolted to the east wall, supports a noseport assembly for experiments utilizing the reactor thermal column.

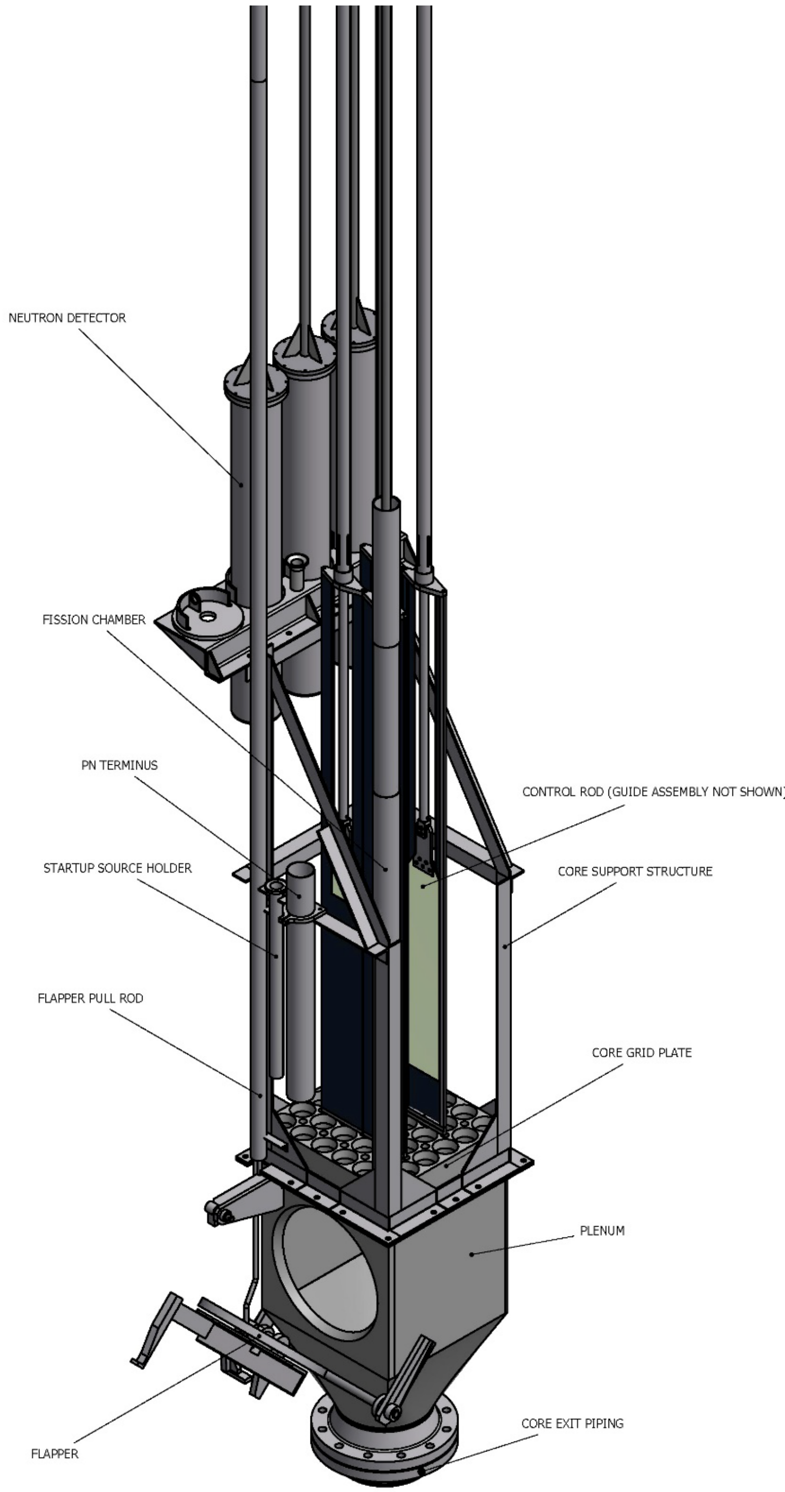

Figure 4-6 – Reactor core support structure detailing grid plate structure , plenum with flapper valve, control rod location, and neutron detector and fission chamber housings.

4.3. Reactor Pool

The reactor is an open pool type operating at atmospheric pressure. The upper elevation of the pool measures approximately [redacted], this steps down to [redacted]. The upper elevation is [redacted] deep while the lower elevation is [redacted] deep giving a total depth of [redacted], not including the fuel storage pits. The pool nominally holds 15,000 gallons of high resistivity light water. The pool is lined with reinforced [redacted] aluminum 6061 plate which is integral to the biological shield. Figure 4-7 details the plan view of the reactor pool.

[redacted]

Figure 4-7 – Reactor plan view detailing reactor core and peripheral experimental facilities.

There are multiple penetrations through the pool liner to accommodate primary coolant piping and experimental beamtubes. Two cylindrical fuel storage pits, each [redacted] in diameter by [redacted] deep, extend below the bottom of the pool liner. Each fuel pit can accommodate up to [redacted] fuel assemblies. The pool liner and associated primary piping are discussed in more detail in Section 5.

4.4. Biological Shield

The biological shield consists of three main sections. The lower section is high density barytes concrete which measures approximately [redacted] thick. It starts at the reactor bay floor elevation and continues upward to a height of [redacted] at which point the shielding transitions to standard concrete. This middle layer is nearly [redacted] in height and approximately [redacted] in thickness. The upper layer is standard concrete and is [redacted] in thickness and forms a parapet at the pool deck elevation. The biological shield is designed to reduce the external radiation fields to acceptable levels during full power operations. An analysis of radiation fields in the reactor facility arising from routine reactor operations are discussed in detail in Section 11, and in Section 13 for accident conditions. Figure 4-8 details the elevation view of the reactor pool and biological shield.

[redacted]

Figure 4-8 – Reactor elevation view detailing biological shield.

4.5. Nuclear Design

The NCSU PULSTAR reactor has been designed and built to operate for extended periods of time at power levels of up to 2 MW. The reactor fuel is similar in form to that of light water power reactor fuel, i.e., low enriched UO_2 pellets clad in zircaloy tubes. In the following sections, the analysis of the kinetic behavior of the reactor during steady state operation is presented. The analytical methods used to determine the nuclear parameters and operating characteristics of the reactor core are discussed in detail with further information provided in references as noted.

4.5.1. Normal Operating Conditions

The major components of the NCSU PULSTAR core have been described in the previous sections. This section contains a description of reactor operating characteristics, including the power distribution, excess reactivity, and effects of burnup and resulting reactivity changes. To characterize the current and future core configurations, a Monte Carlo MCNP model and analysis methodology was developed and validated for the PULSTAR reactor.[4-6] This validated model is to be used (in association with existing procedures) in the analysis of any future core configurations to ensure that all technical specification limits are met.

4.5.1.1. Core Configuration

Reflected Core No.11 was loaded in March 2019 and is currently in use. The core consists of 21 four percent enriched assemblies and 4 six percent enriched fuel assemblies arranged as detailed in Figure 4-9 below. The fuel was shuffled from the Core No.10 configuration to increase excess reactivity. This resulted in an increase in core excess reactivity of 528 pcm. The pin peaking factor for Core No.11 was calculated to be 2.48,[4-6] which is below the technical specification limit of 3.0. This calculation was verified by measurement using approved facility procedures.

4.5.1.2. ^{235}U Burn-up

The current reactivity burn-up rate has been calculated to be 0.2 pcm/MW·hr.[4-6] Given this rate of reactivity loss, and considering the burnup of ^{235}U, buildup of ^{239}Pu and accumulation of fission products, a 2000 pcm reactivity allowance was calculated to be sufficient to operate the reactor at 2 MW for up to 5000 hr.

Calculations indicate that throughout the core life, the buildup of ^{239}Pu has not had a significant effect on the effective delayed neutron fraction for the core.[4-6,4-9]

4.5.1.3. Xenon and Fission Product Accumulation

Reactivity effects of ^{135}Xe accumulate during routine reactor operations and its influence is evident over operating periods of several hours to days. The reactivity needed to compensate for this effect must be included in the excess reactivity requirement for the core, and should be known to reactor operations so that estimates of critical rod positions can be accurately predicted. At a minimum, xenon reactivity for two operation cycles of xenon are needed; operational xenon reactivity which occurs after 8 hours of 2 MW operations and equilibrium xenon reactivity which occurs after xenon saturates at approximately 48 hours of continuous 2 MW operations.

For Reflected Core No.9 at 1 MW, the reactivity loss due to the equilibrium buildup of ^{135}Xe was measured to be 900 pcm and 300 pcm for 8 hour operation. Using the xenon buildup equation, it has been calculated that the xenon reactivity will increase by approximately 25% for 2 MW operations resulting in an equilibrium xenon reactivity of approximately 1200 pcm and an 8 hour xenon reactivity of 400 pcm. Verification of the xenon reactivity for 2 MW operations will be conducted as part of the 2 MW Startup Plan.

A1 BERYLLIUM REFLECTOR	A2 BE4 BERYLLIUM REFLECTOR		A3 BE3 BERYLLIUM REFLECTOR	A4 BE2 BERYLLIUM REFLECTOR		A5 BE1 BERYLLIUM REFLECTOR	A6 BE5 BERYLLIUM REFLECTOR
B1 BERYLLIUM REFLECTOR	B2 7 1.37	SAFETY ROD #1	B3 2 1.43	B4 32 1.50	SAFETY ROD #2	B5 17 1.39	B6 21 0.80
C1 BERYLLIUM REFLECTOR	C2 35 1.61		C3 34 1.69	C4 27 1.73		C5 4 1.64	C6 45 0.65
D1 BERYLLIUM REFLECTOR	D2 28 1.69	REGULATING ROD #1	D3 30 1.78	D4 29 1.87	SHIM ROD	D5 31 1.74	D6 43 0.78
E1 BERYLLIUM REFLECTOR	E2 10 1.46		E3 16 1.60	E4 33 1.57		E5 8 1.52	E6 13 0.87
F1 FISSION CHAMBER	F2 26 1.12		F3 42 0.90	F4 44 1.01		F5 3 1.20	F6 20 0.68

Figure 4-9 – Reflected Core No.11 – March 2019 to present. Each fuel assembly is labeled according to its numeric index in the upper right corner, the core position in the upper left and the measured fuel assembly power peaking factor in the lower right. Fuel assemblies in locations C6, D6, F3 and F4 are 6% enriched, all others are 4% enriched.

4.5.1.4. Control Rod Worth

The control rods are described in detail in Section 4.2.2.

The reactivity worths of the control rods are measured to assure that the required shutdown margin is available and to provide a means for determining the reactivity worth of experiments inserted in the core. Control rod calibration curves are prepared individually for the control rods and for the gang rod configuration to meet the following license conditions:

- Prior to routine operation for any new core or control rod configuration.

- Annually but at intervals not to exceed fifteen (15) months for the current core in use.

The measurement of reactivity worth on an annual basis provides a correction for the slight variations expected due to burnup. This frequency of measurement has been found acceptable for this facility. A representative PULSTAR control rod calibration curve is given in Figure 4-10. Measured reactivity worths for the control rods for the historical cores as well as the current Reflected Core No.11 are listed in Table 4-5, along with the measured maximum reactivity insertion rates.[4-6,4-10] In the critical region, the maximum rate of reactivity insertion by the ganged control rods is limited to 200 pcm/sec.

Table 4-5 – Summary of control rod worth and reactivity parameters for historic cores. All values given have been obtained via measurement using facility procedures.

Core	Safety No.1 (pcm)	Safety No.2 (pcm)	Regulating Rod (pcm)	Gang (pcm)	Shutdown Margin (pcm)	Excess Reactivity (pcm)	Reactivity Insertion Rate (pcm/s)
Standard Core	4165	2631	2521	9316	-3104	2047	47
Reflected Core No.1	4084	3185	2379	9648	-3462	2102	52
Reflected Core No.2	Low Power Testing Only						
Reflected Core No.3	2655	2267	3982	8903	-2031	2890	48
Reflected Core No.4	2943	2502	3664	9110	-3858	1588	53
Reflected Core No.5	2348	2961	2787	8096	-3145	1991	55
Reflected Core No.6	2583	2808	2757	8168	-3459	1901	56
Reflected Core No.7	2246	2695	2634	7576	-2041	2839	56
Reflected Core No.8	2534	2901	2910	8345	-3630	1805	57
Reflected Core No.9	2850	2950	3000	8800	-3300	2500	54
Reflected Core No.10	2500	2800	2900	9100	-3514	1475	41
Reflected Core No.11	2550	2850	2900	9000	-3664	2436	59

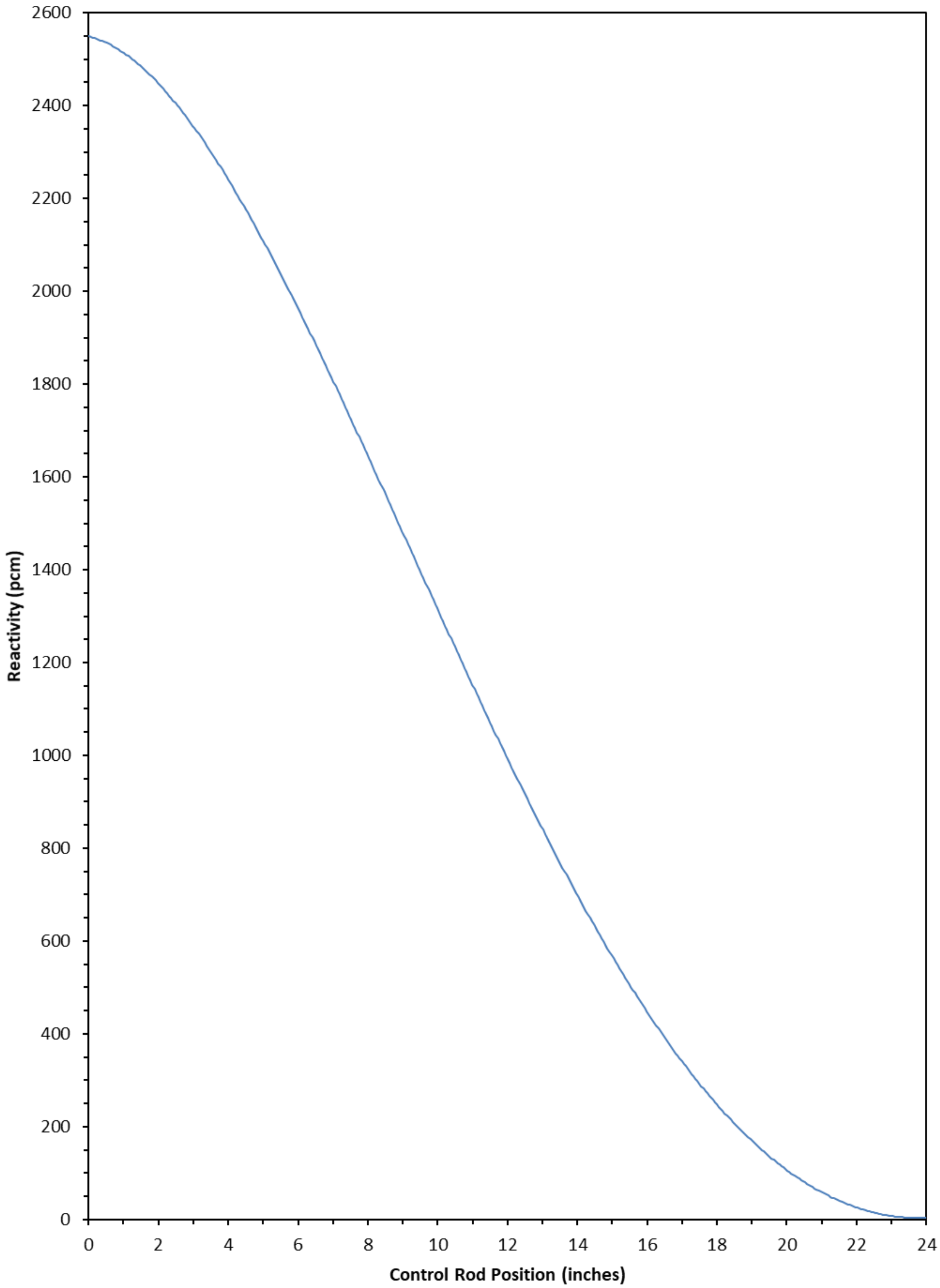

Figure 4-10 – Safety No.1 Integral Rod Worth Curve for Reflected Core No.11

4.5.1.5. Fuel Assembly Worths

Fuel assembly worths are calculated by the MCNP model and are defined as the change in excess reactivity of the reactor when that fuel assembly is loaded to the reactor core grid plate.

A fuel loading pattern can be generated, from the MCNP model, to ensure that the bases for a fuel handling accident as analyzed in Section 13 are never exceeded. This pattern will restrict fuel loading such that the individual fuel assembly worth, while being loading into the reactor core grid plate, shall not result in a k_{eff} of greater than 1.01626, which is equivalent to the step insertion of 1600 pcm during a fuel loading accident as analyzed in Section 13.

For Reflected Core No.11, fuel assembly 3 located in grid location F3 is calculated to have a worth that would cause an excess reactivity of 1228 pcm if it were inadvertently inserted in a fuel loading accident scenario. Reflected Core No.11 is well within the bounds of the fuel assembly worth limit given in the technical specifications.

4.5.1.6. Reflector Worths

Reflector assemblies are detailed in Section 4.2.3. Five graphite reflectors were first used in Reflected Core No.1 and resulted in a total reactivity gain of 920 pcm. Five additional graphite reflectors were added in Reflected Core No.3 for a total of ten graphite reflector assemblies. Due to bowing, five graphite reflectors were replaced with new graphite reflectors during the lifetime of Reflected Core No.3. Reflected Core No.4 exchanged five graphite reflectors for five beryllium reflectors and resulted in a reactivity gain of 740 pcm. The final five graphite reflectors were replaced with beryllium in Reflected Core No.8. The reactivity worth of each of the five beryllium reflectors ranged from 40 pcm to 240 pcm for an average of 160 pcm and total gain of 800 pcm.

4.5.1.7. Beamtube Worths

The PULSTAR beamtubes are shown in Figure 4-7 and detailed in Section 10. The reactivity worths resulting from flooding and voiding of the beamtubes are measured for active beamtubes with values listed in Table 4-6. These values compare well with the worths calculated using the MCNP model.[4-6]

Table 4-6 – Reactivity Worth of the PULSTAR Beamtubes

BT 1 (pcm)	BT 2 (pcm)	BT 3 (pcm)	BT 4 (pcm)	BT 5 (pcm)	BT 6 (pcm)
150	75	25	75	75	750

4.5.1.8. Future Core Configurations

There is not a predetermined set of planned core configurations for the PULSTAR Reactor. Rather, any configuration may be acceptable provided that all criteria specified in this Safety Analysis Report and the reactor technical specifications are satisfied. Any potential core configuration must be verified using the MCNP PULSTAR core model described in the License Amendment for the Use of 6% Enriched Fuel Appendix A – Examination of Mixed Enrichment Core Loading for the NCSU PULSTAR Reactor.[4-6] This analysis includes the modeling definitions and evaluation of excess reactivity, fuel assembly worth, and power peaking factors. Mixed enrichment core configurations with multiple fuel assemblies enriched to 6% in ^{235}U were considered.[4-6] Core configurations for Reflected Core No.10 and Reflected Core No.11 are shown in Figure 4-11 and Figure 4-12. The core positions of the 4% and 6% assemblies were selected such that all criteria including excess reactivity, fuel assembly worth, and core power peaking factors were within technical specification limits.

A1 BERYLLIUM REFLECTOR	A2 BE4 BERYLLIUM REFLECTOR		A3 BE3 BERYLLIUM REFLECTOR	A4 BE2 BERYLLIUM REFLECTOR		A5 BE1 BERYLLIUM REFLECTOR	A6 BE5 BERYLLIUM REFLECTOR
B1 BERYLLIUM REFLECTOR	B2 7 1.39	SAFETY ROD #1	B3 2 1.46	B4 32 1.51	SAFETY ROD #2	B5 17 1.33	B6 24 0.98
C1 BERYLLIUM REFLECTOR	C2 35 1.51		C3 34 1.62	C4 27 1.69		C5 4 1.49	C6 13 1.10
D1 BERYLLIUM REFLECTOR	D2 28 1.63	REGULATING ROD #1	D3 30 1.72	D4 29 1.82	SHIM ROD	D5 31 1.59	D6 43 1.43
E1 BERYLLIUM REFLECTOR	E2 10 1.37		E3 16 1.51	E4 33 1.61		E5 21 1.39	E6 20 1.02
F1 FISSION CHAMBER	F2 26 1.09		F3 3 1.18	F4 44 1.57		F5 23 1.14	F6 8 0.83

Figure 4-11 – Mixed Reflected Core No.10. Each fuel assembly is labeled according to its numeric index in the upper right corner, the core position in the upper left and the fuel assembly power peaking factor in the lower right. The six percent fuel assemblies are shown colored in green.

Figure 4-12 – Mixed Reflected Core No.11. Each fuel assembly is labeled according to its numeric index in the upper right corner, the core position in the upper left and the fuel assembly power peaking factor in the lower right. The six percent fuel assemblies are shown colored in green.

The excess reactivity, SDM, reactivity insertion rate, and core pin peaking factors are tabulated in Table 4-7 for the mixed enrichment Reflected Cores No.10 and No.11 as illustrated in Figure 4-11 and Figure 4-12 The power peaking factor, F_Q, is the maximum pin power peaking factor in the core. The license limits (as specified in the technical specifications) for excess reactivity, SDM, reactivity insertion rate, and power peaking factor are less than 4800 pcm, less than -400 pcm, less than 200 pcm/s, and less than 3.0 respectively. The predicted worth a fuel assembly in the configurations were determined to satisfy the license limit for reactivity insertion by a single fuel assembly.

Based on the table below,[4-6,4-9] Mixed Reflected Core No.10 and Mixed Reflected Core No.11 were considered acceptable for loading. The ρ_{excess}, SDM, $\rho_{insertion}$ and F_Q are well within the technical specifications limits. All future cores must satisfy all license conditions and limits to be considered acceptable.

Table 4-7 – Summary of MCNP Core Reactivity Parameters for Mixed Enrichment Cores

Parameter	Limit	Reflected Core No.10	Reflected Core No.11
ρ_{excess} (pcm)	4800	2661	2507
SDM (pcm)	-400	-2805	-2873
$\rho_{insertion}$(pcm/s)	200	66	65
F_Q	3.0	2.5	2.48

4.5.2. Reactor Core Physics Parameters

Core reactor physics characteristics for the current core and potential core configurations were determined using the MCNP Monte Carlo code model of the PULSTAR reactor core and analysis methodologies detailed in the License Amendment for the Use of 6% Enriched Fuel Appendix A – Examination of Mixed Enrichment Core Loading for the NCSU PULSTAR Reactor.[4-6,4-9] The parameters are then verified by measurement using approved facility procedures. The measured values are listed below in Table 4-8.

4.5.2.1. Reactivity Parameters for PULSTAR

The measured reactivity coefficients are listed in Table 4-8 for Reflected Core No.8 and Reflected Core No.9, and the mixed enrichment core Reflected Cores No.10 and No.11. Historically, a β_{eff} of 730 pcm has been used for 4% enriched cores. For mixed enrichment cores, the value of 733 pcm calculated by MCNP will be used in this Safety Analysis Report[4-6].

Since there is no indication of fuel temperature at the PULSTAR, the fuel temperature coefficient (α_F) cannot be directly measured. It however can be derived from the power coefficient and the fuel temperature coefficient measurements.

The power defect is the total reactivity loss from both the rise in fuel temperature and the rise in moderator temperature. It is measured by taking the change in critical rod position for a change in power level. The power defect, while slightly stronger (more negative) at lower fuel temperatures (power levels), can be assumed to be constant from the point of adding heat of 10kW up to full power of 2 MW without introducing significant error.

Power defect is given by:

$$\boldsymbol{\alpha_P = (\alpha_F \cdot \Delta T_F) + (\alpha_T \cdot \Delta T_{mod})}$$

where,

α_P is the reactivity from nominal hot zero power to nominal hot full power

α_F is the fuel temperature coefficient

α_T is the moderator temperature coefficient

ΔT_F is the average change in fuel temperature from nominal hot zero power to nominal hot full power

ΔT_{mod} is the average change in moderator temperature across the core

For steady-state cases, hot zero power refers to a primary coolant temperature at the nominal condition of 100 °F. In a power level increase from hot zero power at nominal conditions to 2 MW, the moderator temperature increases by 13.8°F across the core. Therefore, using a moderator temperature coefficient of -3.7 pcm/°F and an average increase in moderator temperature of 13.8°F/2 = 6.9°F would insert -25.5 pcm of reactivity. Using the calculated average fuel temperature of 434°F at 2 MW[4-14] yields a temperature increase of 334°F over hot zero power at nominal conditions. Applying the fuel temperature coefficient of -1.6 pcm/°F results in a negative temperature feedback of -534.4 pcm. Combining the fuel and moderator temperature coefficients results in a total power defect of -560 pcm.

Verification of the power defect for 2 MW will be performed as part of the 2 MW Startup Plan.

Table 4-8 – Summary of Measured Reactivity Coefficients for Representative Core Configurations

Core	Isothermal Temperature Coefficient α_T (pcm/°F)	Fuel Temperature Coefficient α_F (pcm/°F)	Void Coefficient α_V (pcm/cm^3)	1 MW Power Defect $\Delta\rho_P$ (pcm)	2 MW Power Defect $\Delta\rho_P$ (pcm)	β_{eff} (pcm) *calculated
Reflected Core No.8	Measurement no longer performed. Assumed to be -3.7	-1.6		-321	N/A	730
Reflected Core No.9		-1.5	-1.1	-313	N/A	730
Mixed Core No.10		-1.7	-0.9	-348	N/A	733
Mixed Core No.11		-1.6	-1.1	-332	-560 (estimated)	733

4.5.3. Operating Limits

4.5.3.1. Excess Reactivity and Shutdown Margin

The design criteria for shutdown margin is that the core excess reactivity shall never exceed a value which would permit the core to be made critical by withdrawal of the control rod with the maximum worth to the fully withdrawn position with the other control rods in the full down position.

To ensure that this design criteria is always met, the shutdown margin for any geometry shall be less than -400 pcm with the highest worth control rod completely withdrawn from the reactor core and the other control rods fully inserted into the core. To make sure that the -400 pcm shutdown margin limit is always satisfied, the excess reactivity for the PULSTAR core shall be limited to 4800 pcm. The operational reactivity requirements for the PULSTAR are shown in Table 4-9.

$$SDM = \rho_{Ganged\ Rod\ Worth} - \rho_{Highest\ Worth\ Rod} - \rho_{Excess}$$

The limit on excess reactivity is obtained by adding 3000 pcm allowance for experiments and 1890 pcm for operational requirements and rounding to 4800 pcm. The operational requirements account for 24-hour xenon poisoning, the increase in bulk pool temperature from the cold condition of 70°F to the hot condition of 105°F, and the power defect from low power to 2 MW.

From Table 4-5, Reflected Core No.11 has a total measured Gang worth of 9000 pcm, a highest rod worth of 2900 pcm, and a core excess reactivity of 2436 pcm which yields a SDM of 3664 pcm, which is well above the minimum SDM requirement of 400 pcm.

Table 4-9 – 2 MW Operational Reactivity Requirements

Requirement	Reactivity
Xenon – 24 hour	1200 pcm
Nominal Moderator Temperature Reactivity – Cold (70°F) to Hot Condition (105°F)	130 pcm
Power Defect	560 pcm
Experiments	3000 pcm
TOTAL	4890 pcm

To assure that the reactor will be operated within the bounds of established Safety Limits and Limiting Conditions for Operations, the following specifications apply to the reactivity parameters for the reactor core and the control rods.

Technical Specification 3.1

e. The worth of a fuel assembly while being loaded into the reactor grid plate shall not result in a k_{eff} of greater than 1.01626.

Technical Specification 3.2

The reactor shall not be operated unless the following conditions exist:

a. The minimum shutdown margin, with the highest worth control rod fully withdrawn and with experiments at their most reactive condition, relative to the cold critical condition, is greater than 400 pcm.

b. The excess reactivity is not greater than 4800 pcm.

c. The drop time of each control rod is not greater than 1.0 second.

d. The rate of reactivity insertion of the control rods is not greater than 200 pcm per second (critical region only).

Specification 3.1.e provides assurances that a fuel loading accident will not result in a safety limit being exceeded.

The shutdown margin required by Specification 3.2.a assures that the reactor can be shut down from any operating condition and will remain shut down after cool down and xenon decay, even if the highest worth control rod should be in the fully withdrawn position.

The upper limit on excess reactivity in Specification 3.2.b ensures that an adequate shutdown margin is maintained.

The rod drop time required by Specification 3.2.c is the time interval measured between the instant of a test signal input to the scram logic unit and the instant of the rod seated signal.

The maximum rate of reactivity insertion by the control rods which is allowed by Specification 3.2.d assures that the Safety Limits will not be exceeded during a startup accident (linear ramp reactivity insertion).

4.5.3.2. Experiment Reactivity

There are three classifications of experiments defined in the technical specifications that are based on experimental reactivity worth; movable, secured, and non-secured experiments. See Table 4-10 below. A movable experiment is one where it is intended that all or part of the experiment may be moved in or near the reactor core while the reactor is operating. A secured experiment is any experiment, experimental facility, or component of an experiment that is held in a stationary position relative to the reactor by mechanical means and may only be moved in or near the reactor core while the reactor is secured. A non-secured experiment is an experiment that is similar to a secured experiment in that it can only be moved into and out of position while the reactor is secured, but it does not have to be held stationary by mechanical means.

To assure that the reactor can be shutdown at all times and that the Safety Limits will not be exceeded, the following specifications apply to the reactivity worth of experiments.

Technical Specification 3.2

The reactor shall not be operated unless the following conditions exist:

e. The absolute reactivity worth of experiments or their rate of reactivity change shall not exceed the values indicated in Table 3-1.

f. The sum of the absolute values of the reactivity worth of all experiments shall not be greater than 3000 pcm.

Table 4-10 – Technical Specifications Table 3-1 – Reactivity Limits for Experiments

Experiment Type	Limit
Movable	300 pcm or 200 pcm/sec, whichever is more limiting
Non-secured	1000 pcm
Secured	1600 pcm

Specification 3.2.e is intended to prevent large inadvertent reactivity changes during reactor operation caused by the insertion or removal of an experiment. It further provides assurance that the failure of a single experiment will not result in a reactivity insertion which could cause safety limits to be exceeded. Analyses of inadvertent reactivity insertion of these magnitudes will not result in consequences greater than those analyzed in Section 13.

The specification 3.2.f limits the total reactivity associated with experiments to ensure that an adequate shutdown margin is maintained.

4.5.3.3. Reactivity Transient Analysis

The continuous withdrawal of the control rods would create a ramp reactivity addition. The failure of an in-core experiment could create either a ramp or a step reactivity insertion depending on the mode of failure. Bounding ramp and step reactivity additions are analyzed in Section 13.2.2 of this report and in neither case is the reactor damaged or fuel clad integrity lost.

4.5.3.4. Redundancy of Reactor Shutdown Mechanism

Redundancy for the NCSU PULSTAR shutdown mechanism is provided by four identical control rods that can be operated individually or as a bank. To satisfy the one stuck rod criterion, any one of the four control rods can be fully withdrawn and the Technical Specification 3.2.a for shutdown margin will be satisfied. Individual control rod worths and shutdown margin are calculated using the MCNP model[4-6] and then verified by measurements using facility procedures for all core configurations.

Electricity is required to energize the electromagnets that support the control rods, thus, on loss of electricity, the PULSTAR automatically shuts down.

4.5.3.5. Limiting Core Configuration

There is no set of pre-planned core configurations for the PULSTAR Reactor. Rather, any configuration is acceptable provided that all criteria specified in this SAR and the Technical Specifications are satisfied. To assure that the reactor will be operated within the bounds of established Safety Limits and Limiting Conditions for Operations, the following Specifications apply to the reactor core configuration during forced convection or natural convection flow operations.

Technical Specification 3.1

The reactor shall not be operated unless the following conditions exist:

a. A maximum of twenty-five fuel assemblies.

b. Any number reflector assemblies of either graphite or beryllium or a combination of these located on the core periphery.

c. Unoccupied grid plate penetrations plugged.

d. A minimum of four control rods guides are in place with operable control rods.

f. The total pin power peaking factor in any fuel assembly shall not exceed 3.0.

Specifications 3.1.a through 3.1.d require that the core be configured such that there is no bypass cooling flow around the fuel through the grid plate.

Specifications 3.1.d requires control rods are operable to ensure that shutdown margin requirements are satisfied.

Specification 3.1.f provides assurances that fuel integrity is maintained as discussed in the Safety Analysis Report.

4.6. Thermal-Hydraulic Design

4.6.1. Design Criteria

The steady-state heat transfer design of the NCSU PULSTAR reactor is based on five criteria:

No Bulk Boiling Criterion	Under forced convection cooling with downward flow, no coolant bulk boiling is allowed in any channel.
Flow Instability Criterion	There should be no coolant flow instability in any fuel channel that could lead to a significant decrease in fuel cooling.[4-11]
DNBR Criterion	The ratio of the calculated heat flux at the point of departure from nucleate boiling (DNB) to the maximum steady-state heat flux is greater than 2.0.[4-11]
Fuel Temperature Criterion	The maximum temperature of the fuel is less than 4352 °F.[4-11]
Cladding Temperature Criterion	The maximum temperature of the cladding is less than 2200 °F.[4-11][a]

The criterion for selecting a safety limit is to ensure the integrity of the fuel cladding. The interrelated variables associated with the core thermal and hydraulic performance with forced convection flow are:

P	Reactor thermal power
W	Reactor primary coolant flow rate
H	Height of water above the top of the core
T	Reactor primary coolant inlet temperature

When all values are jointly maintained with the limits determined by the safety analysis, fuel cladding integrity will not be lost. The safety limits preclude flow instabilities in the hottest channel and ensure that minimum steady-state DNB ratio is at least 2.0.

The core thermal hydraulic analyses are the framework for determining the core Safety Limits (SL) and Limiting Safety System Settings (LSSS) as presented in the Technical Specifications. The objective of the safety analysis calculations performed for the NCSU PULSTAR reactor is to determine the limits of

[a] Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

operation beyond which these design criteria may be violated. Reactor operation is then restricted to within levels established from these limits, including additional margin for uncertainty.

4.6.2. Safety Analysis

4.6.2.1. RELAP Model

The thermal hydraulic analysis of the PULSTAR reactor for steady state and transient conditions was performed using the RELAP5/MOD3.3 code of the US Nuclear Regulatory Commission (NRC). Specifically, the RELAP5 version used for the PULSTAR analysis is US-NRC RELAP5/MOD3.3 Patch04.[4-12] The QA installation verification was carried out on a Windows platform using two cases: "Edward Pipe" and "Zion 1 SBLOCA". The verification runs showed that the results coincide with those of the software supplier. Therefore, it is confirmed that RELAP5 is correctly installed and configured. The report Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor[4-7] describes in detail the RELAP5 model of the PULSTAR systems, steady-state initializations, and results of the accident analyses.

The RELAP5 model of the PULSTAR facility is shown in Figure 4-13 through Figure 4-15. The model is based on facility drawings and on data obtained by onsite walk downs. The model simulates the transport of heat and coolant in the primary system. The pool and the primary coolant loop are represented by a series of hydrodynamic volumes. Fuel assemblies in the core region are represented by heat structures. Fission and decay power are calculated by importing MCNP model data[4-6] into RELAP5. Schematic diagrams showing the main components of the PULSTAR primary system are shown in Figure 4-13 through Figure 4-15 representing the reactor pool, core configuration, and the coolant loop configuration. The discussion of the PULSTAR model will be grouped into four sub-sections: the reactor pool, the reactor core, the primary coolant loop, and the secondary cooling loop. A component number, as defined in the RELAP5 input deck, is used to identify each hydrodynamic volume modeled.

Reactor Pool

The reactor pool is divided into a number of interconnected hydrodynamic volumes. During normal operation, primary coolant heated from the reactor flows downward through the core into the reactor coolant loop and, then, returns to the bottom of the reactor pool after being cooled in the reactor coolant loop. The top of the reactor pool is open to atmosphere, which is modeled as a time-dependent volume of RELAP5. The flapper valve at the core outlet plenum is modeled to open by differential pressure between the pool and the plenum in case that the main coolant flow stops.

Reactor Core

Figure 4-15 shows the fuel channel modeling and nodalization. The core, consisting of 5×5 fuel boxes each containing 5×5 fuel rods, is modeled with four lumped fuel channels. A total of 25 fuel channels are lumped into three averaged fuel channels each representing 5, 9 and 10 fuel channels, and a hot channel that represents the highest power fuel channel. Axial nodalization of flow channels consists of 14 volumes, with 10 volumes in the active core and additional volumes at the top and bottom of the assemblies. Fuel channels are modeled open to the reactor pool via the top assembly inlet and the bypass flow holes at the top of the active core. Thus, core inlet flow is formed in two flow paths, one from the core top and one from the bypass flow holes. Downward core flow is collected in the core outlet plenum and then flows to the primary coolant loop. Flow through the reactor core is calculated to be uniformly distributed and is based on the number of fuel assemblies that have been lumped into each core channel.

Fuel rods in a lumped fuel channel are modeled with heat structures. A single lumped fuel pin is used to represent fuel rods in a lumped fuel channel and an additional hottest pin is modeled in a hot channel for assessment of safety margin. Fuel material properties are obtained from IAEA-TECDOC-1496.[4-13] One-dimensional radial or transverse conduction in a heat structure is modeled and additional axial heat conduction is modeled when the heat structures get uncovered. Convection heat transfer is modeled at surfaces of fuel pins and fuel boxes. An AECL critical heat flux look-up table is used to estimate the fuel DNBR transients. And, when the surface of the heat structure is uncovered, radiation heat transfer occurs from fuel pin to fuel box and then to the heat sink represented by the pool wall.

Each group of fuel assemblies is represented as an idealized core channel. It is assumed in the PULSTAR model that the core channel flow paths are connected in parallel. Power distribution to each channel is obtained by lumping power distributions from the NCSU PULSTAR core physics analysis.[4-6] Conservatively, a core peaking factor of 3.0 is used.

The hot channel is selected from the NCSU PULSTAR core analysis. Active fuel is divided into 10 nodes and the hottest pin in the hot channel is conservatively assumed to have a peaking factor equal to 3.0.

Primary Coolant Loop

The primary coolant flow path in the PULSTAR is a single loop with components located in series. Figure 4-14 depicts the layout of the primary coolant loop from the reactor outlet to the reactor pool inlet.

The primary coolant loop is modeled with primary components including the ^{16}N decay tank, coolant pump, heat exchanger and pipes connecting the components. Actual PULSTAR piping layouts and elevations are represented in the RELAP model with the exception of the heat exchanger which has been model as a tube and shell. This simplification does not have a significant effect on the RELAP5 analysis.

Secondary Coolant Loop

The secondary cooling loop is modeled simply as a once-through circuit. At one end a source supplies the cooling water to the heat exchanger, while the other end flows to a sink (cooling tower).

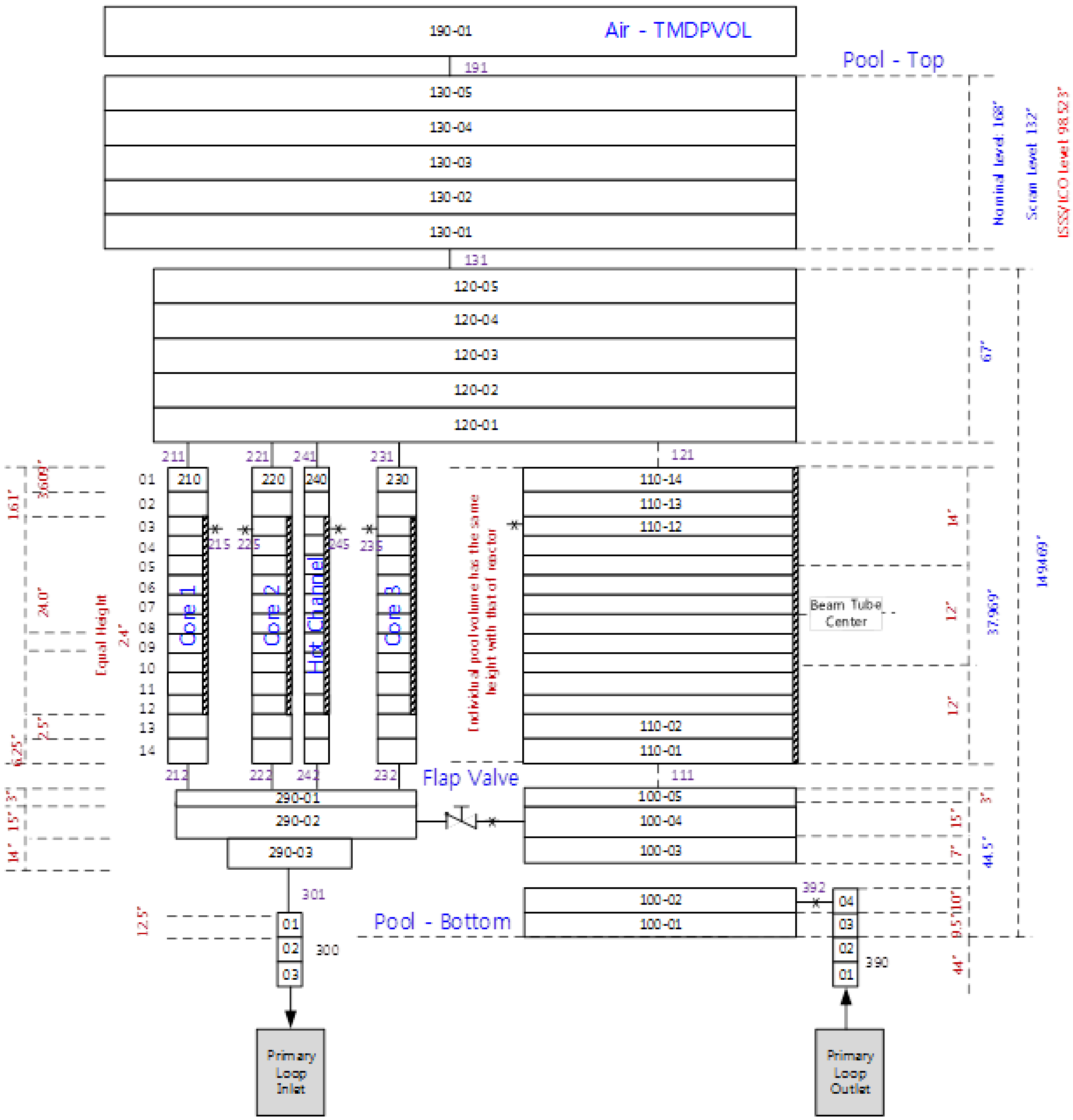

Figure 4-13 – RELAP5 Model Nodalization of the PULSTAR Reactor Pool and Core

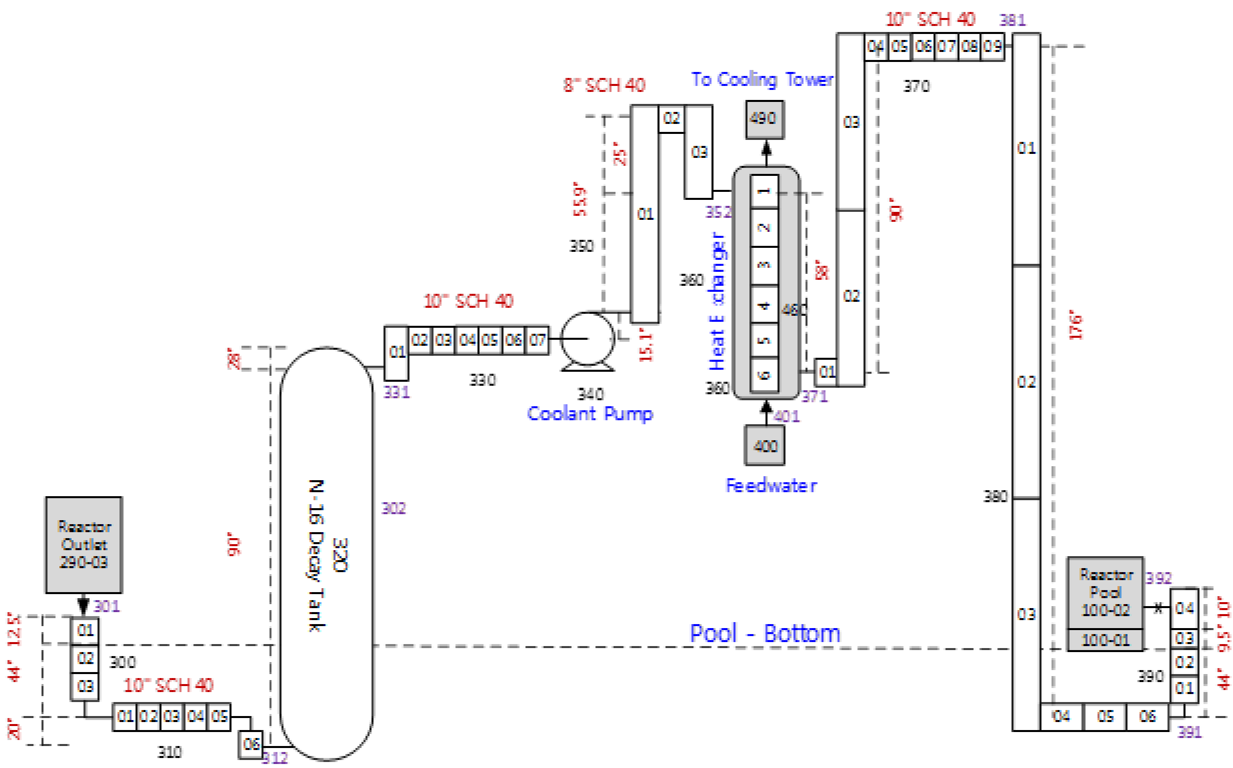

Figure 4-14 – RELAP5 Model Nodalization of the PULSTAR Coolant System

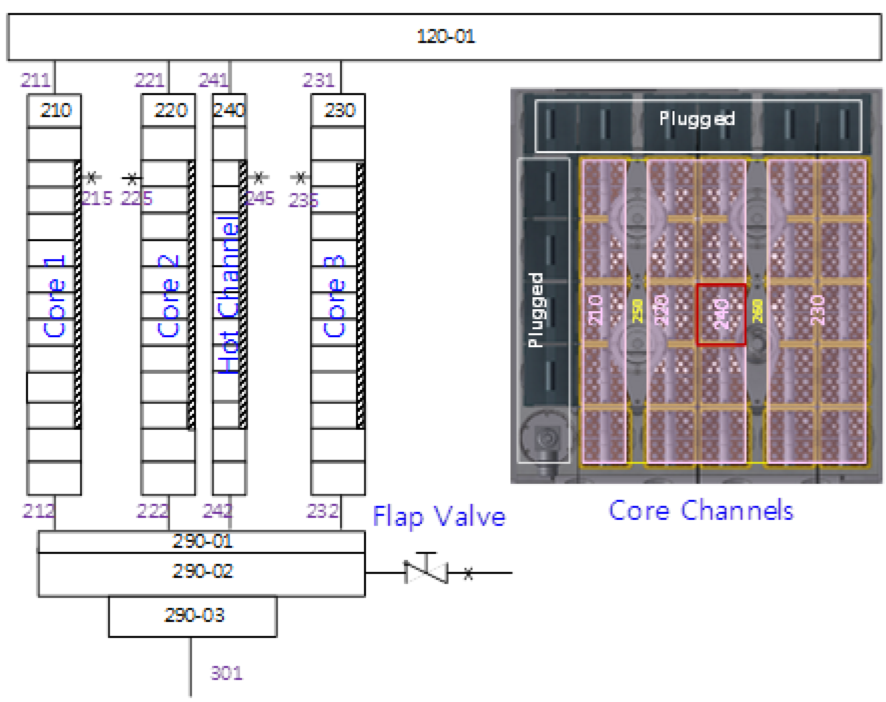

Figure 4-15 – PULSTAR Fuel Channel Modeling and Nodalization

4.6.3. Steady-State Analysis

The NCSU PULSTAR can operate in two flow modes, namely, forced and natural convection. The important parameters with respect to the thermal hydraulics of the core are the power level, flow rate, coolant channel inlet temperature (i.e., bulk pool temperature) and coolant pressure as determined by the height of water above the top of the core. A complete thermal hydraulic analysis of the core requires that the operating parameters be examined to determine their worst or most adverse operating limit since this will bear upon the design limits of the core.

4.6.3.1. Forced Convection Mode

Table 4-11 provides a summary of the steady-state initialization under limiting conditions which are the limiting safety system setpoints and are compared with nominal operating conditions. The results of the steady-state analysis are illustrated in Figure 4-16 which shows the fuel temperature profiles for a limiting condition of 2.0 MW. The temperatures remain well below the NUREG 1537 limits of 4352 °F for UO_2 fuel and 2200 °F for zircaloy cladding.[4-11][a] Thermal margins for the limiting conditions are summarized in Table 4-12. The minimum DNBR remains well above the design criterion limit of 2.0. There is sufficient subcooled temperature margin in the hot fuel channel exit to preclude bulk boiling and fuel centerline temperatures are well below the melting temperature as are the clad temperatures. These calculations assume a hot pin power peaking factor of 3.0.

[a] Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

Table 4-11 – Steady-state thermal-hydraulic initialization parameters under limiting conditions as compared with reference nominal operating conditions.

Parameters	Nominal	Limiting Conditions
Reactor Power (MW)	1.8	2.0
Peaking Factor	2.54	3.0
Primary Flow (gpm)	1000	900
Core Inlet Temperature (°F)	105	117
Core Temperature Rise (°F)	12.4	15.3
Height of Water Above Top of Core (inches)	240	204
Secondary Flow (gpm)	1000	1000
Secondary Inlet Temperature (°F)	92	92
Secondary Pressure (psig)	18.5	18.5

Table 4-12 – Steady-state thermal margins for limiting conditions

Parameters	Limiting Conditions
Peak Fuel Centerline Temperature	1110.8 °F
Peak Cladding Temperature	226.9 °F
Minimum DNBR	> 19

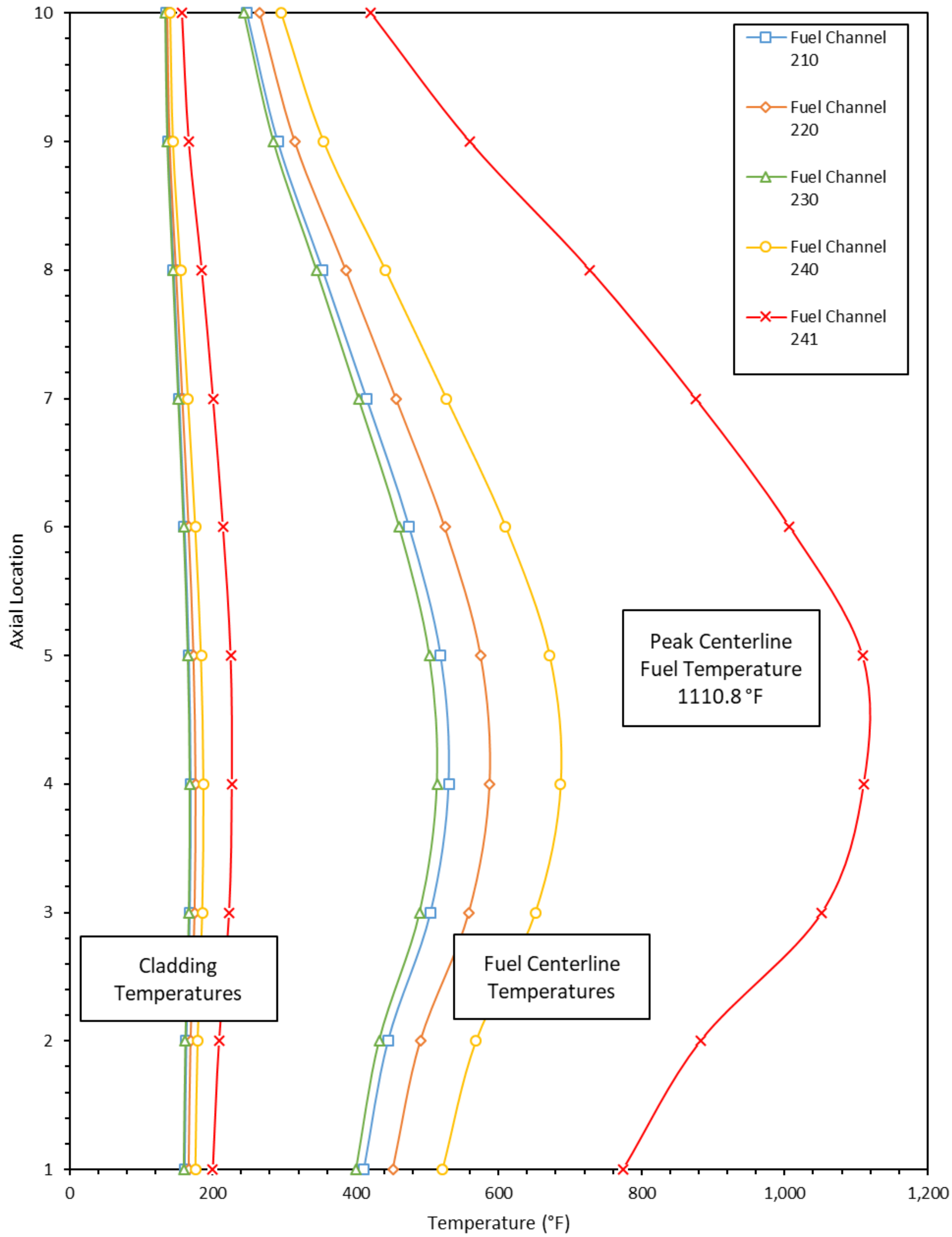

Figure 4-16– Steady-State Axial Fuel Temperature Profiles for limiting conditions given in Table 4-11. Maximum peak centerline temperature in the hot pin with a pin peaking factor of 3.0.

4.6.3.2. Power versus Flow Analysis

The power-vs.-flow analysis establishes the steady-state Safety Limit (SL) as a function of reactor power and primary coolant flow rate. Using the RELAP5 model,[4-7] the initial conditions of pressure (height of water above the core) and inlet temperature were established at the safety limits and then at various power levels the flow rate was reduced until the major safety parameters listed below were challenged.

No Bulk Boiling Criterion	Under forced convection cooling with downward flow, no coolant bulk boiling is allowed in any channel.
Flow Instability Criterion	There should be no coolant flow instability in any fuel channel that could lead to a significant decrease in fuel cooling.[4-11]
DNBR Criterion	The ratio of the calculated heat flux at the point of departure from nucleate boiling (DNB) to the maximum steady-state heat flux is greater than 2.0.[4-11]
Fuel Temperature Criterion	The maximum temperature of the fuel is less than 4352 °F.[4-11]
Cladding Temperature Criterion	The maximum temperature of the cladding is less than 2200 °F.[4-11][a]

Analysis Procedure:

1. Pool Level set to the Safety Limit of 168 inches from the top of the core.
2. Pool temperature set to the Safety Limit 120 °F.
3. Boundary flow initially set to 50% (500 gpm) of nominal full core flow.
4. Step-wise reduction in flow rate of 5% from nominal to zero flow for a given power level.
5. Repeat above Steps increasing Power Levels (0.5, 1, 2, 3...) until major safety parameters listed above are challenged.

Table 4-13 lists the flow rates that challenge the major safety parameters in the hot channel. DNBR is challenged by dynamic flow instability rather than static heat transfer and occurs at 5.5 MW at 50% of full 1000 gpm nominal flow which ultimately results in cladding failure.

The power levels required to reach the major safety parameters in the hot channel are depicted in Figure 4-17.

Safety limits for nuclear reactors are limits upon important process variables (e.g., power level and coolant flow) that are found to be necessary to reasonably protect the integrity of certain of the

[a] Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

physical barriers that guard against the uncontrolled release of radioactivity.

The data presented in Table 4-13 and shown in Figure 4-17 is used to establish the power-vs.-flow safety limit curve in the Technical Specifications as presented in Figure 4-20.

Table 4-13 – Thermal Margins for Forced Convection at Various Power Levels

Core Power Level (MW)	Onset of Flow Instability (gpm)	Core Bulk Boiling (gpm)	DNBR < 2.0 (gpm)	Fuel Failure (gpm)
0.5	75	75	Does not Occur	Does not Occur
1.0	125	125	Does not Occur	Does not Occur
2.0	200	200	200	50
3.0	275	275	275	275
4.0	400	400	400	400
4.5	450	450	450	450
5.0	500	500	500	500

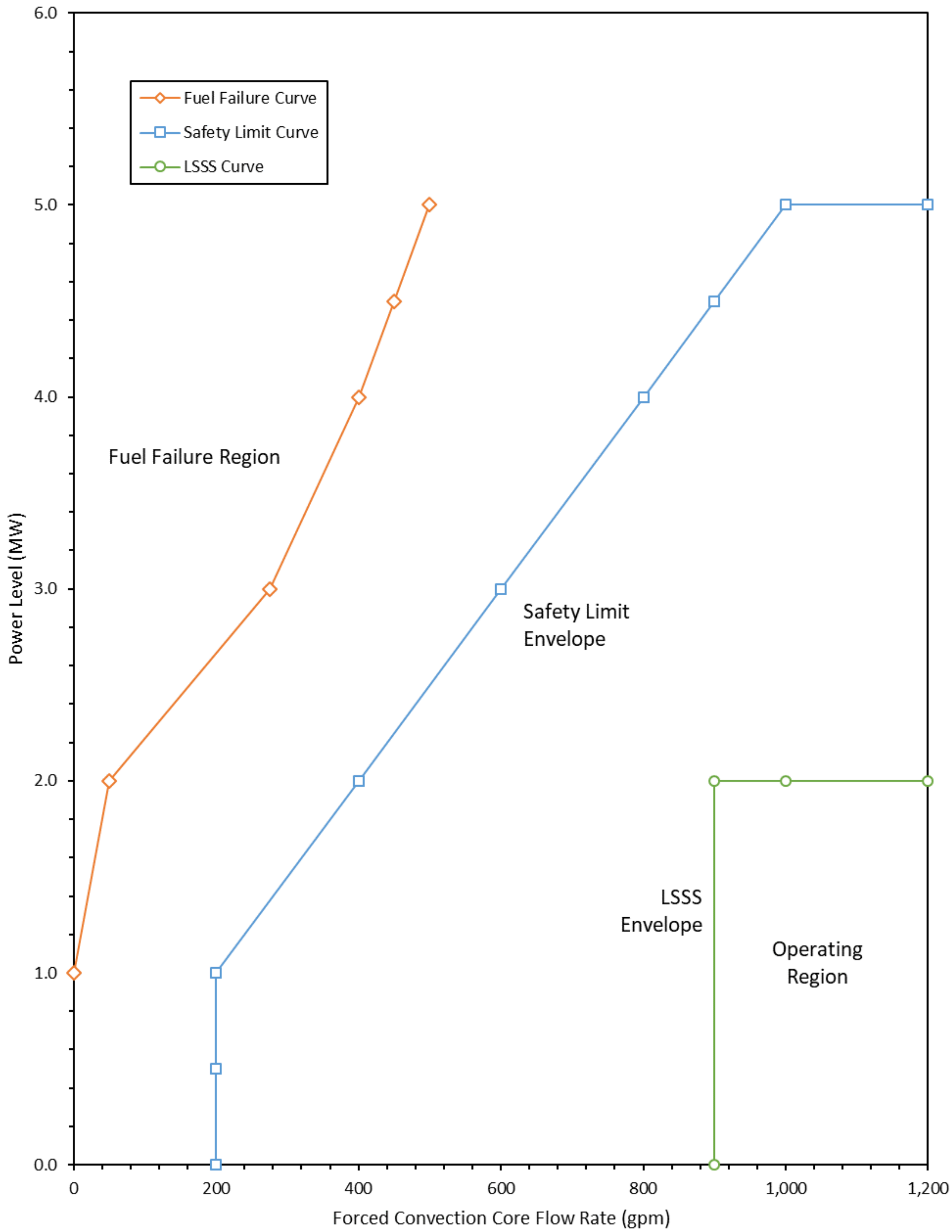

Figure 4-17– Safety Limit Power vs Flow Curve. The operating region is defined as the area bounded by the limiting safety system settings (LSSS).

4.6.3.3. Natural Convection Mode

Using the RELAP5 model[4-7] the parameters of pressure (height of water above the core) and inlet temperature were established at the safety limit and then the power level was increased until the major safety parameters were challenged.

Analysis Procedure

1. Pool Level set to the Safety Limit of 168 inches from the top of the core.
2. Pool temperature set to the Safety Limit 120 °F.
3. Boundary flow initially set to zero.
4. Flapper valve is set open
5. Repeat above steps increasing power levels and evaluate if major safety parameters are challenged. Refer to Table 4-14.

The criterion for establishing a Safety Limit with natural convection flow is established as the fuel clad temperature. The analysis of natural convection flow given in the report, Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor[4-7] and presented in Figure 4-18 shows fuel and cladding temperatures for natural convection flow at various power levels. The power level of 1.0 MW has been chosen as the safety limit for natural convection flow. Figure 4-19 shows that at 1.0 MW, departure from nucleate boiling does not occur in the hot channel and maximum fuel centerline and cladding temperatures are well below the levels at which damage would occur.

The Limiting Safety System Setting for natural convection assures that an adequate safety margin exists between the LSSS and the SL for natural convection. The safety margin on reactor thermal power was chosen with the additional consideration related to bulk boiling at the outlet of the hot channel. This criterion is not related to fuel clad damage which was the criterion used in establishing the Safety Limits but to minimize ^{16}N dose at the pool surface which might be aided by steam bubble rise during up-flow in natural convection. Analysis of coolant bulk boiling given in the report, Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor[4-7] shows that an LSSS of 250 kW will satisfy this additional criterion of no bulk boiling in any channel.

Table 4-14 - Thermal Margins for Natural Convection at Various Power Levels

Core Power Level MWt	Onset of Local Boiling and Flow Instability (sec)	Onset of Core Bulk Boiling (sec)	Hot Pin Fuel Centerline Temperature (°F)	Hot Pin Cladding Temperature (°F)	Minimum DNBR
0.1	Does not Occur	Does not Occur	243.5	203.0	Not Calculated
0.25	~4500	Does not Occur	343.9	242.9	22.8
0.5	~1800	~6900	456.9	248.0	9.8
1.0	Immediately	~3000	734.1	342.7	1.1
1.4	Immediately	~2300	983.1	424.2	0.7
1.6	Immediately	~2000	3553.8	2603.3	0.1

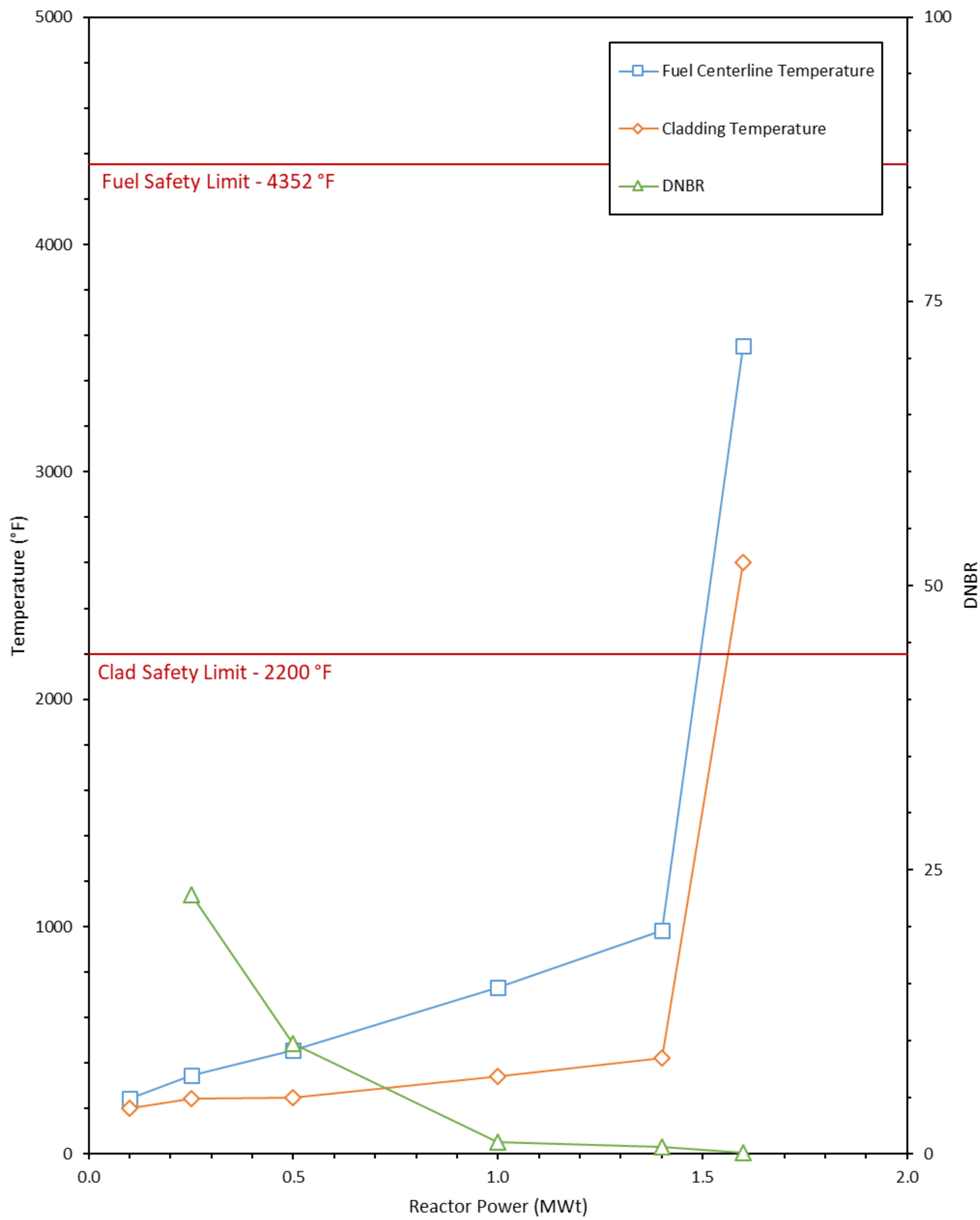

Figure 4-18 – Fuel and cladding temperatures in the hot channel for natural convection at various power levels.

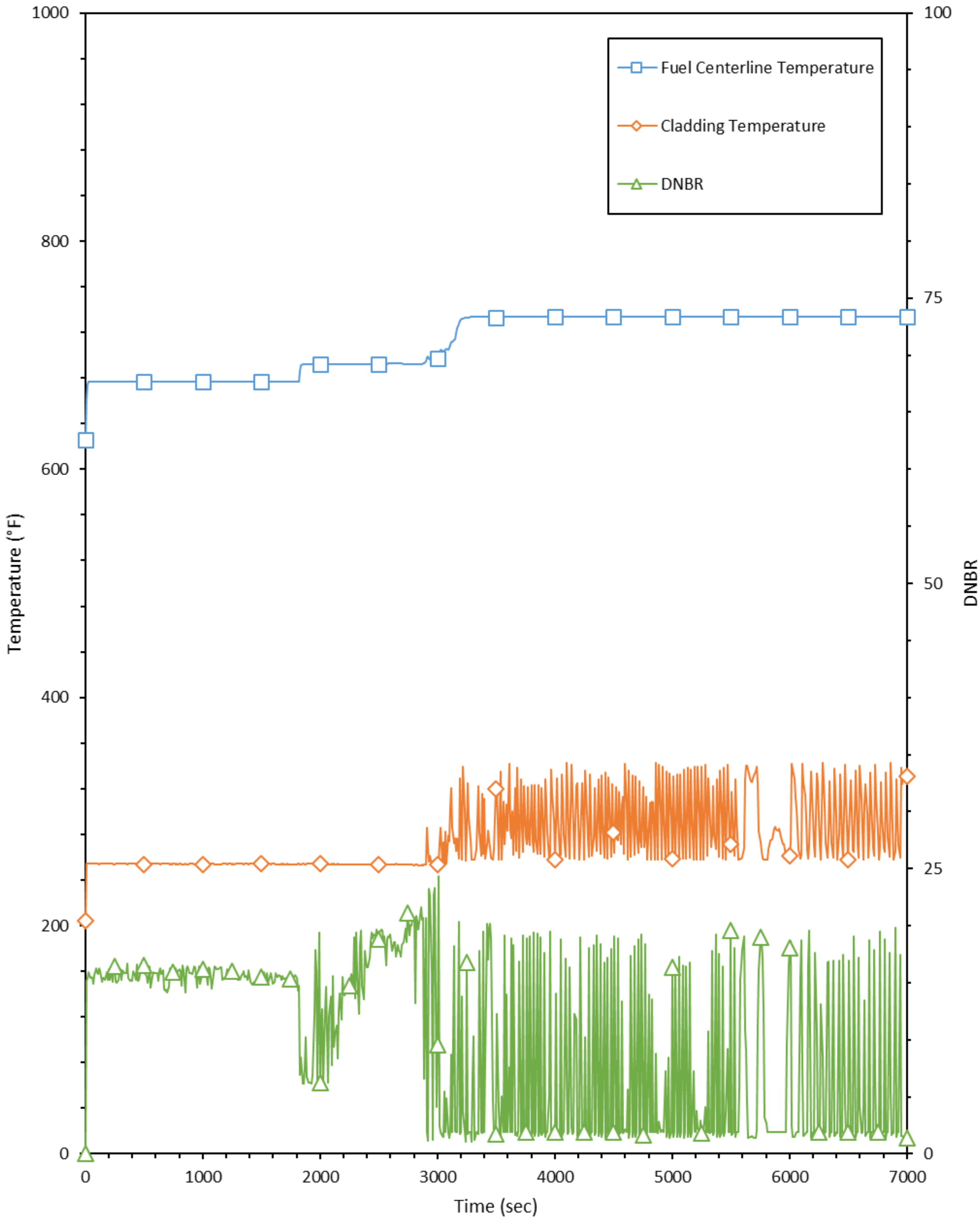

Figure 4-19 – Fuel and cladding temperatures and DNBR in the hot channel for natural convection at 1.0 MW

4.6.4. Safety Limits

Safety Limits are established for two modes of reactor operation, forced convection flow and natural convection flow. The analyses for the bases for these limits are given in Section 4.6.

4.6.4.1. Forced Convection Flow

The objective for establishing Safety Limits for forced convection flow is to ensure the integrity of the fuel cladding which is based on the five criteria listed in Section 4.6. The safety limits ensure that all the criteria are satisfied and fuel cladding integrity will be maintained. The analysis for forced convection flow given in the report, Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor[4-7] shows that if the reactor is operated within the bounds of the safety limits given below, the maximum fuel clad temperature is well below the temperature at which fuel clad damage could occur.

To assure that the integrity of the fuel cladding is maintained, Technical Specification 2.1.1 applies to the interrelated variables associated with the core thermal and hydraulic performance during forced convection flow. These interrelated variables are:

P Reactor Thermal Power

W Reactor Coolant Flow Rate

H Height of Water Above the Top of the Core

T_{inlet} Reactor Coolant Inlet Temperature

<u>Technical Specification 2.1.1</u>

Under the condition of forced convection flow, the Safety Limit shall be as follows:

a. The combination of true values of reactor thermal power (P) and reactor coolant flow rate (W) shall not exceed the limits shown in Figure 4-20 (TS Figure 2-1) under any operating conditions. The limits are considered exceeded if the point defined by the true values of P and W is at any time outside the operating envelope shown in Figure 4-20.

b. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

c. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

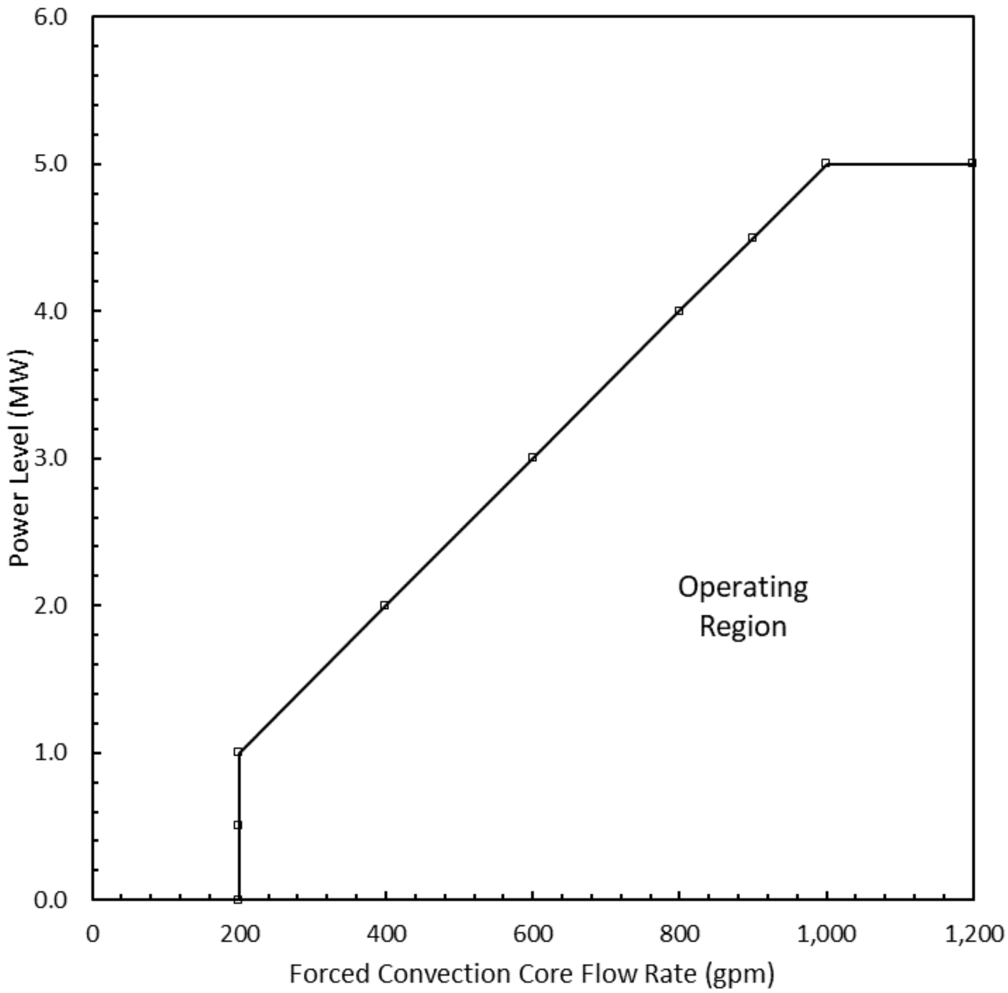

Figure 4-20 – Technical Specifications Figure 2-1 – Power -vs- Flow Safety Limit Curve

4.6.4.2. Natural Convection Flow

The criterion for establishing a Safety Limit with natural convection flow is the maximum fuel clad temperature. The analysis of natural convection flow given in the report, Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor[4-7] shows that at 1.0 MW the maximum fuel clad temperature is well below the temperature at which fuel clad damage could occur.

This specification applies to the interrelated variables associated with the core thermal and hydraulic performance with natural convection flow. These interrelated variables are:

P	Reactor Thermal Power
H	Height of Water Above the Top of the Core
T_{inlet}	Reactor Coolant Inlet Temperature

Technical Specification 2.1.2

Under the condition of natural convection flow, the Safety Limit shall be as follows:

a. The true value of reactor thermal power (P) shall not exceed 1.0 MW.

b. The true value of pool water level (H) shall not be less than 14 feet above the top of the core.

c. The true value of reactor coolant inlet temperature (T_{inlet}) shall not be greater than 120°F.

4.6.5. Limiting Safety System Settings

4.6.5.1. Forced Convection Flow

The Limiting Safety System Settings (LSSS) that are given in Technical Specification 2.2.1 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent Safety Limits from being exceeded during the most limiting anticipated transient.

The margin provided between the Safety Limits and the Limiting Safety Limit Settings is adequate to assure that the Safety Limits would not be exceeded given the uncertainties associated with measuring physical reactor parameters.

The analyses presented for all credible accident scenarios in Section 13 indicate that if, at the initiation of a transient, the interrelated variables were at their LSSS as specified in Technical Specification 2.2.1, the bases for the Safety Limits specified in 2.1.1 would not be exceeded.

This specification applies to the setpoints for the safety channels monitoring reactor thermal power (P), coolant flow rate (W), height of water above the top of the core (H), and pool water temperature (T).

Technical Specification 2.2.1

Under the condition of forced convection flow, the Limiting Safety System Settings shall be as follows:

P 2.0 MW (max.)

W 900 gpm (min.)

H 17 feet (min.)

T 117°F

4.6.5.2. Natural Convection Flow

The Limiting Safety System Settings that are given in Technical Specification 2.2.2 represent values of the interrelated variables which, if exceeded, shall result in automatic protective actions that will prevent Safety Limits from being exceeded. The specifications given above assure that an adequate safety margin exists between the LSSS and the SL for natural convection. The safety margin on reactor thermal power was chosen with the additional consideration related to bulk boiling at the outlet of the hot channel. This criterion is not related to fuel clad damage (for these relatively low power levels) which was the criterion used in establishing the Safety Limits (see Specification 2.1.2). It is desirable to minimize ^{16}N dose at the pool surface which might be aided by steam bubble rise during up-flow in natural convection. Analysis of coolant bulk boiling given in the report, Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor[4-7] indicates that the large safety margin on reactor thermal power given in Specification 2.2.2 above will satisfy this additional criterion of no bulk boiling in any channel.

Technical Specification 2.2.2

Under the condition of natural convection flow, the Limiting Safety System Settings shall be as follows:

P 250 kWt (max.)

H 17 feet (min.)

T 117°F

4.7. Summary and Conclusions

This section contains a detailed description of the PULSTAR reactor components along with analyses of steady state operating conditions. Current and potential future core configurations are described. An MCNP[4-6] core model has been developed and utilized to calculate reactivity parameters for the existing core and potential mixed enrichment cores. A thermal hydraulics RELAP5 model[4-7] has been developed and utilized to calculate core thermal parameters for the existing core and potential mixed enrichment cores. The results obtained from these models have been utilized to establish Safety Limits (SL), Limiting Safety System Settings (LSSS), and Limiting Conditions for Operations (LCO) for the PULSTAR Reactor. The results of the analyses detailed in this section demonstrate that operating the reactor at steady state conditions within the boundaries set by the SL, LSSS, and LCOs will assure that the design criteria for fuel and cladding temperatures will not be exceeded.

Therefore, operation of the PULSTAR reactor will present no undue hazard to any member of the general public or to personnel.

4.8. References

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5. REACTOR COOLANT SYSTEMS

5.1. Summary Description

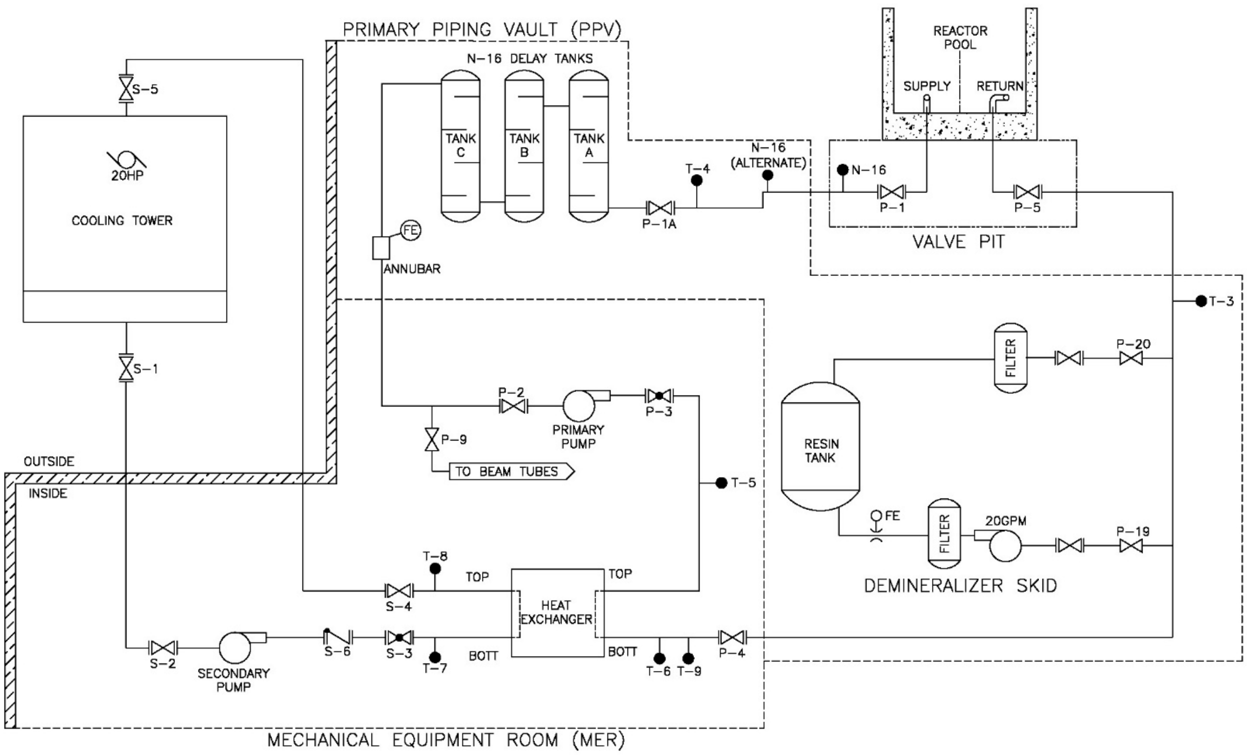

Figure 5-1 – Reactor Primary and Secondary Coolant Systems Overview

The reactor coolant system, utilizing demineralized water, is designed to remove up to two megawatts (2 MW) of heat from the PULSTAR reactor operating in the steady-state mode with forced convection cooling and have sufficient capacity to operate at power levels up to 100 kW with natural convection cooling. Equally important is the ability to cool the reactor during reverse flow upon failure of the primary coolant pump. The analyses in support of this design are found in Section 4 and Section 13 with pertinent features of the system design found in the following description. The primary and secondary coolant systems and their supporting components are shown schematically on Figure 5-1 through Figure 5-6.

The primary coolant system is an open pool loop operating at atmospheric pressure. The heat from fission is absorbed by the coolant as it passes down through the reactor core and is transferred to the secondary coolant system through a plate type heat exchanger. The heat is then removed from the secondary coolant system by a standard counter-flow evaporative cooling tower.

The coolant system is designed to remove up to two megawatts of heat from the PULSTAR reactor and maintain a core inlet temperature of 105°F with a 1000 gpm flow rate on the primary side. The secondary coolant system is designed to keep the primary system pool inlet temperature below 105°F.

The purification system is designed to maintain high resistivity and purity of the primary coolant system by continuously filtering and demineralizing a 20 gpm bypass flow of the coolant.

The reactor coolant systems have no requirement for auxiliary power for the coolant pumps or special coolant sources. As discussed in Section 4.6, the reactor is capable of natural convection cooling to power levels of at least 1.0 MW without any damage.[5-1] Analysis of transient cooling requirements is covered in Section 13.

The following tests and inspections are performed as required by the technical specifications[5-2] and internally generated PULSTAR surveillance files:

1. The resistivity of the primary coolant is checked to determine if the resin is spent.
2. The pressure drop across the purification system filters are checked for loading.
3. All alarms or alerts associated with the coolant system are checked to determine operability.
4. Samples of the reactor primary coolant are checked periodically for radioactivity and impurities. Samples can be taken from the reactor pool, inlet and the outlet of the primary purification system, and primary coolant piping. The presence of fission products and abnormal levels of corrosion products would be identified from these samples.
5. All of the primary system mechanical and measuring equipment is periodically checked or calibrated.
6. A leak surveillance procedure has been established which provides for the earliest possible detection of any significant leakage from the primary system.
7. The radioactive liquid drain pipe from the sump pump to the waste tank vault is inspected to insure system integrity.
8. Pool liner test coupons are periodically inspected for material loss due to corrosion.

All materials in the primary coolant system, excluding the aluminum pool liner, are type 304/316 or 304L/316L stainless steel. Piping is a combination of schedule 10S and schedule 40S. The secondary coolant system piping is mainly schedule 40 carbon steel with some stainless steel components. Secondary chemical control system also contains PVC and polyethylene.

All of the embedded primary piping was full penetration welded, radiographed, and dye penetrant tested. Most piping connections to equipment including the delay tank are flanged for maintenance purposes.

The lower [redacted] of the reactor biological shield is barytes concrete while the upper section is standard concrete. The shield is poured around an aluminum liner which ensures water tightness and provides a surface that is easily decontaminated. The surface of the liner next to the concrete is coated to prevent concrete induced corrosion. The welds were radiographed and the liner was filled with water for a leak test prior to pouring the biological shield. Galvanic action between the aluminum liner and the stainless steel piping is prevented by the use of special couplings for dissimilar metals.

The three waste holding tanks are identical in design. The tanks are fiberglass and the sump discharge line to the holding tanks is constructed of PVC/CPVC piping. Refer to Figure 5-7.

5.2. Primary Coolant System

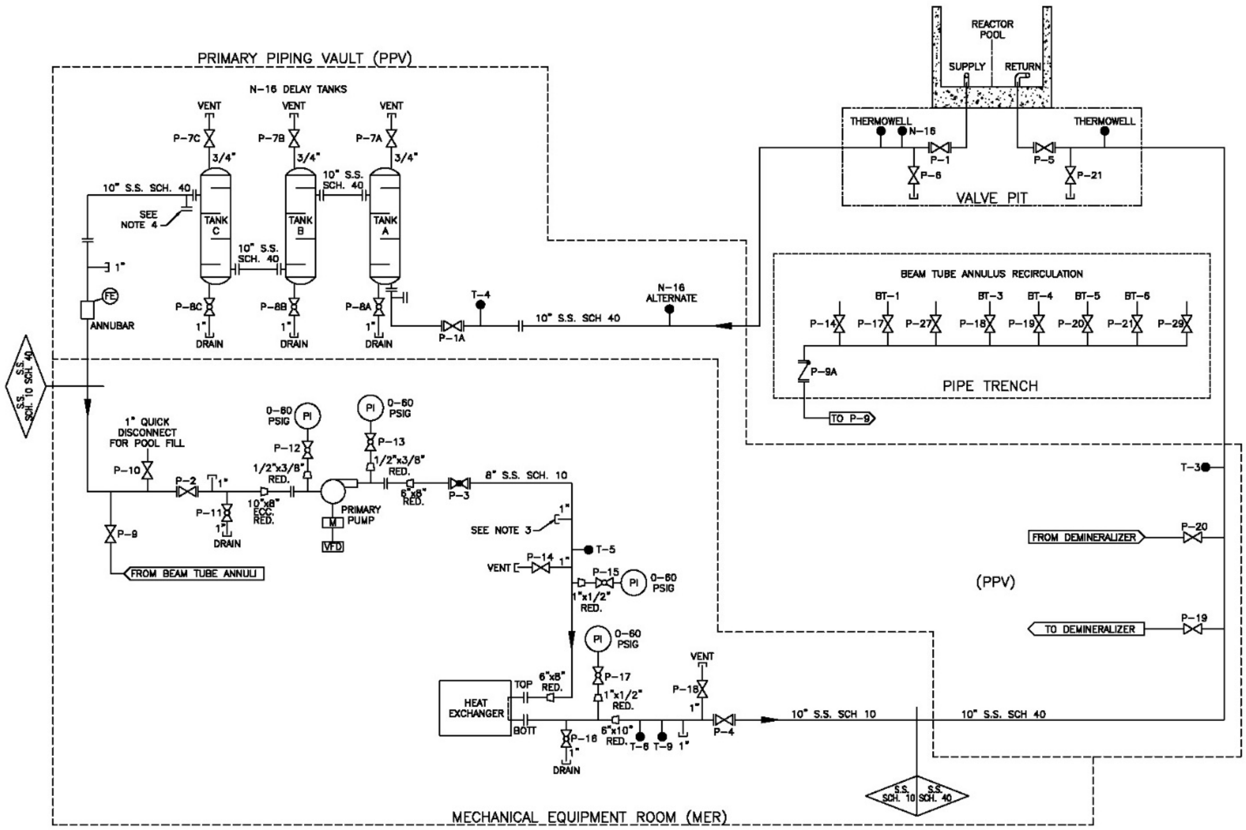

Figure 5-2 – Primary Coolant System

The primary coolant system is comprised of the reactor pool and liner, the ^{16}N delay tanks, the primary pump, the heat exchanger, a purification loop and associated piping as detailed in Figure 5-2.

The reactor pool consists of a [redacted] aluminum liner which is surrounded by normal and special high density reinforced concrete structure for shielding purposes.

Two cylindrical fuel storage pits, [redacted] in diameter and [redacted] deep extend below the bottom of the pool liner. Each fuel storage pit has [redacted] locations to store reactor fuel in a subcritical configuration. An additional [redacted] fuel storage locations are provided by [redacted] linear racks mounted along the north and south wall of the pool in a subcritical configuration. The rack along the north wall has [redacted] locations at a depth of [redacted] below the pool overflow weir and the rack on the south wall has [redacted] locations at the depth of [redacted] below the overflow weir. These [redacted] racks are suspended from hangers at the top of the liner. Refer to Section 6.1.5 for details.

There are two 10-inch pipe penetrations at the bottom of the pool liner for the primary coolant inlet and outlet connections. There is a 2-inch pipe attached to a weir about 2 feet from the top of the pool liner. This pipe directs any primary overflow to the liquid radioactive drain system. In addition to these penetrations for the passage of primary coolant, there are a number of liner penetrations which accommodate the beamtubes located on three sides of the core. Two sleeves, located near the top of the pool liner and approximately 1 foot above the overflow weir, allow passage for the tubing of the pneumatic transfer system.

During forced convection the primary water flows downward from the reactor pool through the reactor core and then flows into the core plenum. The outlet plenum is bolted above the 10-inch outlet pipe at the bottom center of the pool liner. It serves as a transition from the square reactor core grid to the round outlet pipe. It also serves as the support for the reactor core, and during forced convection, directs the flow of water from the reactor core to the coolant outlet pipe. The plenum is 3 feet high, 22 inches by 20.5 inches at the top, and 10 inches in diameter at the bottom. A flapper valve, which is a 15.75-inch diameter flat disc, on the side of the plenum is manually shut prior to initiation of forced flow and is held shut by the differential pressure created by the downward flow through the plenum. The flapper is counter-balanced to fall to a 30° open position upon loss of forced flow. The flapper valve is manually operated by means of a reach rod from the reactor bridge area. See Section 6.2 for more details.

The primary coolant flows from the plenum by means of a 10-inch pipe, through the pool liner and through a 10-inch manual isolation valve (P-1) in the valve pit. All of the stainless steel pipe that is embedded in concrete is bituminous coated and felt wrapped to prevent corrosion. The coolant passes through a tunnel, a second 10-inch manual isolation valve (P-1A) and then into the ^{16}N delay tanks located in the Primary Piping Vault (PPV). The PPV is below grade at the south side of the reactor building. The ^{16}N tanks, with a total nominal volume of 1350 gallons, are constructed of stainless steel and have internal baffling to delay the primary coolant for approximately 1½ minutes at a 1000 gpm flow rate. The 1½ minute delay allows for the high energy ^{16}N gamma rays produced as the coolant flows through the core to decay significantly. To allow for system venting and draining, the tanks have manual drain and vent valves.

The primary coolant then passes through a flow measuring device. See Section 7 for a description of the primary flow measuring system.

After the coolant leaves the PPV it enters the mechanical equipment room (MER) where it passes through another 10-inch manual gate isolation valve (P-2). The 10-inch pipe is reduced to an 8-inch pipe and enters the suction side of the primary pump. The primary pump is a single stage horizontal centrifugal pump constructed of stainless steel and provides flow at 1000 gpm with a discharge pressure of 15 psig. The speed of the pump is manually set with a variable frequency drive (VFD) to provide a constant flow rate of 1000 gpm.

Flowing at 1000 gpm the coolant passes through an 8-inch manual throttling globe valve (P-3) and reduces to 6 inches before entering into the inlet of the heat exchanger. The heat exchanger is a water to water counter-flow single pass plate type heat exchanger. The heat exchanger has 126 plates with the capability for additional plates for increased cooling capacity. All wetted parts are constructed of stainless steel. A heat exchanger pressure boundary breach accident is analyzed in Section 13.

Exiting the heat exchanger, the primary pipe diameter expands back to 10 inches and coolant passes through another isolation gate valve (P-4). The coolant exits the mechanical equipment room and re-enters the PPV. A small portion of the coolant is shunted to the primary demineralizer for purification. The primary coolant (cold leg) then enters the tunnel parallel to the hot leg and travels to a 10-inch manual gate isolation valve (P-5) before re-entering the reactor pool. The coolant is discharged from the bottom of the reactor pool through a 90° elbow which directs the water away from the core where it mixes with the bulk of the coolant in the pool.

The temperature of the pool is a nominal 90°F – 105°F when operating at 2 MW with a temperature rise across the core being 13.8°F. This results in a nominal pool outlet temperature of 103.8°F –

113.8°F. The core inlet temperature is controlled by regulating the temperature of the secondary coolant via a variable frequency drive (VFD) on the cooling tower fan motor.

In the natural convection mode of operation, the primary pump is secured and forced flow ceases. The cessation of flow through the pool outlet plenum results in a loss of the differential pressure across the flapper, and the flapper falls open due to the force of gravity. Water from the pool can now enter the outlet plenum through the open flapper valve and flow upward through the reactor core by convection, thus cooling the reactor.

In addition to the main coolant loop, there are several auxiliary components. There is an overflow weir located near the top of the reactor pool. Pool water entering the weir flows by gravity to the liquid radioactive drain system sump in the mechanical equipment room. The primary coolant system contains drain lines, vent lines and test connections. Each of the lines has a threaded pipe cap or a self-sealing quick disconnect to prevent accidental loss of primary coolant due to a valve operating error.

Instrumentation for the reactor coolant system and related support systems is shown in Figure 5-1 through Figure 5-5. The reactor pool is equipped with high and low water level alarms and a low water level reactor scram. Abnormal pool level alarms annunciate in the control room when the water level varies from the zero reference level (overflow) by +6 and -6 inches. If the water level drops to -36 inches and the reactor is operating, a scram signal will shut down the reactor and annunciate in the control room. The pool level gauge is located on the console. The loss of pool water is discussed further in Section 13.

The resistivity of the primary coolant is measured by a sensor suspended in the reactor pool and displayed on the radiation recorder in the control room. The demineralizer decontamination factor, which is a measure of the effectiveness of the resin, is checked monthly by reactor personnel.

The temperature of the primary coolant system is measured at six locations;

T-2	bulk pool (core inlet)
T-3	cold leg (pool return)
T-4	hot leg (core exit)
T-5	heat exchanger inlet
T-6	heat exchanger outlet
T-9	heat exchanger outlet – secondary control system

Each sensor is a resistance temperature detector (RTD) connected to a temperature transmitter which sends a 4-20 mA signal to the temperature recorder mounted on the control console. With the exception of T-9, all of the primary system temperatures will generate an annunciation in the control room when a predetermined setpoint is reached. The setpoints for each RTD are listed in the operations procedures.

There is an additional temperature monitor suspended in the reactor pool that annunciates in the control room when the coolant reaches a nominal 114°F±2°F.

The temperature of the secondary coolant is measured at the inlet and outlet of the heat exchanger with RTDs and temperature transmitters identical to those employed for the primary coolant instrumentation.

The primary coolant flow rate is measured using an annubar differential pressure sensor mounted in a straight section of the 10-inch pipe in the PPV. Integral to the differential pressure sensor is a 4-20 mA transmitter which transmits sensor measurement information to the control room. The flow rate, in gallons per minute, is displayed by a gauge mounted on the transmitter and also on the control console. If the reactor is operating above 150 kW, a scram will be annunciated in the control room when the primary flow drops below 950 gpm.

The flow rate of the purification system is 20 gpm and is locally displayed on the demineralizer. There are additional pressure gauges to show the pressure drop across the demineralizer resin and the filter cartridges.

The sump located in the MER has a high sump water level sensor which is connected to the annunciator panel on the control console. This alarm is actuated by a float switch which is set above the pump cut-on point. If the pump fails to start the water level in the sump will rise to the alert setpoint and cause the annunciator panel to sound along with the illumination of a high sump level alarm warning light.

A leak surveillance procedure has been established which provides for the earliest possible detection of any significant leakage from the primary system. Pool water level and evaporation measurements are taken each work day with adjustments for pool water temperature and any maintenance or experimental activities that may affect pool level. Both daily and trending period data is evaluated. A leak rate of 2 gallons per hour is detectable on a daily basis, whereas a leak rate of 1 gallon per hour is detectable over a longer trending period. Environmental procedures require sampling of water from Rocky Branch Creek and the groundwater monitoring well on site every calendar quarter. Refer to Sections 2.4.2 and 13.2 for details.

To minimize the potential for corrosion in the primary piping or pool liner and to reduce the amount of activated contaminants, the technical specifications[5-2] associated with the primary coolant system are as follows:

Technical Specification 3.9

The reactor shall not be operated unless the pool water meets the following limits:

a. The resistivity shall be ≥ 500 kΩ·cm.

The following specification for monitoring primary coolant will ensure the early detection of fuel clad failure while neutron activation analysis will give corrosion data associated with primary system components in contact with the coolant.

Technical Specification 4.6

a. The primary coolant shall be analyzed bi-weekly, but at intervals not to exceed eighteen (18) days. The analysis shall include gross beta/gamma counting of the dried residue of one (1) liter sample or gamma spectroscopy of a liquid sample, neutron activation analysis (NAA) of an aliquot, and resistivity measurements.

5.3. Secondary Coolant System

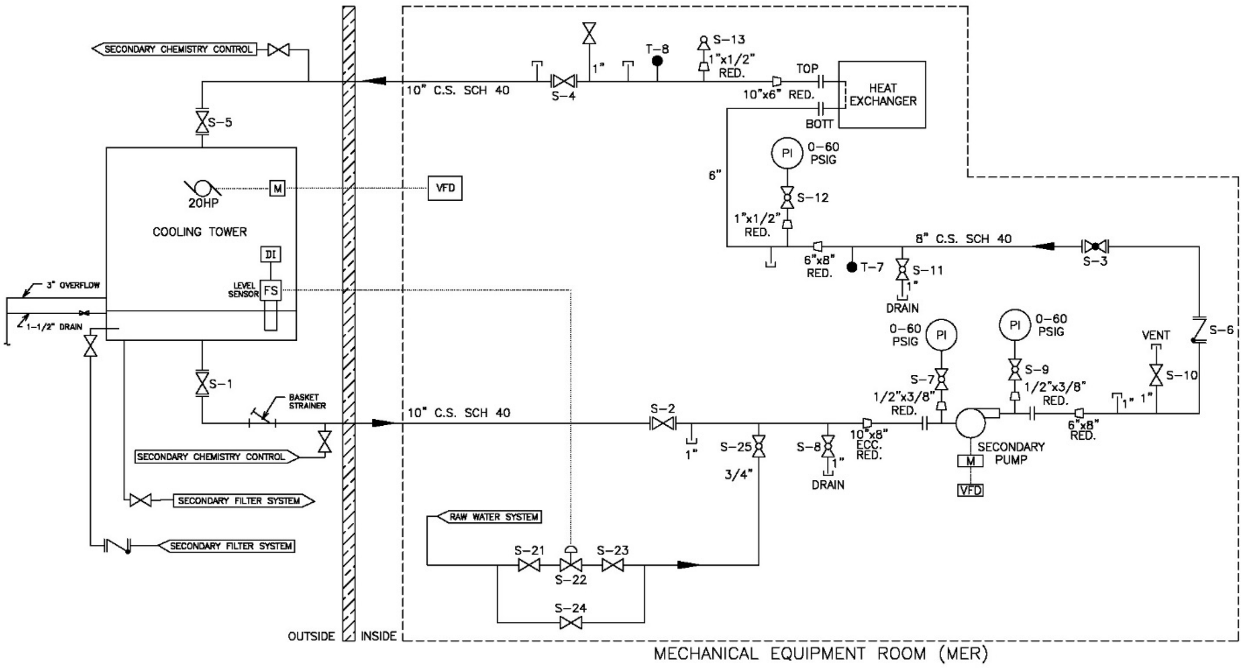

Figure 5-3 – Secondary Coolant System

The secondary coolant system consists of a pump, a heat exchanger, a 270 ton (nominal) cooling tower, a chemistry control system, and a filtration system with basin sweeper as detailed in Figure 5-3.

Cooled water is brought from the cooling tower, at 1000 gpm, by the secondary coolant pump which has a discharge pressure of 18 psig. The secondary system pump is a horizontally mounted, single stage, centrifugal pump located in the mechanical equipment room.

From the secondary coolant pump the water flows through a check valve and then through an 8-inch manual throttling globe valve (S-3) and to the inlet of the heat exchanger. The coolant leaves the heat exchanger and flows through a 10-inch manual isolation valve (S-4) and on to the cooling tower. A temperature sensor located on the exit of the heat exchanger (T9) measures the primary water temperature. This temperature is monitored by the VFD controller which will vary the speed of the cooling tower fan motor to provide the required amount of cooling. The coolant temperature entering the heat exchanger is a nominal 92°F. At full power and with a nominal secondary flow rate of 1000 gpm, the heat exchanger approach temperature is a nominal 9.0°F.

In the event of a loss of secondary cooling (e.g., failure of secondary pump, cooling tower fan or secondary water makeup leading to a low water level in the cooling tower basin) during full power steady-state operations, the primary coolant temperature would increase eventually leading to an abnormal temperature alarm on the primary return to the pool. In the event that the operator fails to take the proper corrective actions, then a High Pool Temperature Switch Alarm along with a Pool Temperature scram would actuate at less than 117°F, shutting down the reactor. The consequences for a loss of secondary cooling is bounded by the loss of primary flow accident. Refer to Section 13.2.

The cooling tower is a standard induced draft, counter-flow design. The basin is equipped with a water level controller and a thermostatically regulated 5.5 kW heater to prevent freezing during the winter. The fan is mounted on a single shaft which is rotated by a 20 HP, 480 volt weatherproof variable speed motor. The fan is electrically interlocked with the secondary pump so that it will not operate without flow in the secondary coolant system.

The chemical control system maintains the desired chemical balance in the secondary water to minimize corrosion and the fouling of heat transfer surfaces in the heat exchanger. This is accomplished by adding treatment chemicals and simultaneously bleeding the system whenever the system resistivity falls below the preset valve. The major components of the chemical control system are chemical addition pumps, chemical holding tanks, flow-through resistivity probe, a resistivity controller and a solenoid operated bleed valve. The chemical addition pumps and solenoid bleed valve are actuated by the resistivity controller. The pumps take suction on the chemical holding tanks and discharge through a check valve and stop valve into the secondary pump discharge. The resistivity controller, which is mounted in the chemistry control cabinet next to the cooling tower, receives the signal from the resistivity probe and actuates the solenoid bleed valve and chemical pumps at a predetermined resistivity value. Water samples are taken on a routine basis to measure the anti-corrosion chemical content in the water.

A filtration system circulates secondary water from the basin of the cooling tower through filtration disks, and back to the tower basin.

The secondary coolant system pipes and valves are constructed mainly of carbon steel with some stainless steel components, with the exception of the lines for the chemical control system which also have PVC and polyethylene components. The piping external to the reactor building is insulated and has heater tape installed between the pipe and insulation to prevent freezing.

The following specification for monitoring secondary coolant ensures the early detection of fission products in the secondary coolant and would provide evidence of a primary heat exchanger leak.[5-2]

Technical Specification 4.6

b. The secondary coolant shall be analyzed bi-weekly, but at intervals not to exceed eighteen (18) days. This analysis shall include gross beta/gamma counting of the dried residue of one (1) liter sample or gamma spectroscopy of a liquid sample.

5.4. *Primary Coolant Purification System*

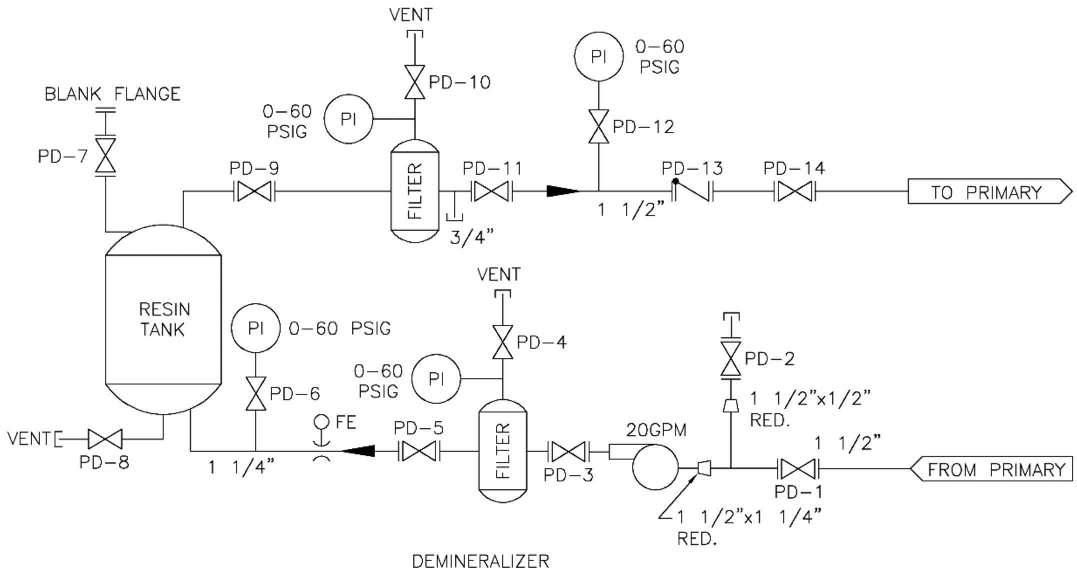

Figure 5-4 – Primary Coolant Purification System

The purification system for the primary coolant system is located in the PPV and the purification system pump draws from the 10-inch primary coolant system piping in the PPV. Refer to Figure 5-4. The purification system pump, providing a flowrate of 20 gpm, is a close-coupled, single stage, stainless steel, centrifugal pump located in the PPV. An isolation valve and a sampling connection are installed on the pump suction line. The pump discharge goes through a 25 micron filter to the demineralizer which uses a mixed ion exchange resin bed. The demineralizer is a closed stainless steel cylinder, 20 inches in diameter and 48 inches tall. In addition to the inlet and outlet connections, a vent on the top and a resin drain on the bottom are provided. All wetted surfaces are stainless steel. The demineralizer holds 8 cubic feet of nuclear grade resin. A retention element at the bottom of the column prevents resin beads from entering into the system. The water leaving the demineralizer flows through an isolation valve, a 25 micron effluent filter and through another isolation valve to a check valve. From the check valve the purified water returns to the primary coolant system in the PPV. A effluent sampling connection is provided between the check valve and the flow adjusting valve.

The effluent of the system is extremely high quality water with a resistivity greater than 500 kΩ·cm. Use of the purification system is an effective method to control corrosion. Samples of pool liner material have been suspended in the pool since the initial startup in 1972 to document corrosion of the liner. Measurements are taken periodically and have yielded no detectable weight loss. Operation of the purification system to maintain high quality water will enable the continued operation of the reactor without corrosion problems.

Since primary coolant passes through the reactor core, elevated levels of radioactive material in the coolant can be concentrated in the demineralizer. Procedures are used when changing the resin to minimize exposure levels and the spent resin is treated as radioactive waste. If a fuel assembly were to start leaking, higher than normal radiation levels would be detected in the demineralizer resin or the stack gas monitoring system. A fuel pin clad failure is analyzed in Section 13.

5.5. Primary Coolant Makeup Water System

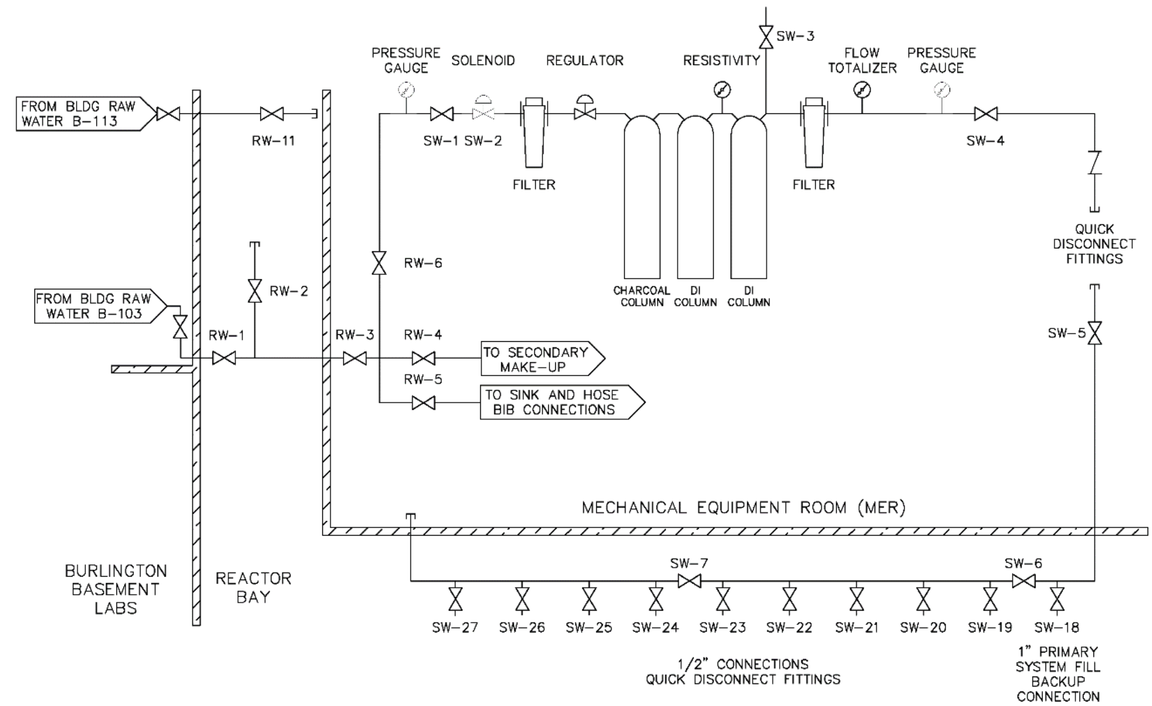

Figure 5-5 – Primary Coolant Makeup Water System

Since the PULSTAR is an open pool type reactor, primary coolant is lost as a result of evaporation. The primary coolant makeup water system consists of ion exchange resin beds and charcoal filters connected to the domestic city water supply which provides a source of highly demineralized water. Refer to Figure 5-5.

The city water flows through the purification beds, past a resistivity cell connected to an automatically controlled solenoid valve and through a check valve before entering the primary coolant system at valve P-10. To prevent inadvertent additions to the primary coolant system, the primary coolant makeup water system is normally physically disconnected and only connected during makeup operations by using a quick connect fitting.

A resistivity cell continuously measures the resistivity of the water exiting the primary coolant makeup water system. If the water purity drops below a preset level a solenoid valve closes, turning off the flow of makeup water thus preventing the addition of untreated water to the primary coolant system.

A check valve installed in the primary coolant makeup water system prevents the backflow of primary water into the domestic city water supply. Also the domestic city water supply is at a higher pressure than the primary coolant system which also prevents the backflow of primary coolant water.

To minimize losses to the primary coolant system and leakage of contaminated coolant all operations involving the primary coolant makeup water system are performed following approved facility procedures.

5.6. ^{16}Nitrogen Control System

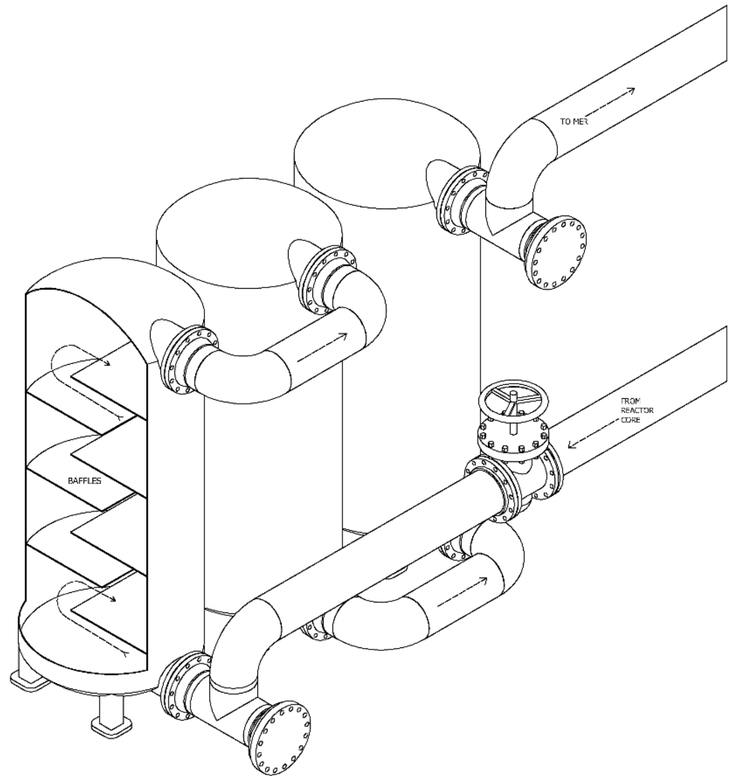

Figure 5-6 – Delay Tank Layout

As water passes through the reactor core, stable oxygen is activated via fast neutron reaction [$O^{16}(n,p)N^{16}$] forming radioactive ^{16}N, a high-energy beta and gamma emitter with an approximate 7 second half-life. When water cooled reactors operate above a few hundred kilowatts, the production of this radionuclide may require specific systems or procedures for limiting personnel exposure.

In reactors with forced convection cooling, the coolant carrying the ^{16}N out of the core may be passed through a system such as a large shielded and baffled tank. This delay allows the short lived ^{16}N radioactivity to decay significantly before the coolant emerges from the shielding.

At the PULSTAR reactor the system used is a series of delay tanks installed in a shielded subterranean room. Refer to Figure 5-6. Each delay tank is constructed out of stainless steel with a minimum free fluid volume of 450 gallons. Internal baffling is installed in each tank to create a torturous route between the inlet and outlet. Baffle dimensions and spacing are designed to maximize the uniformity of flow, minimize areas of stagnant water, and to minimize regions of greater than average velocity that could result in a delay time significantly less than the average delay time.

Three tanks in total are connected in series to provide no less than 10 half-lives of decay time which will reduce ^{16}N to acceptable levels before circulating coolant through to the rest of the system. See Section 11 for complete radiological analysis.

To minimize the possibility of rupture leading to the uncontrolled loss of primary coolant and the subsequent release of contaminated primary coolant, the design and fabrication of the tanks adhere to ASME boiler and pressure vessel codes and associated ANSI codes for pressure piping.

5.7. Auxiliary Systems Using Primary Coolant

5.7.1. Liquid Radioactive Drain System

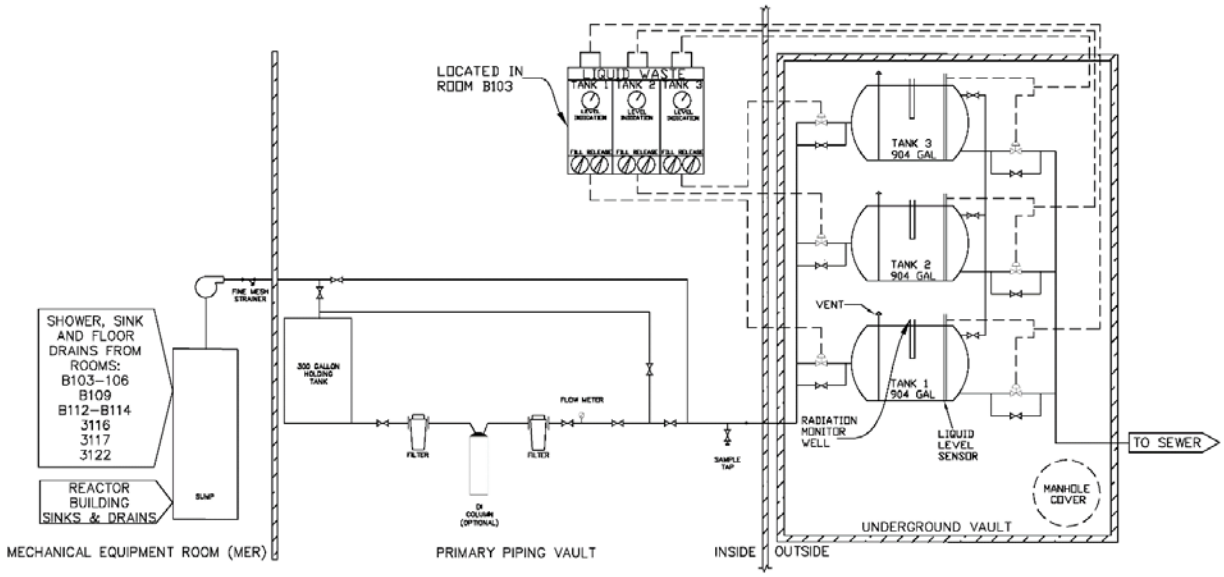

Figure 5-7 – Liquid Radioactive Drain System

The liquid radioactive drain system is designed to receive liquid waste water from designated spaces in Burlington Engineering Laboratories which may be contaminated with radioactive material. Refer to Figure 5-7. All such liquid is collected in the sump located in the MER. The sump is a rectangular concrete pit in the floor of the mechanical equipment room. The dimensions of the sump are 6 feet deep by 4 feet long by 1.5 feet wide. It has a special coating on the walls to aid any necessary decontamination operations. The sump pump is a vertically mounted 55 gpm pump with adjustable cut-on and cut-off points actuated by a float switch. From here the waste is pumped to the 300 gallon holding tank in the PPV. The waste water in the holding tank is filtered and then pumped to one of three waste tanks located in an underground concrete vault outside the reactor building. The tanks are denoted as Waste Tank No.1 through 3 consecutively, with Waste Tank No.1 closest to the reactor building. The waste tanks are provided with manual valves to release the waste in a controlled manner to the sanitary sewer system. The three waste tanks are identical in design. They are right circular cylinder, 904 gallon fiberglass tanks equipped with removable detector wells which extend to a depth corresponding to the tanks centerline. Manually operated valves are provided to completely isolate the holding tanks.

5.8. References

5-1 North Carolina State University PULSTAR Reactor, *Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor*, February 2019.

5-2 North Carolina State University PULSTAR Reactor, *R120 Facility License-Appendix A: Technical Specifications*, <date TBD>.

6. ENGINEERED SAFETY FEATURES

6.1. Summary Description

The concept of engineered safety features (ESFs) evolved from the defense-in-depth philosophy of having multiple redundant layers of controls designed to ensure the safe operation of the reactor. ESFs are designed to prevent and/or mitigate accidents, should they occur, by controlling the release of radioactive materials resulting from accidents to the environment. The ESFs at North Carolina State University (NCSU) include:

1. Confinement system
2. Natural convection cooling
3. Cooling of partially plugged fuel assembly
4. Control rod hold down
5. Shielded storage for radioactive fuel

ESFs can automatically be actuated by the protection instrumentation that monitors various parameters during reactor operation, be manually actuated by the reactor operator or be completely passive requiring no activation. The principal accidents they protect against are the uncontrolled release of radioactive material to the surrounding environment and the overheating of the core should forced convection flow of primary coolant be unavailable. Chapter 13 shows that the NCSU confinement building mitigates the consequences of the most limiting accident scenario to acceptable levels. Accordingly, a more extensive containment facility is not required and, therefore, not included as part of the ESFs. Several ESFs have been incorporated into the design of the PULSTAR reactor facility. The major safety features which have been incorporated into the facility design have resulted from the experience gained during the operation of pool type research reactors. The design bases for the following ESFs are that safety actions must be automatic and/or passive and require no reactor operator action to be initiated.

6.1.1. Confinement System

The confinement system design is based on the requirements dictated by the postulated design bases accident and its associated releases as discussed in Section 13. The results of Section 13 analyses confirm that the release of fission products is most improbable, while the amounts involved, if released, would be well within the capabilities of the confinement system. Further, the self-regulating properties of the PULSTAR reactor, along with the intrinsic safety features of the low enriched fuel design, limit the source and amount of materials which might be released into the confinement structure. By housing the NCSU PULSTAR reactor in a relatively airtight concrete confinement structure called the reactor building, and using confinement fans with high efficiency filters, charcoal adsorbers and automatically closing dampers, the release of any radioactive material will be controlled within the limits established by appropriate regulations consistent with 10 CFR Part 20.

The reactor building main ventilation system has two redundant 600 cfm confinement exhaust fans with separate filter and charcoal adsorber banks. This redundancy ensures that in the event of airborne contamination in the reactor building, the main ventilation will shut down and a confinement exhaust fan will maintain the reactor building at a negative pressure and purge the contaminated air through filters prior to discharge from the 100 foot high exhaust stack. In addition, the confinement fans may be powered by the auxiliary generator. A detailed description of the confinement system is given in Section 6.2.1.

6.1.2. Natural Convection Cooling

In the event of failure of the primary coolant pump or accidental shutoff or blockage of the discharge line, either the low flow or flapper open condition will automatically scram the reactor. The loss of flow will cause the flapper to open on the side of the plenum below the core region, allowing a flow path for natural convection cooling. The loss of flow analysis in Section 13.2.5.2 demonstrates that the core will not be damaged during loss of flow and the subsequent flow reversal period, even in the event that the flapper fails to open. A detailed description of natural convection cooling is given in Section 6.2.4.

6.1.3. Cooling of Partially Plugged Fuel Assembly

One type of accident which has resulted in the partial meltdown of MTR plate type fueled cores has been blockage of fuel assembly flow by foreign objects such as gaskets, plastic sheets, etc. The PULSTAR fuel assembly has an ESF in its design to mitigate the blocked flow type of accident. [redacted] detailed description and analysis of the cooling of a partially plugged fuel assembly is given in Section 6.2.5.

6.1.4. Control Rod Hold Down

The control rods for the PULSTAR reactor are engineered so that it is impossible to remove the complete control rod assembly without first unloading the four fuel assemblies adjacent to the control rod position. This is accomplished through small projections installed on the bottom of each control rod guide on which adjacent fuel assemblies rest when seated in the grid plate. This prevents the accidental removal of the complete control rod assembly and any resulting reactivity change that would be caused by such a removal. A detailed description of the control rod hold downs is given in Section 6.2.6.

6.1.5. Shielded Storage for Radioactive Fuel

Improper storage configuration of PULSTAR fuel could cause a reactivity excursion. To eliminate this potential, subcritical wet storage is provided in the PULSTAR pool by [redacted] linear storage racks, one with a capacity of [redacted] fuel assemblies and the other with a capacity of [redacted] fuel assemblies. There are also [redacted] round fuel storage pits located at the bottom of the pool liner each capable of storing [redacted] fuel assemblies.

During the initial loading of fuel into the wet storage racks and pool storage pit, k_{eff} was measured and confirmed to be with the Technical Specification limits of 0.9.[6-1] The k_{eff} for the [redacted] and [redacted] assembly rack is less than 0.6.[6-2,6-3] There are also [redacted] round fuel storage pits located at the bottom of the pool liner each capable of storing [redacted] fuel assemblies with a measured k_{eff} of less than 0.77.[6-2] Therefore, no critical array will exist, and all rack and storage assemblies would be significantly subcritical. A detailed description of shielded storage for radioactive fuel is given in Section 6.2.7.

6.2. Detailed Descriptions

6.2.1. Confinement System

The reactor building for the NCSU PULSTAR reactor includes the reactor bay, the mechanical equipment room (MER), the primary piping vault (PPV), the control room and the third floor ventilation equipment room (VER) as shown in Figure 1-4 through Figure 1-9. The lower east entrance, the loading dock entrance on the east side, the northwest entrance, the control room entrance from the second floor and the ventilation equipment room entrance are the confinement doors. These doors are constructed of steel, gasketed, equipped with double locks, and have security devices attached. The basement laboratory area on the north side of the reactor bay is a restricted area but is not part of the reactor building. The reactor building walls are poured reinforced concrete with a brick veneer on the exterior and a painted surface on the interior. All penetrations of the reactor building are sealed to minimize leakage and the possibility of releasing contamination to the environment. Penetrations are sealed in the following manner:

1. All pipes and ducts passing through the walls are sealed to the wall.
2. Electrical conduits are sealed to the walls and the wires are sealed to the inside of the conduit.
3. Major process pipes and ducts have isolation valves or dampers.
4. All doors leading into the reactor building are gasketed. They are also self-closing and self-latching.
5. All spare pipes and conduits are plugged or capped.

Reactor building confinement will automatically be initiated as a result of radiation levels in the effluent of the building ventilation or of radiation fields within the confinement area exceeding preset levels. These preset values are based on regulatory or emergency requirements, and can be found in Section 7 and Section 11. Confinement can also be initiated by a manual pushbutton or by the loss of power to the Radiation Alarm Panel.

The location of radiation detectors and isolation dampers are such that any release of radioactive material as postulated in Section 13 will be detected early enough to ensure that automatic initiation of confinement will mitigate any uncontrolled releases.

Confinement can be initiated by any of the following:

1. Manual confinement pushbutton in the control room
2. Manual evacuation pushbutton in the control room
3. Manual evacuation pushbutton in the basement laboratory hallway
4. Control Room area radiation monitor
5. Over-the-Pool area radiation monitor
6. West Wall area radiation monitor
7. Stack Gas radiation monitor
8. Stack Particulate radiation monitor
9. Stack Exhaust radiation monitor
10. Loss of power to the Radiation Alarm Panel

Area radiation monitors are located in the control room, above the reactor pool, and on the west wall of the reactor bay. The stack gas and particulate radiation monitors will detect radiation levels in the air being exhausted from the reactor building through the 100 foot (30.5 m) high air stack. In order to get a representative sample, the air is withdrawn isokinetically at a monitored flow rate downstream from the exhaust fan. The stack exhaust radiation monitor will give an indication of the gross radiation levels in the air stream.

The radiation levels detected by the area and stack radiation monitors are displayed on indicators and recorded on a recorder located in the control room. Confinement is automatically initiated as part of the evacuation sequence when any of the radiation monitors listed above reaches the alarm setpoint. Alerts and alarms also annunciate in the control room.

The confinement signal for each of the radiation monitors may be bypassed with a separate key on the radiation alarm panel. When a circuit is bypassed, a bypass indication for that circuit is energized on the radiation alarm panel. To ensure proper use of the bypass, authorization by the designated senior reactor operator (DSRO) is required. Lights are provided for circuits in the radiation alarm panel logic to indicate that each relay is in its normal state. The loss of this indication signifies either that a component failure has occurred, and/or an abnormal condition exists.

To prevent unnecessary initiation of the evacuation and confinement systems, specific radiation monitoring channel evacuation signals may be bypassed for the following times:[6-1]

1.	Ventilation Monitors	Less than five minutes immediately after starting the blower for the pneumatic transfer system.
2.	Over-the-Pool Monitor	Less than five minutes during the return of a pneumatic transfer system sample capsule.
3.	Over-the-Pool Monitor	Less than five minutes during removal of experiments from the reactor pool.

Upon initiation of confinement or evacuation, the supply fan and the exhaust fan shutdown and their spring return isolation dampers automatically close. The main dampers will indicate being closed when the dampers have gone full travel. This essentially closes all free paths of air into the reactor building. At the same time, confinement fan No.1 will start. After a nominal 55 seconds, should confinement fan No.1 fail to start, confinement fan No.2 will start. While either fan is running, air is purged from the reactor building at a nominal rate of 600 cfm, passing through all of the normal filters then through a 99.97% (removal efficiency) high efficiency particulate air (HEPA) filter and a charcoal adsorber.

The isolation dampers are electrically operated devices that can also be manually operated if necessary. All of the isolation dampers fail closed on loss of electrical power. There is an electrically driven damper motor on each of the confinement fans that is powered from the same circuit as the fan motor. In the event of loss of commercial power, the confinement fan and damper motor may be powered by the auxiliary generator.

Fan status indicators, damper position indicators, and differential pressure readings (magnehelic gauges) displayed in the control room provide the reactor operator with the necessary information to determine the status of the confinement system. The confinement fan dampers are designed to be normally closed and will not indicate open until the dampers have gone full travel.

Radioactive material generated by and potentially released by routine operation, or during accidents contain gamma emitting radionuclides, therefore during maintenance to any required radiation monitoring channel, one of the installed channels may be replaced for up to ninety days with a gamma-sensitive instrument which has its own alarm, or is observable by the reactor operator or reactor operator assistant. This time limit was chosen in order to allow sufficient time for procurement and testing of specialized equipment. In order to maintain a permanent record of radiation levels, during maintenance to the radiation rack recorder for up to ninety days, all monitoring channels which cause evacuation will be recorded manually at a nominal interval of 30 minutes while the reactor is not shutdown.

The reactor building is maintained at a negative pressure with respect to the outside (atmosphere) and a slightly smaller negative pressure with respect to the basement nuclear laboratories. The differential pressure (dp) between the reactor building and atmosphere shall be at least 0.2 inches of water when the main HVAC system is running or at least 0.1 inches of water with either confinement fan running.

Doors may be opened by authorized personnel for less than five minutes for personnel and equipment transport provided that audible and/or visual indication is available for the reactor operator to verify door status. If differential pressure is lost for greater than five minutes, an alarm in the control room will inform the reactor operator. Reactor operation may continue after a loss of dp (with main HVAC operating) for up to thirty minutes, while the loss of dp is investigated and corrected.

Several potential sources of radioactive gas and particulate releases exist at the PULSTAR reactor. These vary from the production of ^{41}Ar gas in the beamtubes or similar facilities to fission products from a failed experiment or ruptured fuel pin. The handling of these potentially radioactive effluent during both normal and confinement conditions requires an adequate ventilation system capable of minimizing uncontrolled releases to the environment and providing adequate ventilation for personnel and equipment. The size of the ventilation system is based on the magnitude of the release of potential sources. The functional requirements for operation are discussed in Section 13.

The ventilation and confinement system is shown schematically in Figure 6-1. Outside air for the reactor building is supplied at 180 cfm through the intake located on the third floor of the south side of the reactor building. The intake is protected by a steel grating. The air is filtered, heated or cooled as necessary, and distributed throughout the reactor building. The control room receives its air through a separate duct branching from the main supply box. It is separately filtered and conditioned to maintain personnel comfort and electronic equipment stability. Air from the reactor bay is drawn through a pre-filter and the main filters, monitored for radioactivity, and then discharged to the atmosphere through the 100-foot-high stack. There is a separate exhaust duct for the beamtubes and the thermal column utilizing a booster fan and an absolute particulate filter. This air is discharged into the exhaust plenum prior to the pre-filters. The pneumatic transfer system also discharges air to the exhaust plenum using a booster fan which is operated as required.

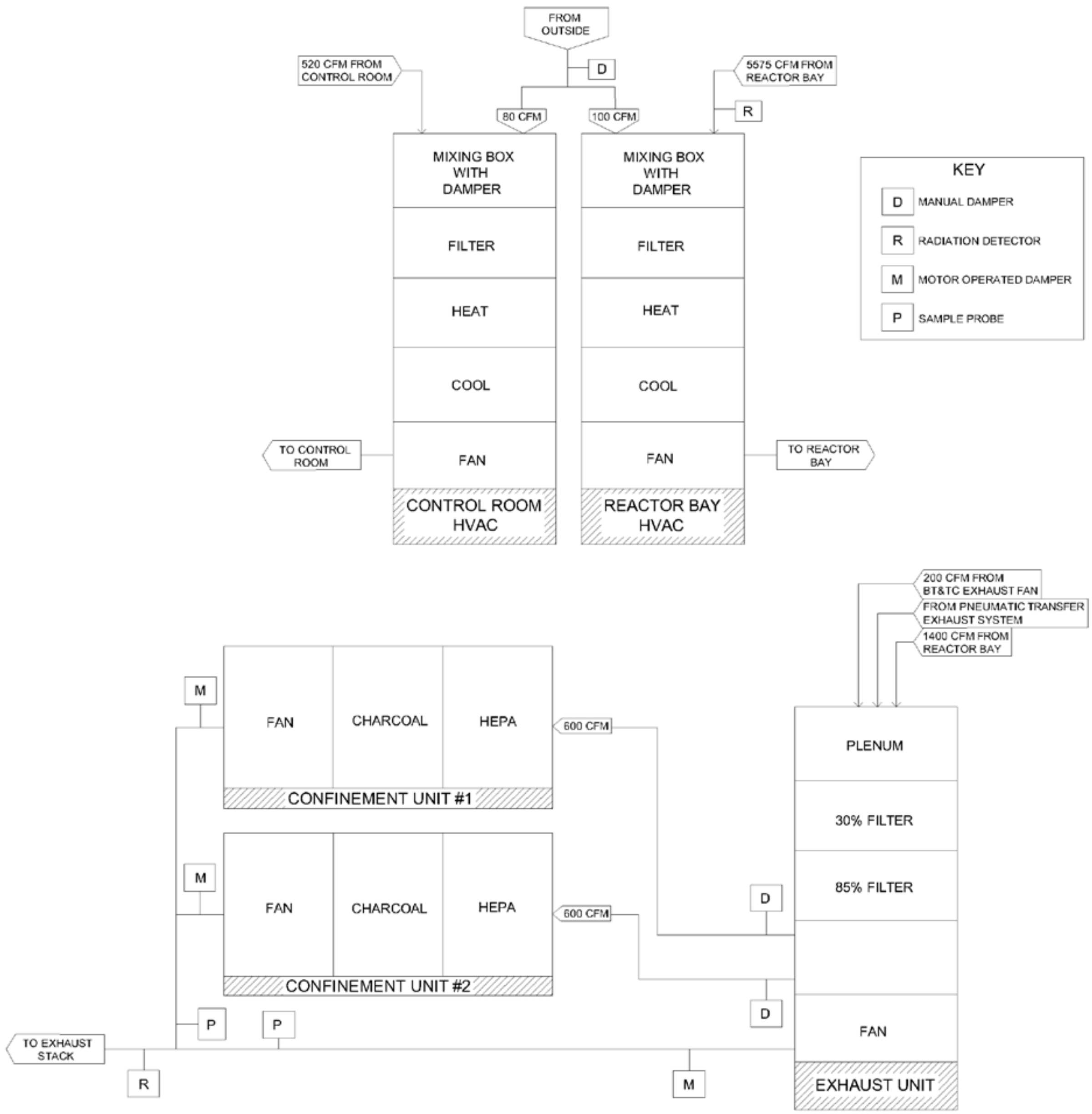

Figure 6-1 – Reactor Building Ventilation and Confinement System

Normally, all the air discharged from the reactor building passes through the exhaust plenum containing a pre-filter with an average efficiency of 30%, a main filter with an average efficiency of 85%, the exhaust fan, and up the stack at a rate of 1870 cfm. If confinement mode is initiated, the air being discharged is diverted from the main exhaust fan, which is automatically shut down, to one of the confinement filter trains. The air is now discharged at a 600 cfm rate through a 99.97% HEPA filter and a charcoal adsorber. The two confinement fans are interlocked so that only one can be operating at a time.

There are two exhaust stacks in a concentric configuration. The one that is visible from the outside, i.e. the outer stack, discharges air from the Burlington Engineering Laboratories south wing and is 100 feet tall. The reactor building stack is located concentrically inside the original stack to within 10 feet of the top to prevent back-flow.

The confinement system is designed to function automatically. It contains manual backups to ensure confinement operability. Confinement integrity is accomplished by sealing all reactor building penetrations and using industry proven components. An auxiliary generator fueled by natural gas provides a backup source of power to operate the confinement system.

The location of sensors and dampers ensures early detection and maximum isolation between the reactor building and the environment. Redundant relays and other design features are incorporated in the logic and actuation circuits in the confinement system. The overall system is designed to fail-safe to confinement.

Confinement system tests and inspections are specified in the technical specifications and in the internally generated PULSTAR surveillance files.

Technical specifications[6-1] associated with the confinement system are:

> Technical Specification 3.6
>
> The reactor shall not be operated, nor shall irradiated fuel be moved within the pool area, unless the following equipment is operable, and conditions met:
>
> a. All doors, except the control room, and basement corridor entrance: self-latching, self-closing, closed and locked. (1)
>
> b. Control room and basement corridor entrance door: self-latching, self-closing and closed. (2)
>
> c. Reactor building under a negative differential pressure of not less than 0.2” H_2O with the normal ventilation system or 0.1” H_2O with one confinement fan operating. (3)
>
> d. Confinement system (4)(5)(7)
>
> e. Evacuation system (6)
>
> (1)Doors may be opened by authorized personnel for less than five minutes for personnel and equipment transport provided audible and visual indications are available for the reactor operator to verify door status.
>
> (2)Doors may be opened for periods of less than five minutes for personnel and equipment transport between corridor area and the reactor building.

(3)During an interval not to exceed 30 minutes after a loss of dp is identified with Main HVAC operating, reactor operations may continue while the loss of dp is investigated and corrected.

(4)Operability also demonstrated with an auxiliary power source.

(5)One filter train may be out of service for the purpose of maintenance, repair, and/or surveillance for a period of time not to exceed 45 days. During the period of time in which one filter train is out of service, the standby filter train shall be verified to be operable every 24 hours if the reactor is operating with the reactor building in normal ventilation mode.

(6)The public address system can serve temporarily for the reactor building evacuation system during short periods of maintenance.

(7)When the radiation levels reach the alarm setpoint on any single area, or stack exhaust monitor, listed in Table 3-3, the building will be automatically placed in confinement as described in the Safety Analysis Report.

<u>Technical Specification 4.5</u>

The reactor shall not be operated, nor shall irradiated fuel be moved within the pool area, unless the following equipment is operable, and conditions met:

a. The confinement and evacuation system shall be verified to be operable within seven (7) days prior to reactor operation.

b. Operability of the confinement system on auxiliary power will be checked monthly but at intervals not to exceed six (6) weeks.[1]

c. A visual inspection of the door seals and closures, dampers, and gaskets of the confinement and ventilation systems shall be performed semi-annually but at intervals not to exceed seven and one-half (7 ½) months to verify they are operable.

d. The control room differential pressure (dp) gauges shall be calibrated annually but at intervals not to exceed fifteen (15) months.

e. The confinement filter train shall be tested biennially but at intervals not to exceed thirty (30) months and prior to reactor operation following confinement HEPA or carbon adsorber replacement. This testing shall include iodine adsorption, particulate efficiency and leak testing of the filter housing.[2]

f. The air flow rate in the confinement stack exhaust duct shall be determined annually but at intervals not exceed fifteen (15) months. The air flow shall be not less than 600 CFM.

(1)Operation must be verified following modifications or repairs involving load changes to the auxiliary power source.

(2)Testing shall also be required following major maintenance of the filters or housing.

6.2.2. Containment

Section 13 shows that the NCSU confinement system mitigates the consequences of the most limiting accident scenario to acceptable levels. Accordingly, a more extensive containment facility is not required and therefore is not included as an ESF.

6.2.3. Emergency Core Cooling System

Section 13.2.3 shows that the NCSU reactor core can air cool in the event of total loss of cooling water. Accordingly, an emergency core cooling system is not required and, therefore is not included as an ESF.

6.2.4. Natural Convection Cooling

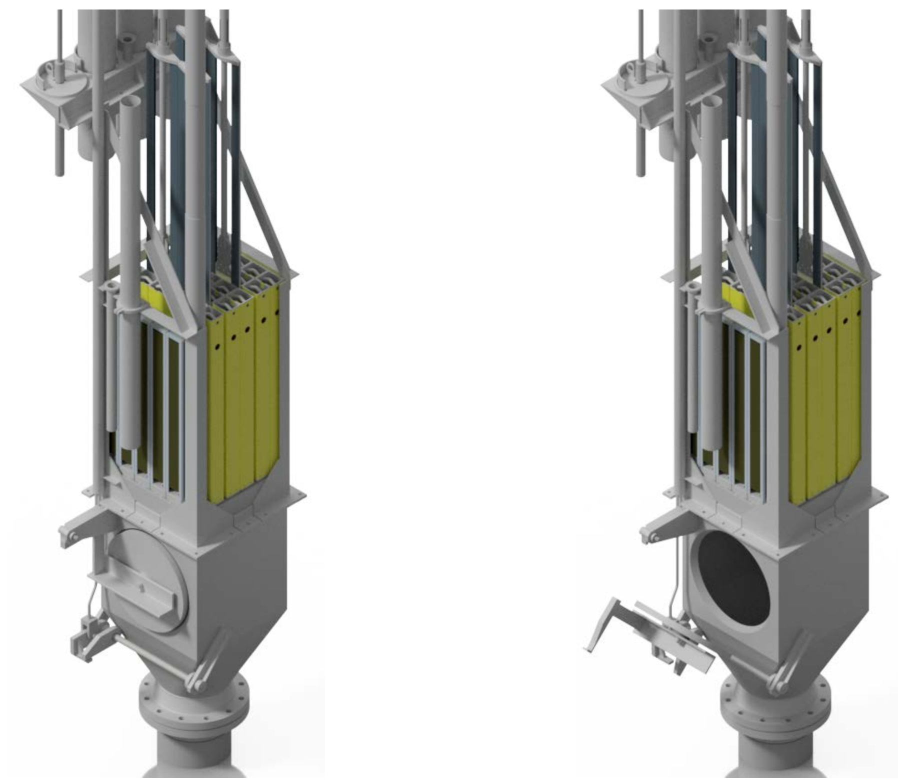

Figure 6-2 – Flapper Assembly Shown in the Closed and Open Positions

During normal operation of the reactor at power levels above 100 kW, core cooling is accomplished with a 1000 gpm primary coolant flow rate. In the event that the primary flow is interrupted by loss of the pump or other causes, a flapper valve on the plenum, which is located directly under the grid plate, will open and provide a path for natural convection cooling to be established within the reactor pool. Refer to Figure 6-2. The flapper valve is held in a closed position by the differential pressure created by the coolant flowing through the core and the pool static head at the plenum level. Closed is the normal operating position during forced convection flow cooling.

Calculations show that the PULSTAR core could be operated in natural convection cooling mode at a power level of 1 MW without departure from nucleate boiling (DNB) occurring in the core. Refer to Section 4.6.3.3. In the event that the coolant flow through the core is disrupted and the power level is above 150 kW, a scram signal will occur for low primary flow and the flapper will open due to a decrease in the differential pressure holding it shut. For scrams originating from a steady-state power level of 2 MW, calculations show that fuel centerline and cladding temperatures remain well below the limits during the flow transition period, even in the event that the flapper fails to open. Refer to Section 13.2.5 for detailed analysis.

The proper operation of the flapper and associated scram circuits are tested prior to daily operation. The flapper valve is tested under reduced flow situations (i.e., low differential pressure) as required by technical specification 4.3.e.ii using approved facility surveillance procedures.

Technical specifications[6-1] associated with this feature:

Technical Specification 3.3

The reactor shall not be operated unless the reactor safety system channels described in Table 3-2 are operable.

b. Safety Power Level Enable for Flow/Flapper scrams at ≤ 250 kW

d. Log N Power Level Enable for Flow/Flapper scram at ≤ 250 kW

e. Flow Monitoring scram when flapper not closed and Flow/Flapper scrams are enabled(2)

(2)Either the flapper scram or the flow scram may be bypassed during maintenance testing and/or performance of a startup checklist in order to verify each scram is independently operable. The reactor must be shutdown in order to use these bypasses.

Technical Specification 4.3

This specification applies to the surveillance requirements for the Reactor Safety System (RSS) and other required reactor instruments.

A channel calibration of the following channels shall be made semi-annually but at intervals not to exceed seven and one-half (7 ½) months:

e.ii. Primary Cooling and Flow Monitoring (Flapper)

6.2.5. Cooling of a Partially Plugged Fuel Assembly

One type of accident which has resulted in partial meltdown of MTR plate type fueled cores has been due to blockage of the fuel assembly flow by foreign objects such as gaskets, plastic sheets, etc. The PULSTAR fuel assembly has an ESF in its design to eliminate the blocked flow type of accident. [redacted] Refer to Section 13.2.5.1 for detailed analysis for the blocked flow accident.

6.2.6. Control Rod Hold Downs

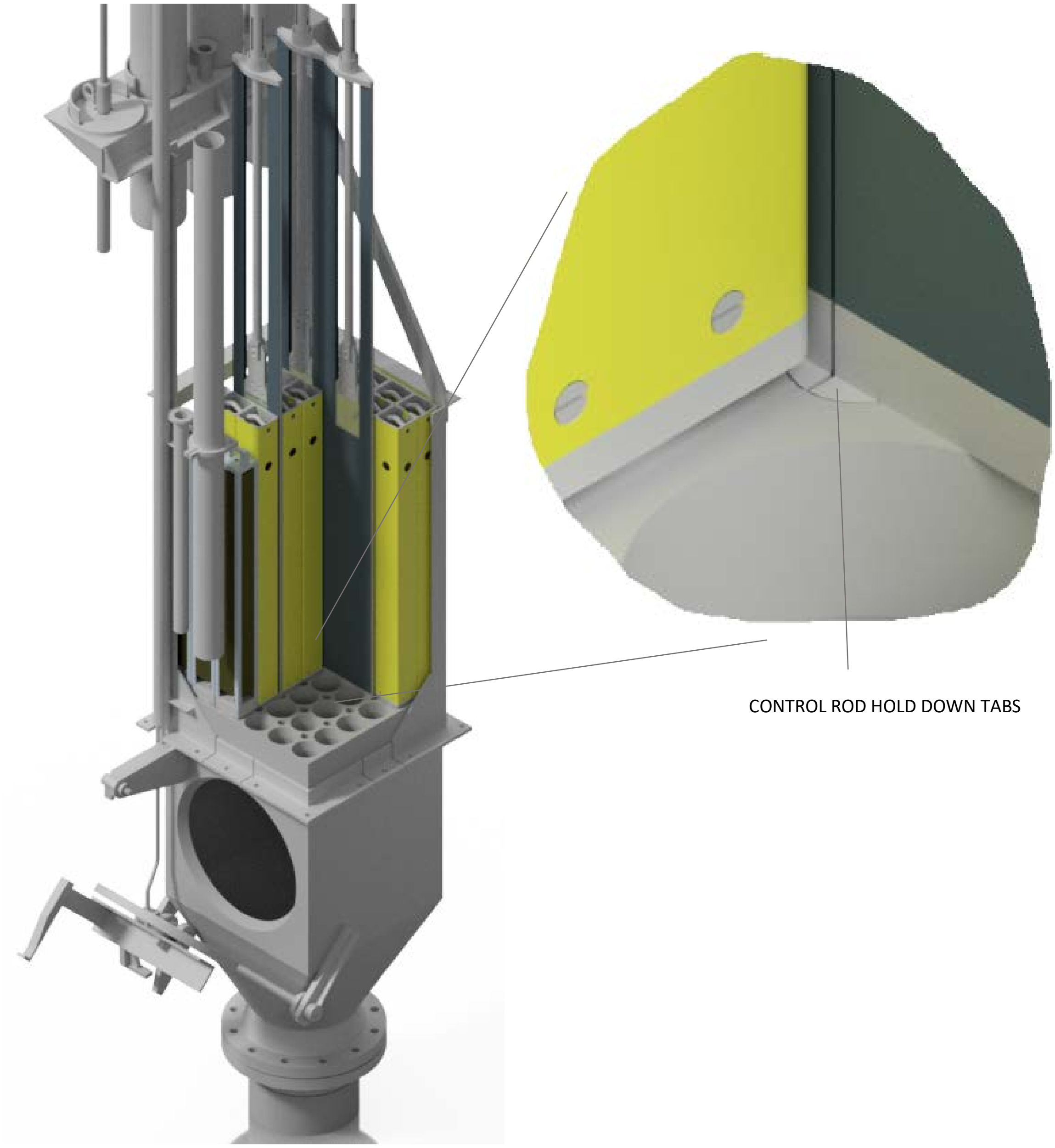

Figure 6-3 – Control Rod Hold Down Tabs

The control rods for the PULSTAR reactor are engineered so that it is impossible to remove the complete control rod assembly without first unloading the four fuel assemblies adjacent to each control rod position. This is accomplished by small projections on the bottom of each control rod guide on which adjacent fuel assemblies rest when seated in the grid plate. Refer to Figure 6-3. This ESF prevents the accidental removal of the complete control rod assembly and any resulting reactivity change that would be caused by such removal.

6.2.7. Shielded Storage for Radioactive Fuel

The pool design incorporates [redacted] fuel storage pits. The spacing and capacity of these storage pits are designed to have a multiplication factor of less than 0.9 when loaded with new 4% or 6% enriched fuel assemblies.[6-2] [redacted] irradiated fuel assemblies can be placed in each fuel storage pit. Being the [redacted], the storage pits are always filled with water to provide cooling by natural convection. The k_{eff} for the storage pits was measured to be 0.77 when loaded with 4% enriched fuel.[6-2] Six percent fuel is not permitted in the storage pits.

In the event repairs must be made near the core location, these fuel storage pits would be used to store fuel while repair work is executed. Additional shielding can be provided by positioning a lead slab or similar shield over the top of the loaded fuel storage pit. The shielding would be positioned in such a manner as to not obstruct cooling to the irradiated fuel assemblies stored in the pit.

An additional [redacted] fuel storage locations are provided by [redacted] linear racks mounted along the north and south wall of the pool in a subcritical configuration. The rack along the north wall has [redacted] locations and the rack along the south wall has [redacted]. These [redacted] racks are suspended from hangers about midway between the core and the top of the pool. The k_{eff} for the linear storage racks was measured to be 0.35 when loaded with 6% enriched fuel assemblies and 0.26 for 4% enriched fuel.

In the event of loss of pool water, the fuel assemblies in the storage racks and storage pits can air cool without exceeding the fuel or cladding safety limit. Refer to Section 13.2.3 for details.

Technical specifications[6-1] associated with this feature:

> <u>Technical Specification 5.3</u>
>
> Fuel, including fueled experiments, shall be stored in a geometrical configuration where k_{eff} is no greater than 0.9 for all conditions of moderation and reflection using light water. In cases where a fuel shipping container is used, the licensed limit for k_{eff} of the container shall apply.

6.3. References

6-1 North Carolina State University PULSTAR Reactor – Facility License – Appendix A – Technical Specifications, <date TBD>

6-2 Startup Test 2.12 Installation of Neutron Source and Fuel in the Reactor Pool, August 1972, January 1973.

6-3 Nuclear Reactor Program, North Carolina State University, Internal Report, December 2015.

7. INSTRUMENTATION AND CONTROL SYSTEMS

7.1. Summary Description

The reactor is operated from a control console located in the control room which is inside the controlled access area. All important information regarding the status of the reactor and reactor facility is displayed and readily available to the reactor operator.

The instrumentation for the North Carolina State University (NCSU) PULSTAR reactor includes both nuclear and non-nuclear channels. Also included is the scram logic unit and associated trip circuits that make up the reactor safety system. A combination of alarms, interlocks, drive inhibits and reverse drive functions are provided for the safe and efficient operation of the reactor. In this section, trips are frequently referred to as fail-safe. This means that upon loss of electrical power to an instrumentation channel, all trip circuits contained therein will act to limit reactor power or initiate reactor shutdown. Refer to the PULSTAR technical specifications for the current values of trip setpoints, power level settings and flow rates and to the PULSTAR operating procedures for the normal operating levels.

Engineered safety features (ESFs) present in the facility are designed to prevent accidents and control the release of radioactive materials to the environment should an accident occur. The PULSTAR ESFs include the confinement system and natural convection cooling. ESFs may be automatically actuated by the protection instrumentation that monitors various parameters, or manually by the reactor operator. For example, a high radiation alarm will actuate the confinement system to prevent the release of radioactivity to the environment. This signal may also be generated manually by the reactor operator in response to high radiation levels detected by radiation monitors. Engineered safety feature systems and components are discussed in detail in Chapter 6.

The reactor safety related instrumentation is comprised entirely of analog circuitry with the exception of the digital data recorders which have no control functions.

7.2. Design of Instrumentation and Control System

Independent power measuring channels provide for continuous indication of power from the source range to full power. Redundant scrams are provided for overpower conditions. Other pertinent facility parameters are monitored and displayed on the reactor control console to provide the reactor operator with current facility status.

7.2.1. Design Criteria

The instrumentation and control system (I&C) for the NCSU PULSTAR reactor is designed to perform the following functions:

1. Provide the reactor operator with information on the status of the reactor and reactor facility
2. Provide the means for insertion and withdrawal of control rods
3. Provide for automatic control of reactor power level
4. Provide for detecting conditions that could lead to limiting safety systems being exceeded and automatically scram the control rods to terminate the condition and shut down the reactor.

The elements of the I&C system that are important to safety include both redundancy and diversity, therefore a single failure will not prevent a return to a safe shutdown condition. The single failure criteria and the use of redundant and diverse channels are elements of the design bases for the I&C system. The I&C system is designed to be fail-safe and control logic chains have been developed to ensure safe operation under all conditions. The main parameters which are monitored and provide inputs to the control logic are:

1. Neutron flux level
2. Control rod status
3. Primary coolant flow, temperature and level
4. Electrical and control power status
5. Radioactivity levels
6. Experiment status

7.2.2. Design-Basis Requirements

The primary design basis for the NCSU PULSTAR is the safety limit of 2200 °F for fuel cladding temperature for PULSTAR type fuel.[7-1][a] To prevent exceeding this safety limit, automatic scrams are provided for those important process variables that could lead to a safety limit being exceeded.

The design basis for the reactor control console is that the reactor operators be provided with a central location from which they can safely operate the reactor. Instrumentation associated with the reactor core, primary coolant system, secondary coolant system, auxiliary systems, and experimental facilities, provide the operator with the ability to monitor conditions throughout the facility. The I&C system provides the operator with the ability to remotely start and stop equipment throughout the facility. The I&C are grouped as described later in this section. The parameters they monitor and the equipment they control are both diverse and redundant. An annunciator system is provided on the panel to alert the operator to an abnormal condition and to facilitate both the diagnosis of the abnormal condition in the facility as well as the selection of the appropriate response to the condition.

7.2.3. System Performance Analysis

The NCSU PULSTAR I&C system has had an excellent performance history since the reactor achieved initial criticality in August, 1972. All of the equipment and subsystems that comprise the I&C system have been well designed and maintained. Components important to safety are both diverse and redundant. They are calibrated and inspected/tested for operability as required by the technical specifications using approved facility procedures. Each major component of the Reactor Control and Safety Systems, along with all surveillances required by the Technical Specifications, are monitored and tracked by the PULSTAR Surveillance File System. Inspections, testing, maintenance and calibrations are performed and reviewed by qualified reactor personnel. All maintenance is documented and reviewed so that any trends, such as a drifting channel, become apparent. In addition, there is an ongoing program to upgrade and/or replace these systems, components and equipment with the latest available technology.

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[a] Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

7.2.4. Determination of Operating Setpoints for Reactor Power

The total 2σ uncertainty associated with determining true reactor power has been calculated to be 3.2%.[7-4] In addition, the reactor power scram channels have a measurement uncertainty of 0.5%, therefore the maximum setpoint for the reactor power scram should be no less than 3.7% of 2 MW. Table 7-1 lists the setpoints for reactor overpower scrams and the overpower reverse.

Table 7-1 – 2σ Setpoints for 2 MW Forced Convection Mode

Power Level	Description
$License\ Limit = 2.0\ MW$	Maximum power level allowed by license.
$LSSS = 2.0\ MW$	Maximum safety system setpoint for power level.
$SSS_{max} = 2.0\ MW - (2.0\ MW * 3.7\%) = 1.92\ MW$	Maximum safety system setpoint for power level accounting all uncertainties.
$SSS_{SCRAM} = 2.0\ MW - (2.0\ MW * 5.0\%) = 1.90\ MW$	Actual safety system setpoint for SCRAM.
$SSS_{reverse} = 2.0\ MW - (2.0\ MW * 7.5\%) = 1.85\ MW$	Actual safety system setpoint for Reverse Drive.
$OPL = 2.0\ MW - (2.0\ MW * 10.0\%) = 1.80\ MW$	Nominal operating power level.

7.3. Reactor Control System

7.3.1. Control Rods

The reactor is controlled by positioning the neutron absorbing rods in the space between the rows of fuel assemblies. Three of the control rods are labeled Safety No.1, Safety No.2 and Safety No.3; the fourth is the Regulating Rod which may be operated by an automatic channel to maintain the reactor at a specified power. The automatic channel consists of a flux controller, which is incorporated within the linear channel, a power demand potentiometer mounted on the control console and auto and manual pushbuttons. The interlocks and inhibits for withdrawing the control rods are:

1. Reactor keyswitch ON
2. No scram demand
3. Linear level trip #1 – reverse
4. Source range trip #1 -- less than 2 cps
5. Source range trip #2 – greater than 9×10^4 cps

There are no interlocks that will prevent driving the control rods into the core.

The control rod magnets are energized when the reactor keyswitch is in the ON position and there are no scram demands present. The rods may be withdrawn either individually or in gang of any combination of rods. The worth of the rods and their withdrawal speed are such that the total reactivity insertion rate of all the rods on gang will not be greater than 200 pcm/second.[7-2]

Control rod drive mechanism (CRDM) position indication from 0 to 24 inches is provided on the control console via a 4-20 mA signal generated by absolute encoders mounted on the control rod drive motors. DOWN and UP lights will also illuminate when the CRDM is at the travel limits. A rod SEATED light illuminates when the control rod is fully inserted. Any of the above indications along with the OFF magnet light will inform the reactor operator if it is the CRDM or the CRDM along with the control rod that is at the indicated position. Position indication is also provided by precision rulers mounted locally on the control rod drive housings.

7.3.2. Automatic Control System

Once a steady-state power level has been attained through manual control, this level may be maintained by the automatic channel if the following interlocks are satisfied:

1. Regulating rod above 13.5 inches.
2. Gang drive switch is in the neutral (middle) position.
3. Linear level trip #4 – flux controller absolute deviation (FC ABS DEV) is not lit.

If the interlocks are satisfied, then the reactor may be placed into automatic control by pressing the AUTO pushbutton. Indications that the reactor is in automatic control are:

1. The amber light in the MANUAL pushbutton goes out.
2. The green light in the AUTO pushbutton illuminates.
3. The Auto Channel Disengaged annunciator goes out.

When one of these interlocks is no longer satisfied after automatic control is achieved, the automatic channel will disengage and an annunciator on the console will activate. The automatic channel may be used to continually position the regulating rod to maintain reactor power at a specified level. The flux controller produces a series of pulses that actuate either an up-drive or a down-drive relay supplying electrical power to the CRDM motor that will bring the power level back to the desired point. For safety purposes, the maximum deviation is limited by the flux controller absolute deviation trip (FC ABS DEV). This deviation is set to ±3%. If this value is exceeded, the automatic channel will disengage and an annunciation is generated. The automatic channel is not required for reactor operations and may be used at the discretion of the reactor operator.

7.3.3. Interlocks and Protective Actions

A control rod inhibit is defined as the prohibition of the withdrawal of any control rod. Inhibits are discussed in Section 7.3.1.

An interlock is a mechanical or electrical device that will prevent a particular action from occurring until all prerequisites for that action are satisfied. The PULSTAR reactor has the following interlocks:

1. Fission chamber movement

   The fission chamber up-drive and down-drive requires that there is no movement of any of the control rods.

2. Up-drive power for the individual control rod drive mechanisms

Up-drive power for the individual control rod drive mechanisms requires the gang insert switch be in the OUT position, and no reverse demand and/or no source range channel inhibit be present.

3. Up-drive power for the gang control of the drive mechanisms

   Up-drive power for the gang control of the drive mechanisms requires the ganged insert switch be in the OUT position, and no reverse demand and/or no source range channel inhibit be present with the reactor keyswitch in the ON position. To allow for testing of the control rod drive mechanism (without the rods magnetically coupled) during the performance of the startup checklist the source range channel inhibit does not prohibit the up-drive power for the gang control with the reactor keyswitch in the OFF position.

4. Magnet power

   Magnet power is energized if electricity is available to the console, the reactor keyswitch is ON, and there are no scram demands present.

5. Automatic control

   Automatic control of the CRDM is allowed if the Regulating Rod is above 13.5 inches, the gang switch is in the neutral position and the Linear Level Trip #4 (FC ABS DEV) is not tripped.

One of the functions of the reverse drive is to automatically decrease power without unnecessarily dropping all the control rods by a protective scram. If the reactor power were to slowly increase and exceed 1.85 MW on the linear channel, the four control rods would drive to the lower limit position as long as the acknowledge pushbutton was not pressed. The ganged insert switch drives the control rods into the core as long as the switch is in the IN position. This is the normal method of shutting down the reactor. Whenever magnet power is lost (scram) and the reactor keyswitch is ON, a reverse drive is generated to ensure that in the event a control rod did not fully seat, the CRDM would attempt to push it down.

7.4. Reactor Protection System

A reliable reactor protection system functions to ensure that all modes of reactor operation are safe; therefore, the protective action of the system is designed to automatically terminate operations should safe operating conditions cease to exist.

The NCSU PULSTAR reactor safety system (RSS) features predominately automatic shutdown mechanisms. The reactor safety system is defined as that specified combination of instrumentation channels and associated circuitry which either provides the automatic protective action or provides that alarm which requires that manual protective action be taken. Specifically, the RSS consists of the scram logic unit with the magnet current circuits, the protective instrumentation channels (not including secondary readouts), and the associated circuitry. The protective instrumentation channels which are specified to be part of the reactor safety system are listed in Table 7-2.

Table 7-2 – Reactor protective instrumentation channel setpoints and required actions

Protective Channel	SSS[a]	LSSS[7-2]	Required Protective Action
Reactor On Keyswitch			Manual Scram
Manual Scram Button			Manual Scram
Startup Channel	2 cps		Inhibit when less than 2 cps and reactor power is less than 4 watts
	9×10^4 cps		Inhibit when greater than 9×10^4 cps and reactor power is less than 4 watts
Log N Channel	150 kW	≤ 250 kW	Enable Flow/Flapper (F/F) Scram
Linear Channel	1.90 MW	2.0 MW	Automatic Scram
Safety Channel	150 kW	≤ 250 kW	Enable Flow/Flapper (F/F) Scram
	1.90 MW	2.0 MW	Automatic Scram
Flow Measuring Channel	950 gpm	900 gpm	Automatic Scram when F/F enabled
Flow Monitoring Channel			Automatic Scram when F/F enabled
Flow and Flapper Scram Bypass Test Switches			Only one switch operative at one time
Pool Level Measuring Channel	17’6”	≥ 17’	Automatic Scram
Over-the-Pool Monitor	100 mrem/hr		Alarm
Pool Temperature Measuring Channel	116°F	117°F	Automatic Scram
Pool Temperature Monitoring Channel	114°F	117°F	Alarm

[a] Setpoints listed are the typical setpoints for the channel. Setpoints may be set at more conservative values.

7.4.1. Scram/Alarm Control Circuits

The scram/alarm circuits are energized through the power circuit breaker while the protective channels receive their electrical power through the instrumentation circuit breaker. The power circuit breaker is mounted on the control console and is not connected to a UPS or battery backup, therefore a loss of offsite power will de-energize the scram logic unit causing a scram. Any demand for protective action (scram demand) activates the appropriate scram/alarm circuit. Typical scram/alarm circuits are shown in Figure 7-2. The scram relays are normally energized and any open circuit condition, either an actual scram demand, loss of electrical power, or relay failure will cause the scram relay to de-energize and remove the input signal from the scram logic unit. Upon the loss of this signal, the scram logic unit interrupts power to the control rod magnets and causes an automatic shutdown of the reactor. Simultaneously, the scram annunciator horns sounds and lights illuminate on the control console to notify the reactor operator which channel initiated the scram. When either a scram or an alarm is received, the following sequence takes place:

a. The activated annunciator panel scram or alarm light(s) will be in the fast flash mode, and the alarm will be energized.

b. Pressing the acknowledge pushbutton silences the alarm and the activated alarm light goes to slow flash.

c. Pressing the first reset pushbutton performs the following:

  1. The alarm will lock in solid if the alarm condition still exists.
  2. The alarm clears automatically when the alarm condition clears.

After a scram condition clears, the scram must be cleared by pressing the scram reset pushbutton.

7.4.2. Scram Logic Unit

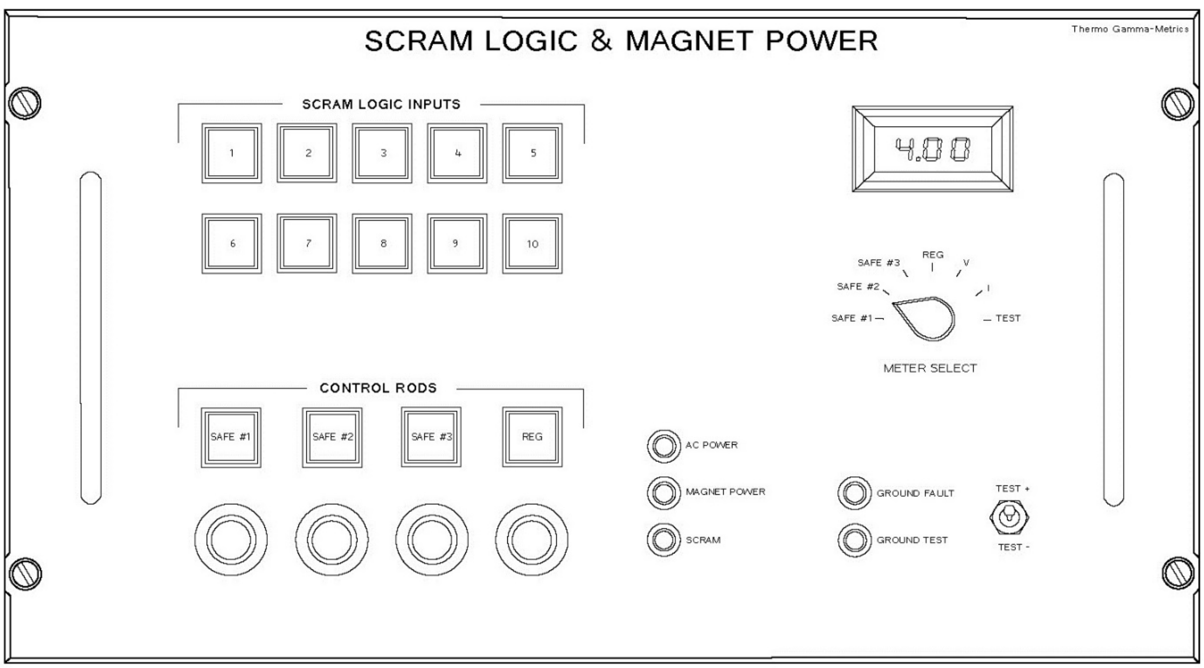


Figure 7-1 – Scram Logic Unit Front Panel Display

The scram logic unit is the central hub of the reactor safety system. This unit contains the power supply for the magnets and the system level logic for all protective channels initiating an automatic scram. A schematic diagram of the scram logic unit is shown in Figure 7-2.

The power supply for the magnets is energized only if the power and instrumentation circuit breakers are closed and the reactor keyswitch is in the ON position and all scrams are reset. The output of this supply is +28 VDC and is capable of supplying an adjustable current up to 125 mA to each magnet. The scram logic unit has ten input channels; however, currently only seven are active.

Once the power to the magnets has been interrupted by a scram demand, a lockout circuit ensures that the current cannot be restored until the reactor operator manually resets the scram logic unit.

Neither the manual scram nor the reactor keyswitch is capable of being bypassed by any means. In order to test the protective action of the flow measuring channel and the flow monitoring channel independently, a scram bypass switch for each channel is mounted on the control console. The two switches are spring return with safety covers and are wired so that only one channel can be bypassed at a time, and if both switches are operated, neither channel is bypassed. This ensures that there is at least one channel active for protection against loss of primary coolant flow.

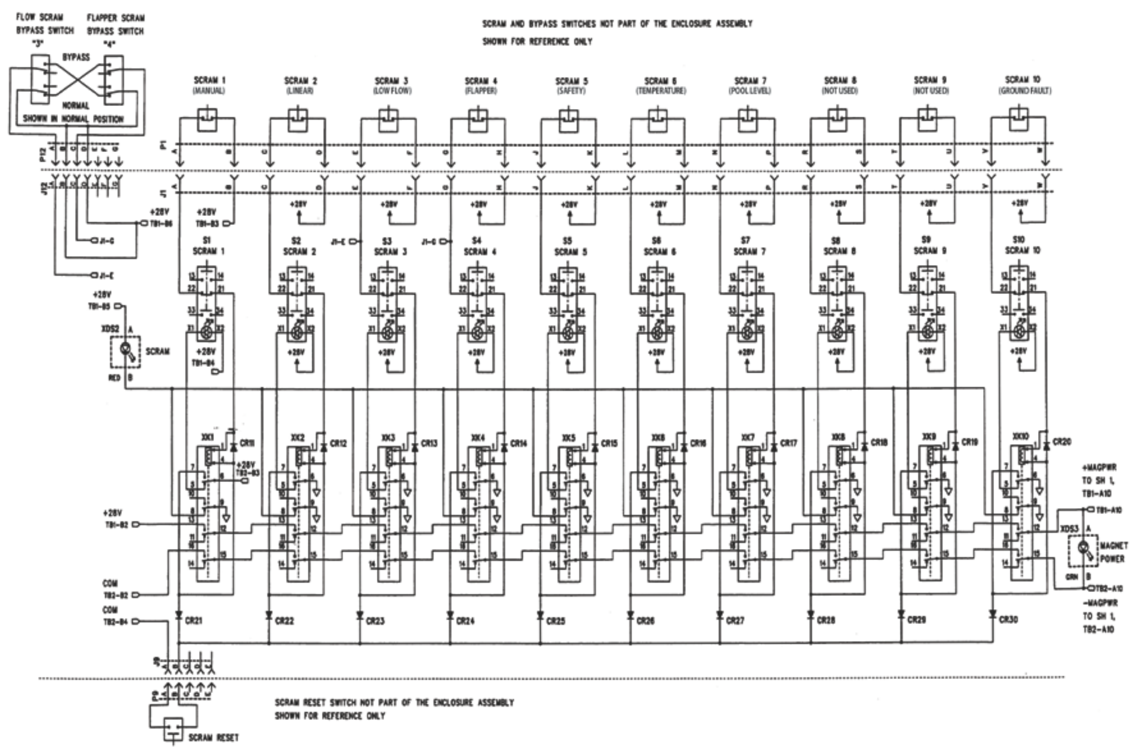


Figure 7-2 – Scram Logic Unit Schematic Diagram

The scram logic unit provides power to the four control rod drive scram magnets. Ten relays, controlled by the protection system trip circuits, are used to interrupt magnet power in the event that a protective action is needed. Refer to Figure 7-2.

The scram logic unit consists of:

1. Power supply

   Primary line power is received at via the J3 connector and routed through a circuit breaker, line filter then to the 28 VDC, 2A power supply.

2. Scram relays

   During operation, all scram contacts and scram logic inputs are closed. Power is provided to each control rod magnet via the current limiting and adjust networks. The scram relays are connected in series in both the positive and the negative supply line. If any scram condition is detected, the contacts open removing the scram logic signal to the unit. The loss of the scram logic signal causes the individual scram relay to de-energize. This action completely removes magnet current from the scram loop. On a scram, the tripped condition is annunciated to the operator via the front panel scram logic inputs LEDs. The pushbutton-switch LEDs simulate a scram condition during testing to verify proper circuit function.

3. Control rod scram pushbuttons

   Magnet current may be interrupted via the front panel magnet control pushbuttons to drop individual control rods.

4. Ground fault detection circuit

   Since simultaneous grounds on the magnet current scram bus could possibly defeat a protective action trip, a ground fault detection circuit is provided in the scram logic unit. This circuit consists of a balanced bridge and a current sensitive relay. One leg of this bridge is intentionally connected to ground. The current sensitive relay is normally de-energized. However, if a ground fault to any part of the magnet power supply bus occurs, the bridge becomes unbalanced and the relay will energize. Annunciators and ground fault test switches are provided on the front panel of the unit.

   A return to center toggle switch is located on the front panel to permit testing of the ground fault circuit. It supplies current from either the +28 VDC bus or the -28 VDC bus to ground to test the ground fault circuit.

5. Multifunction Display meter

   A display meter is provided to monitor individual control rod magnet current, power supply voltage and total supply current. Ohms Law is used to measure individual control rod currents by measuring the voltage drop across a precision fixed value resistor in series with the current path of each of the four control rods. The control rod current may be adjusted via the current adjust potentiometer in the individual control rod current loops.

7.4.3. Nuclear Instrumentation

The nuclear instrumentation consists of four separate channels to measure neutron flux in the reactor and to initiate protective action for specific conditions. The system consists of a source range channel, a log and linear channel, a linear channel, and a safety channel. Their ranges overlap sufficiently to accurately monitor reactor power (neutron flux) from a few neutrons per second up to ten megawatts. A ^{16}N channel also provides an indirect measurement of neutron flux in the core by monitoring decay gamma radiation produced by fast neutron activation of the oxygen in the primary coolant. The overall functions of the system are to:

- measure the power level and the rate of power change.
- provide the reactor operator with meter and recorder outputs of the power level.
- provide usable signals to the reactor safety system and control circuits.

7.4.3.1. Source Range Channel

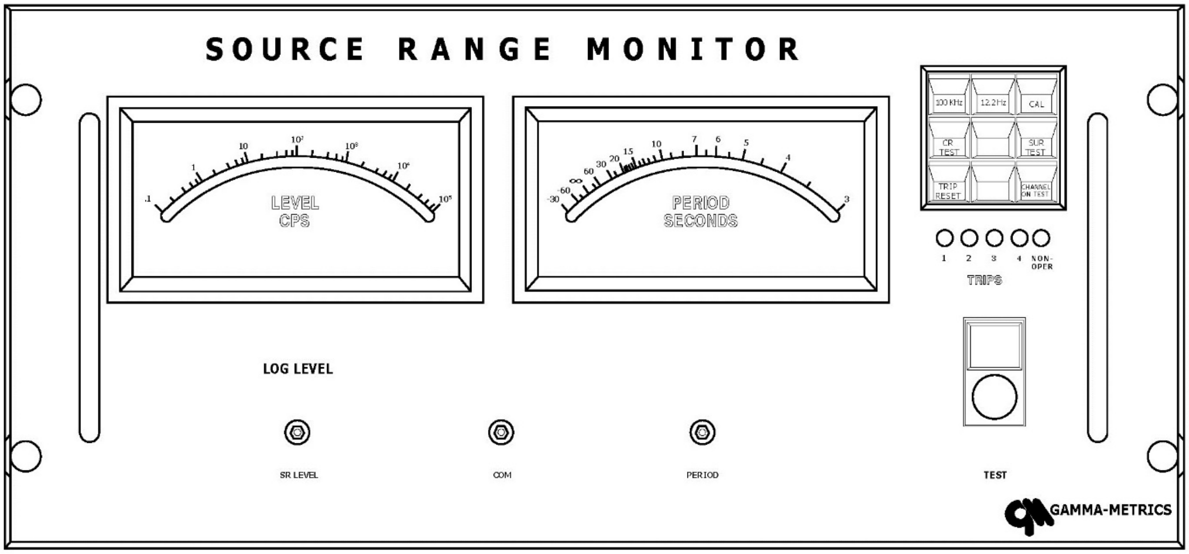


Figure 7-3 – Source Range Monitor Channel Front Panel Display

The source range channel (SR) measures neutron flux levels in the reactor from the source range to approximately 10 watts of power. The SR channel consists of a uranium lined fission chamber detector, drive assembly, source range monitor (SRM) and the reactor power recorder. The SRM contains the high and low voltage power supplies, a pulse preamplifier, discriminator and bypass filter, test generator and a SR log count rate and period circuit. The log count rate meter (LCRM) and the startup rate meter are located on the front panel of the SRM. Refer to Figure 7-3. The LCRM has a range of six decades from 0.1 cps to 10^5 cps. Because the detector is mounted on an adjustable drive mechanism, the SRM can be used over a wide range of reactor power. The startup rate scale indicates a period from -30 seconds through infinity to +3 seconds.

At very low power levels it is necessary to count neutron pulses rather than measure the current as is done with an ionization chamber. The highly enriched uranium coating on the inside of the detector readily fissions by incoming neutrons and the fission products produced ionize the gas within the detector, which is biased by high voltage, generating pulses of electrical current.

The source range detector is mounted in a watertight canister suspended just above the reactor core by a drive mechanism attached to the reactor bridge along with the control rod drive mechanisms. The height of the detector can be adjusted over a 24-inch travel by means of a drive switch on the console. Detector position is indicated by a meter and up and down limit lights. Coarse adjustment of the fission chamber height can be made at the upper extension coupling just below the drive mechanism.

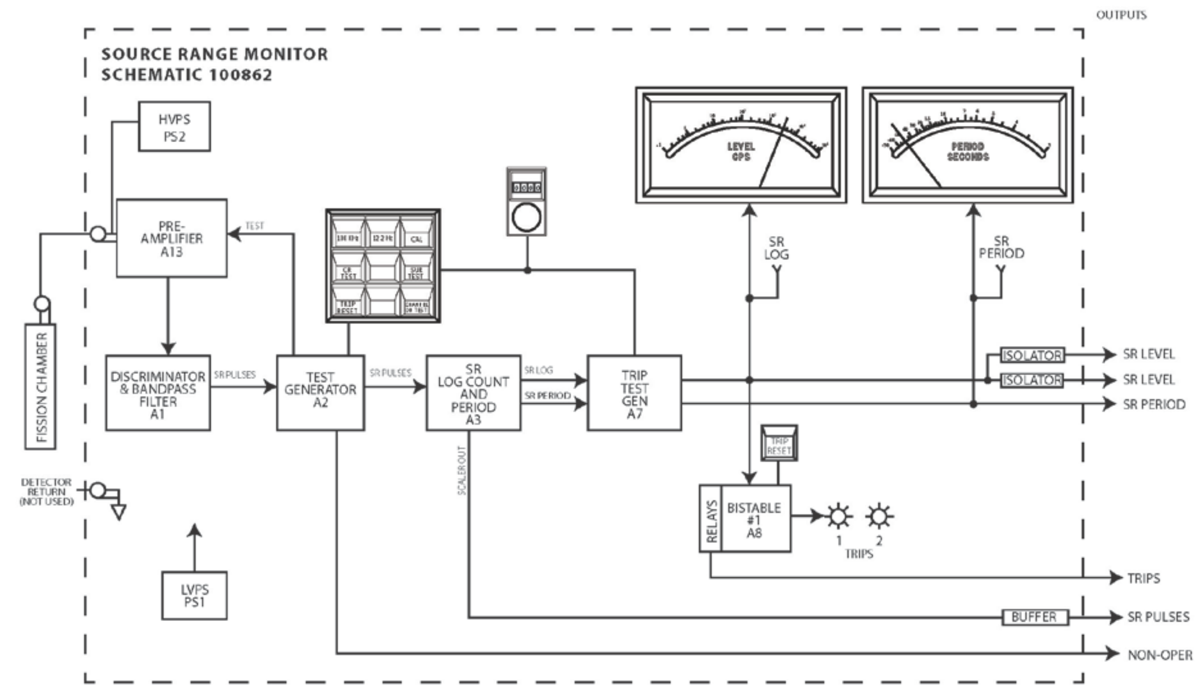


Figure 7-4 – Source Range Monitor Channel Block Diagram

The SRM receives pulses produced not only from fission products, but also from gamma radiation and noise. In general, the pulses produced by fission products are of much greater amplitude than the other pulses and can be electronically discriminated or separated by pulse height in the SRM. The discriminator and bypass filter provides a pulsed signal to the test generator circuit which buffers the signal prior to sending the signal to the SR log count rate and period circuit. The SR log count rate and period circuit provides a level signal and a rate output signal, which drives the front panel meters. Refer to Figure 7-4. The SR log count rate circuit provides a level signal to the reactor power recorder through an isolator.

The SRM contains bi-stable trip circuits which produce a rod withdrawal inhibit whenever a non-operate condition exist or, if count rate is <2 cps or $>9\times10^4$ cps. The <2 cps inhibit ensures that there is sufficient subcritical multiplication taking place and that sufficient counts are being measured to accurately indicate the fission rate in the core. The $>9\times10^4$ cps inhibit ensures the channel is not saturated by an excessively high count rate. The non-operate inhibit is caused whenever the ±15 volt power supply voltage is low, the high voltage power supply is low, or the channel-on-test and any other front panel pushbutton is activated. All of the trip circuits are of fail-safe design so that in the event of loss of power to the channel, a SR Inhibit will be generated. A test circuit provides inputs at 12.2 Hz, 100 Hz and +3 seconds to functionally check the drawer prior to startup.

The source range channel is calibrated semi-annually but at intervals not to exceed 7½ months.

7.4.3.2. Log and Linear Channel

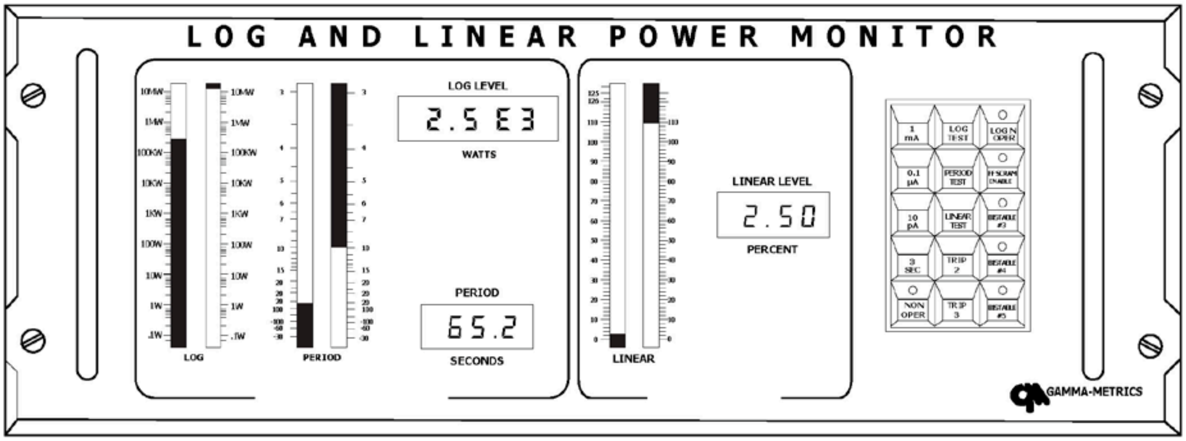


Figure 7-5 – Log and Linear Channel Front Panel Display

The log and linear channel is used to measure the reactor power from less than one watt to 10 megawatts on one continuous scale and is used to provide intermediate range data during the approach to full power. The channel consists of a compensated ion chamber, a high voltage, a low voltage, and compensation voltage power supplies, log amplifier, linear and period amplifiers, bi-stable trips, and the Reactor Power Recorder.

The detector for this channel is a neutron sensitive, gamma compensated ionization chamber (CIC). The CIC is constructed with two volumes having a common collector electrode. One volume is sensitive only to gamma radiation. By using a coating of boron, the other volume is sensitive to neutrons as well as gamma radiation. The two volumes are connected electrically with power supplies of opposite polarities so that the resulting output current is due to neutrons alone. The detector is surrounded by a 1-inch thick lead sleeve with a 1½-inch thick lead bottom plate and a ½-inch thick lead top plate. The detector is mounted in a waterproof aluminum canister suspended from the bridge near the core with a screw mechanism for adjusting the final height of the detector. Water is prevented from leaking into the detector canister by pressurized nitrogen supplied from portable tanks.

The log amplifier measures the current produced by the CIC. As the name implies, the output of the log amplifier is a voltage proportional to the logarithm of the input current. This output voltage is used to drive the display driver, period amplifier, linear amplifier, and the bi-stable trip circuits. Refer to Figure 7-6. The log amplifier provides an isolated output to the reactor power recorder.

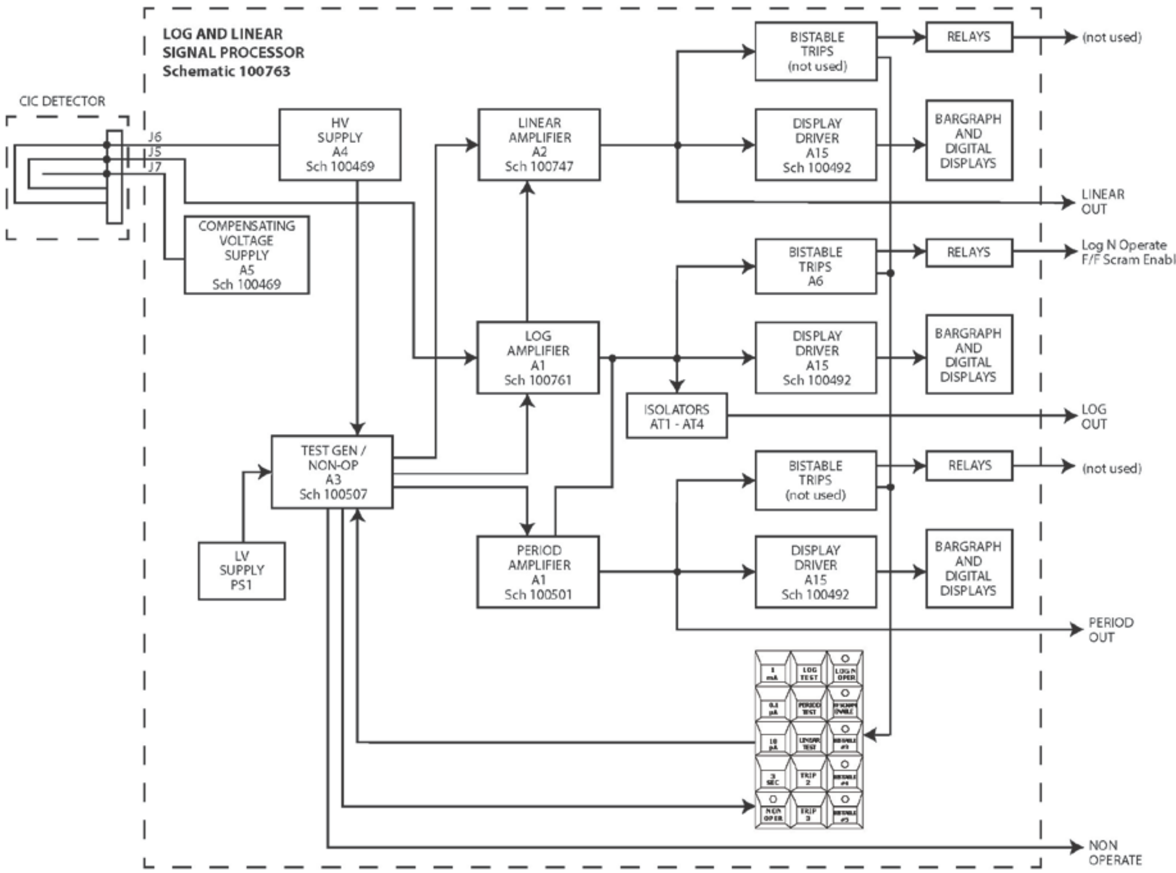


Figure 7-6 – Log and Linear Channel Block Diagram

The period amplifier provides startup rate information on the front panel meter ranging from -30 seconds though infinity to +3.0 seconds. Refer to Figure 7-5. The linear amplifier receives an input from the log amplifier and uses this signal to provide a linear indication of power from 0% to 125% reactor power.

The log amplifier contains two bi-stable trip circuits. The log N Operable is a downscale trip set at 4 watts. The lighted switch illuminates when the circuit has been reset. The F/F enable is an upscale trip that enables the low primary flow and flapper open scram circuits to produce an automatic scram when the power level exceeds 150 kW and the primary coolant flowrate drops below the low flow scram setpoint and/or the flapper is not fully closed. Both of these trip circuits are fail-safe.

Calibration circuits are built in to functionally check the trip circuits, and to check the accuracy of the log, linear and period amplifiers. The following test signals are provided to test the channel: 1 mA, 0.1 pA, 10 pA, 3 seconds, non-operate, log test, period test and linear test. All of the front panel test pushbuttons spring return to the operate position which prevents test information being displayed when the pushbutton is released.

The log and linear channel is calibrated semi-annually but at intervals not to exceed 7½ months.

7.4.3.3. Linear Channel

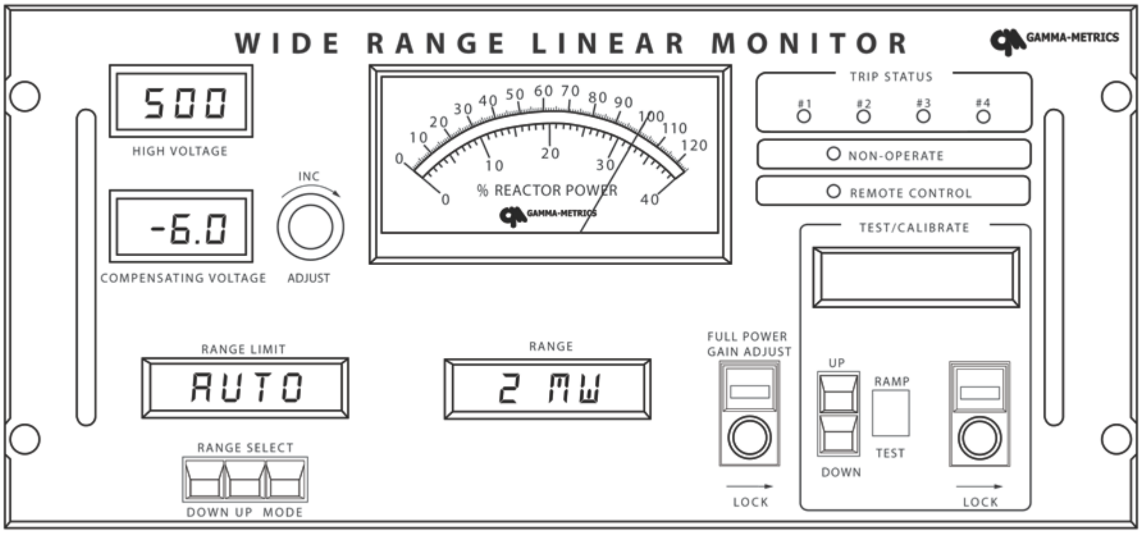


Figure 7-7 – Linear Monitor Channel Front Panel Display

The linear channel provides reactor power monitoring from source to power range levels. It combines the wide operating range of the log based scales with the accuracy of a linear scale to provide accurate power measurement from milliwatt levels to full reactor power. To accomplish this, a dual scale is provided that ranges from 0 to 40% or 0 to 125% of the displayed range depending on whether the detector signal is in the lower or upper part of the decade.

The range has three operating modes:

1. Auto Range
2. Manual Range
3. Upper Range Limit

Auto range is used to ensure that the indicated power reading is always on an accurate part of the meter. Manual range is used if the reactor will be staying at a specific power level and it is desired for the range to not change automatically. Upper range limit is selected to allow the unit to auto range up to a selected power level. The range mode can be changed on the front panel.

The linear channel uses a compensated ion chamber. The signal from the compensated ion chamber connects to the rear panel using a standard BNC connector. High voltage for the ion chamber connects to the rear panel using a standard SHV connector while the compensating voltage for the ion chamber is provided at the rear panel through a standard MHV connector.

The CIC is mounted in a waterproof canister identical to that used for the log and linear channel. Vertical position adjustment and nitrogen pressurization are provided.

High voltage and compensating voltage to the ion chamber are displayed on the front panel. Trip status lights, a non-operate light and remote control indicators are provided on the front panel as well as testing and calibration switches with their adjustments. Refer to Figure 7-7.

Linear output is available at the rear panel connectors as an isolated 0-10 VDC. This voltage can be selected to a voltage proportional to detector current for each decade or, for increased accuracy, for each range. The range information is available at the rear panel connectors as a 0-10 VDC analog output and a TTL digital output.

Bi-stable trips are available at the rear panel connectors as normally open and normally closed contacts. A non-operate signal is also provided at the rear panel connectors.

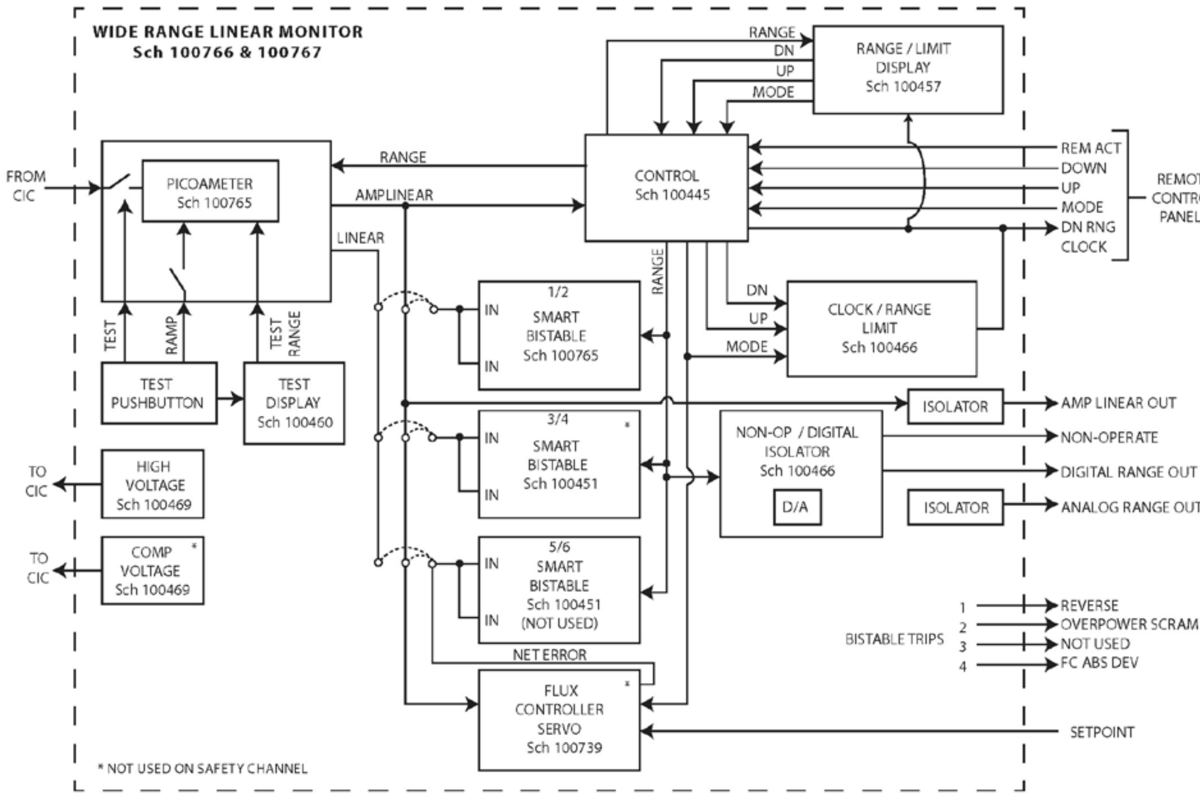


Figure 7-8 – Linear Monitor Channel Block Diagram

The trip circuits are:

1. Overpower reverse.
2. Linear overpower scram.
3. Spare.
4. Flux controller absolute deviation (FC ABS DEV).

The overpower reverse trip (trip #1) will cause all control rods to drive in when reactor power exceeds the setpoint. Refer to Figure 7-8. The second trip (trip #2) produces a linear channel overpower scram. The FC ABS DEV (Trip #4) prevents automatic channel operation when the demand potentiometer setting differs from actual reactor power by more than ±3% of the setpoint. Trip #3 is a spare and not presently used.

Flux controller outputs are provided at a rear panel connector to be used for rod position control. The flux controller which is an integral part of the unit is the major component of the automatic channel. The interlocks that must be satisfied prior to engaging the automatic channel are discussed further in Section 7.3.2.

The linear channel is calibrated semi-annually but at intervals not to exceed 7½ months.

7.4.3.4. Safety Channel

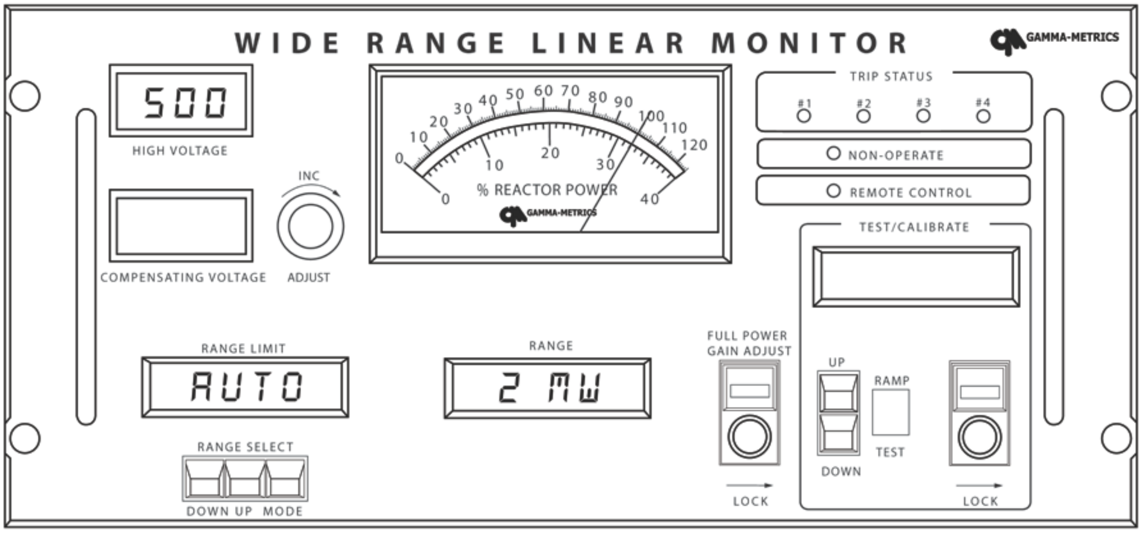


Figure 7-9 – Safety Monitor Channel Front Panel Display

The safety channel is the redundant channel for the linear channel which also measures and indicates the reactor power level. Because the safety channel is used primarily in the power range where current produced by gamma radiation is negligible compared to neutrons, compensation is not necessary therefore the safety channel utilizes an uncompensated ion chamber (UIC) instead of a compensated ion chamber. The UIC is mounted in a waterproof canister identical to those used in the log and linear and linear channels but without the lead shielding. As with the other detectors, vertical position adjustment and nitrogen pressurization are provided.

The monitor has two trip circuits that initiate automatic protective action if any preset levels are exceeded. Refer to Figure 7-10. The two trip circuits in the monitor are of fail-safe design and provide trips.

The trip circuits are:

1. Spare
2. Safety overpower scram
3. Flow/flapper scram
4. Spare

The first active trip (trip #2) produces a safety channel overpower scram. The second active trip (trip #3) is a upscale trip that enables the low primary flow and flapper open scram circuits to produce an automatic scram when the power level exceeds 150 kW and the primary coolant flowrate drops below the low flow scram setpoint and/or the flapper is not fully closed. Trip #1 and Trip #4 are spares and are not presently used.

The safety channel is calibrated semi-annually but at intervals not to exceed 7½ months.

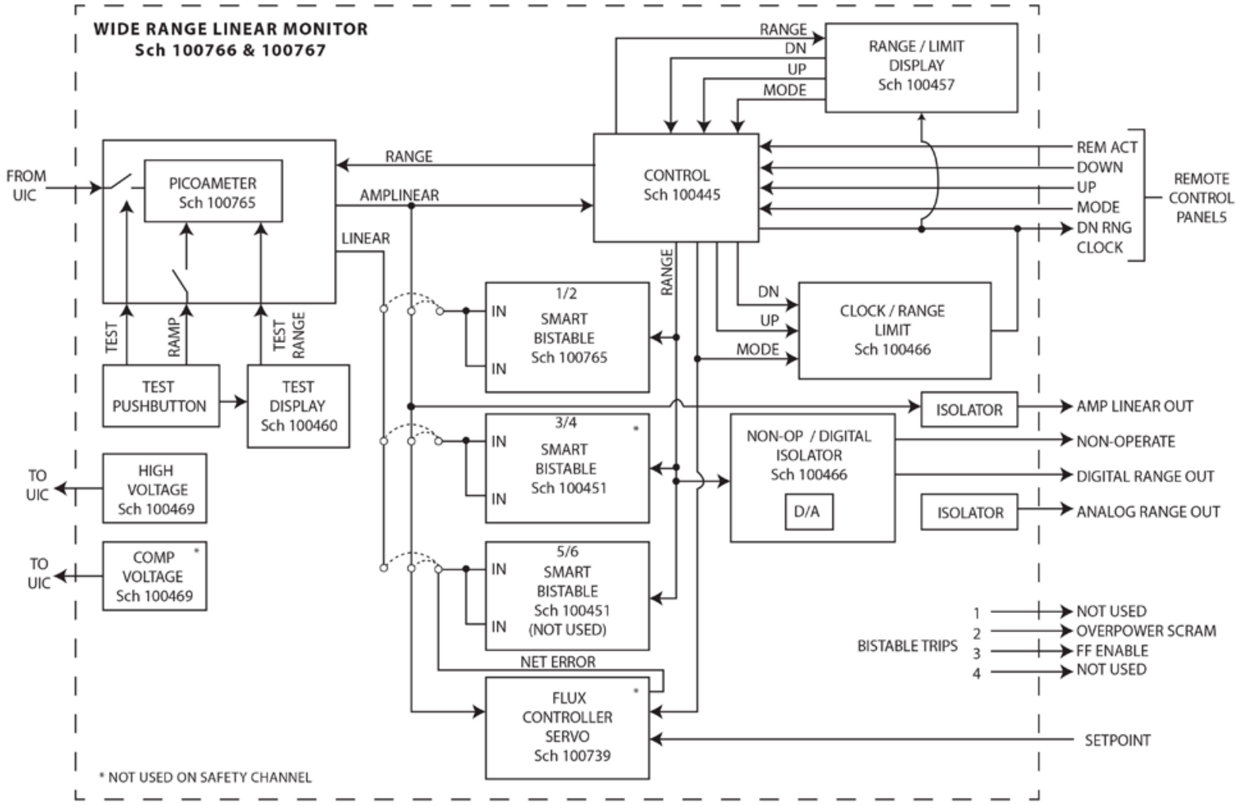


Figure 7-10 – Safety Monitor Channel Block Diagram

7.4.3.5. ^{16}N Channel

The ^{16}N channel is used to indicate the reactor power level as a function of the decay of the ^{16}N radioisotope produced when the coolant passes through the operating core. The channel consists of an ion chamber, a high voltage power supply, and an electrometer. This channel is used as a reference to determine power level while operating at steady-state and when the detectors have to be repositioned due to core changes. The ^{16}N channel has shown that it can be used quite successfully to detect fuel leaks like the one experienced at the BMRC PULSTAR, therefore this channel may provide the first indications of a fuel pin failure.

The detector is a gas filled, gamma sensitive ionization chamber. The detector output is linear with respect to actual reactor power, assuming a constant coolant flow rate through the core, and is not affected by primary coolant temperatures or rod shadowing. The detector is mounted on the 10-inch primary coolant hot leg directly downstream from the reactor core.

The output of the ^{16}N detector is measured and displayed by an electrometer mounted in the control console. Refer to Figure 7-12. The electrometer is set on a current range which displays a current equivalent to reactor power. The electrometer has an auxiliary output which drives a percent reactor power meter. There are no trips associated with the ^{16}N channel.

The ^{16}N channel is calibrated semi-annually but at intervals not to exceed 7½ months. A calorimetric measurement is performed semi-annually but at intervals not to exceed 7½ months to determine the ^{16}N detector current associated with full power operation.

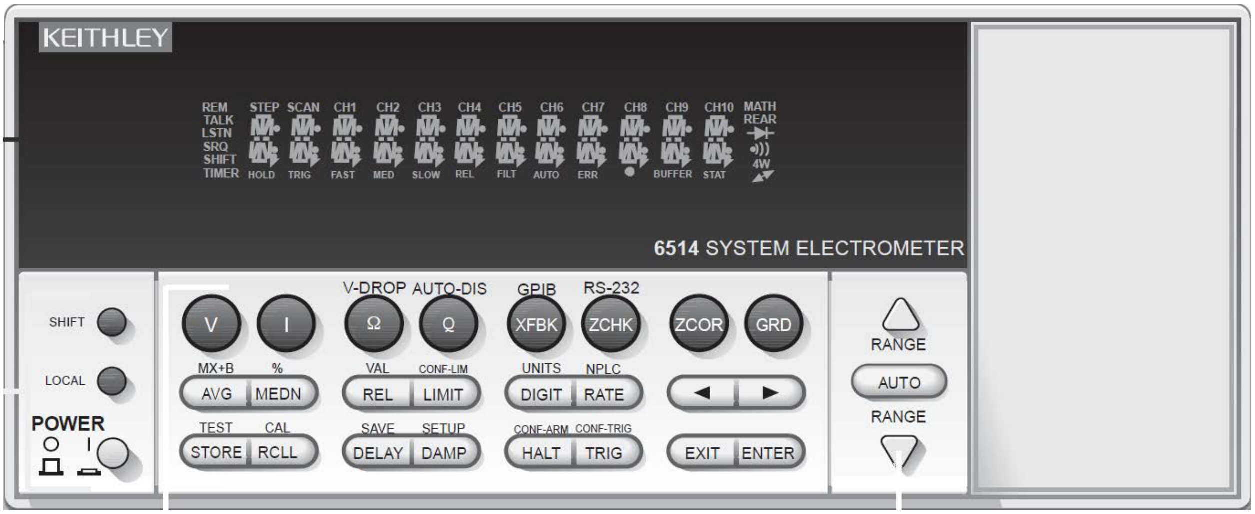


Figure 7-11 – N-16 Channel Front Panel Display

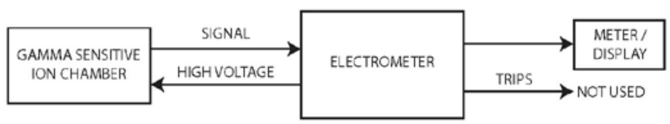


Figure 7-12 – ^{16}N Channel Block Diagram

7.4.4. Non-Nuclear Instrumentation

7.4.4.1. Flow Measuring Channel

The flow of the primary coolant is determined by measuring the pressure drop across a calibrated annubar. The channel consists of an annubar, a differential pressure transmitter, and two display meters, one integral to the transmitter and the other mounted on the control console. A permanently installed inverted U-tube manometer is mounted beside the flow transmitter for calibration purposes.

The flow transmitter produces a 4-20 mA signal that is displayed on the two meters which are calibrated to read out in gallons per minute. When the reactor is operating above 150 kW, a relay mounted in the console will change state generating a low primary flow scram if the coolant rate drops below the safety system setpoint (SSS) of 950 gpm. Loss of transmitter signal will produce the same results. Should the flow rate increase above 1050 gpm or if the transmitter fails to a high current condition, a low primary flow scram will also be generated.

The flow measuring channel is calibrated semi-annually but at intervals not to exceed 7½ months.

7.4.4.2. Flow Monitoring Channel

The flow monitoring channel uses an independent method to detect the loss of flow through the reactor core. This channel consists of a counter-weighted circular metal disk (Flapper) covering an opening in the plenum below the core grid plate, a push-rod, and a micro-switch.

The flapper is held closed when there is sufficient flow down through the core to produce a differential pressure between the plenum and the pool. As long as the flapper is closed, a pushrod actuates a normally open microswitch on the bridge. If the flapper opens for any reason and the reactor power level is greater the 150 kW, a flapper open scram is generated. Since the flapper is a monitoring channel, there are no other status indications on the control console other than the annunciator panel.

The flow monitoring channel is calibrated semi-annually but at intervals not to exceed 7½ months.

7.4.4.3. Pool Level Measuring Channel

The height of the water above the reactor core is continuously measured by a guided wave radar system and displayed on a meter (referenced to negative inches below overflow) on the control console. Overflow level is equivalent to 20’ 6” of water over the top of the core.

The meter has four alarm relays:

1. Abnormal pool level alarm at -1 inch from the overflow level.
2. Abnormal pool level alarm at -12 inches from the overflow level.
3. Security alarm at -24 inches from the overflow level.
4. Scram at -36 inches from the overflow level. Equivalent to 17’ 6” above the top of the core.

The pool level measuring channel is calibrated semi-annually but at intervals not to exceed 7½ months.

7.4.4.4. Pool Level Monitoring Channel

The height of the water above the reactor core is continuously monitored by a float-type device installed in the water. The float is a watertight tube with eighth-inch graduations mounted within a guide attached to the reactor bridge. The float is free to move up and down as the water level varies. The water level is measured by comparing the graduations relative to a fixed index point on the guide. Integrated into the guide tube is a proximity sensor that will generate a scram when the pool water level drops below 30 inches from the overflow (equivalent to 18 feet of water above the core). If the float leaks and fills with water it will sink and cause a low pool water level indication which will generate a low pool level scram. Local audible and visual alarms are also generated.

The pool level monitoring channel is calibrated semi-annually but at intervals not to exceed 7½ months.

7.4.4.5. Temperature Measuring Channel

The temperature measuring channel measures the water temperature at seven locations in the primary and secondary coolant systems. The channel consists of resistance temperature detectors (RTDs) mounted in thermal wells, a power supply, 4-20 mA temperature transmitters, and a recorder mounted in the control console. Temperature is measured in the pool, at the hot and cold leg in the delay tank area, at the heat exchanger primary inlet and outlet, at the heat exchanger secondary inlet and outlet, and at a test transmitter. The cold leg RTDs generate an abnormal pool temperature alarm when 100°F is exceeded while the hot leg RTDs generate an abnormal pool temperature alarm when 110°F is exceeded. The channel will generate a pool temperature scram at a pool temperature of 116°F.

The temperature measuring channel is calibrated semi-annually but at intervals not to exceed 7½ months.

7.4.4.6. Pool Temperature Monitoring Channel

The temperature of the pool water is monitored by a thermal switch suspended in the pool near the core which generates a high pool temperature switch alarm on the control console when 114°F ± 2°F is reached. A normally closed push-button switch is mounted on the console in series with the thermal switch to test the circuit.

The pool temperature monitoring channel is calibrated semi-annually but at intervals not to exceed 7½ months.

7.4.4.7. Primary Coolant Resistivity

The resistivity of the primary coolant is measured by sensors suspended in the pool with a measured value displayed in the control room. A pool water chemistry alarm is generated when the resistivity of the water drops below 1 MΩ·cm. Alternatively, samples of reactor pool water may be analyzed for resistivity using procedures and laboratory equipment.

The resistivity is measured every 14 days, not to exceed 18 days.

7.5. Engineered Safety Features Actuation System

See Section 6 for a detailed description.

7.6. Control Console and Display Instruments

All reactor monitoring and control instrumentation is centralized at the control console in the PULSTAR control room, and switches, gauges, and dials that control most of the equipment associated with the reactor are within easy reach of the operator. The instrumentation located in the console provide the reactor operator with all the information necessary for safe and efficient manipulation of the reactor controls.

Eight lighted pushbutton sets and two single switch assemblies are mounted on the right side of the console to control the following equipment: primary pump, secondary pump, primary demineralizer pump, main HVAC, control room HVAC, confinement fan No.1, confinement fan No.2, PN blower, and the beamtube & thermal column (BT & TC) fan.

The annunciator panels display the alarm status of various sensors and instruments found in the facility. The panels are located on the top portion of the control console. Annunciators are listed in Table 7-3 through Table 7-5.

Table 7-3 – Reactor Annunciators – Scrams with setpoints

Scrams	Setpoint[a]
Linear Channel Over-power Scram	1.90 MW
Safety Channel Over-power Scram	1.90 MW
Low Primary Flow Scram	950 gpm and 1050 gpm and F/F enabled
Flapper Not Closed Scram	Limit switch not actuated and F/F enabled
Low Pool Level Scram	36 inches below overflow
Manual Scram	pushbutton
Pool Temperature Scram	116°F

[a] Setpoints listed are the typical setpoints for the channel. Setpoints may be set at more conservative values.

Table 7-4 – Reactor Annunciators – Alarms with setpoints

Alarm	Setpoint[a]
Source Range Inhibit	2 cps and 9 x 10^4 cps
Low Shutdown Margin	Rods less than 11.3” and LogN Oper Enabled
Linear Channel Over-Power Reverse	1.85 MW
Auto Channel Disengaged	Auto Channel Not Engaged
Abnormal Coolant Temp	Cold Leg 100°F Hot Leg 110°F
High Pool Temp Switch	114°F
Abnormal Pool Level	-1 and -12 inches below overflow
High Sump Level	Sump Level High
Pool Water Chemistry	Resistivity – 1 MΩ
Low Reactor Building Diff Pressure	0.2” Bay vs. Atmosphere for ≥ 4.5 min.
R-63 Fans Not Operating	R-63 Fans Not Operating
Stack Flow Fault Alarm	2 cfm
Low Natural Gas Pressure	Loss of Natural Gas Pressure
Low Oxygen Level	19.5% O_2
Imaging Facility Entry Alarm	Door not closed when high radiation area exists
Positron Facility Entry Alarm	Door not closed
PPV Area Entry Alarm	Door not closed when high radiation area exists
Pneumatic Sample in Reactor	PN Shuttle in reactor
Loading Dock Door Not Closed	Door not closed
Ramp Door Not Closed	Door not closed
Ventilation Room Door Not Closed	Door not closed
Northwest Door Not Closed	Door not closed

[a] Setpoints listed are the typical setpoints for the channel. Setpoints may be set at more conservative values.

Table 7-5 – Reactor Annunciators – Radiation Alarms with setpoints

Alarm	Setpoint[a]
Control Room Radiation Warning	2 mrem/h
Over the Pool Radiation Warning	5 mrem/h
West Wall Radiation Warning	5 mrem/h
Stack Gas Radiation Warning	3000 EC [(1)]
Stack Particulate Radiation Warning	8000 EC [(1)]
Stack Exhaust Radiation Warning	3000 EC [(1)]
Recirculation Radiation Warning	3 DAC [(2)]

[(1)] EC is the effluent concentration in air from 10 CFR Part 20 Appendix B Table 2
[(2)] DAC is the derived air concentration from 10 CFR Part 20 Appendix B Table 1

7.7. Radiation Monitoring Systems

The radiation monitoring system (RMS) is composed of required area and process radiation monitors. RMS radiation monitors are located throughout the reactor building and in or near reactor systems, reactor components, and/or experiments that are radiologically significant. All monitors are capable of measuring background radiation levels up to anticipated accident radiation levels. Section 11 provides additional details on the required and process RMS channels.

7.7.1. Required Radiation Monitoring System Channels

There are five Radiation Monitoring System (RMS) channels required for reactor operation. The required five RMS channels include three area radiation monitors and two airborne effluent monitors. Refer to Table 7-6.

Table 7-6 – Required Radiation Monitoring System Channels

Channel	Radiation Detected	Range		Reading Display
Control Room	Gamma	1.E-02 mrem/h	1.E+06 mrem/h	Control room, optional local display
Over the Pool	Gamma	1.E-02 mrem/h	1.E+06 mrem/h	Control room, optional local display
West Wall	Gamma	1.E-02 mrem/h	1.E+06 mrem/h	Control room, optional local display
Stack Gas	Beta or Gamma	6.E+01 cpm	6.E+06 cpm	Control room, optional local display
Stack Particulate	Beta or Gamma	6.E+01 cpm	6.E+06 cpm	Control room, optional local display

Display of required RMS channels is provided on the radiation recorder in the control room. Readings

[a] Setpoints listed are the typical setpoints for the channel. Setpoints may be set at more conservative values.

may also be displayed locally at the radiation monitor location.

All of the required RMS channels are part of the radiation alarm panel in the control room. Each has two setpoints; warning and alarm. A warning generates an annunciation on the control console to notify the reactor operator of a potential problem. An alarm will automatically initiate evacuation and place the reactor building into confinement. The evacuation system warns personnel about abnormal radiation levels requiring evacuation of the reactor building.

Setpoints for Required RMS channels are described in Section 11.1.4 and listed in Table 7-7.

Table 7-7 – Setpoints for Required Radiation Monitoring System Channels

Channel	Setpoint[a]		Notes
	Warning	Alarm	
Control Room	2 mrem/h	5 mrem/h	2 mrem/h is the public dose rate limit 5 mrem/h is the radiation area dose rate limit 100 mrem/h is the high radiation area dose rate limit Refer to Section 11.1.4 for setpoint bases.
Over the Pool	5 mrem/h	100 mrem/h	
West Wall	5 mrem/h	100 mrem/h	
Stack Gas	3000 EC	3300 EC	EC is the airborne concentration taken from 10 CFR Part 20 Appendix B Table 2 Stack gas EC is for Ar-41. Stack Particulate EC is for Co-60. Refer to Section 11.1.4 for setpoint bases.
Stack Particulate	8000 EC	8500 EC	

The control room, over the pool, and west wall RMS channel setpoints are associated with personnel safety. In addition, the over the pool and west wall radiation monitors provide information on experiments and facility status. The over the pool radiation monitor provides information on experimental samples being withdrawn from the reactor pool and reactor pool level. The west wall radiation monitor provides information on experimental facilities in the reactor building and serves as the accidental criticality monitor required by 10 CFR Part 70.24 for the new fuel storage rack.

The stack gas and stack particulate channel setpoints are associated with public dose. The stack gas and stack particulate channels monitor sampled airborne effluent obtained from the ventilation duct exiting the reactor building prior to entering the reactor exhaust stack. Samples may be obtained in normal or confinement mode of ventilation. Sampled air from the reactor building exhaust is drawn through a fixed paper particulate filter monitored by the stack particulate channel. The sampled air then passed through a gas chamber which houses the stack gas detector. The sampled air is then returned to the exhaust stream downstream of the sample location. In addition, the flow rate through the stack sampling system is measured and provides control room annunciation if it decreases below the minimum flow rate setpoint. The flow rate annunciator notifies the reactor operator of a potential problem with the operability and proper function of the stack sampling system.

[a] Setpoints listed are the typical setpoints for the channel. Setpoints may be set at more conservative values.

7.7.2. Process Radiation Monitoring System Channels

Process RMS channels are used at various locations in or near reactor systems, reactor components, and/or experiments that are radiologically significant. Process RMS channels are not required for reactor operation, but typically are in service. Process RMS channels typically in use are shown in Table 7-8.

Table 7-8 – Process Radiation Monitoring System Channels

Channel	Radiation Detected	Range		Reading Display
Heat Exchanger (HXCH)	Gamma	1.E-02 mrem/h	1.E+06 mrem/h	Control Room, optional local display
Delay Tanks (Delay)	Gamma	1.E-02 mrem/h	1.E+06 mrem/h	Control Room, optional local display
Demineralizer (Demin)	Gamma	1.E-02 mrem/h	1.E+06 mrem/h	Control Room, optional local display
Pneumatic (PN) Sample Return	Gamma	1.E-02 mrem/h	1.E+06 mrem/h	Control Room and local display
Waste Tank Vault (WTV)	Gamma	1.E-02 mrem/h	1.E+06 mrem/h	Control Room, optional local display
Experiment	Gamma or Neutron	1.E-02 mrem/h	1.E+06 mrem/h	Control Room and local display
Recirculation Air Monitor (RCM)	Beta or Gamma	6.E+01 cpm	6.E+06 cpm	Control Room, optional local display
Stack Exhaust	Beta or Gamma	6.E+01 cpm	6.E+06 cpm	Control Room, optional local display
Constant Air Monitor (CAM)	Beta or Gamma	6.E+01 cpm	6.E+06 cpm	Control Room, optional local display

Display of Process RMS channels is provided on the radiation recorder in the control room. For experiments and the PN sample return, local display of readings is provided. For other Process RMS, local display of readings at the radiation monitor location is optional. Experiment area radiation monitors are used as specified in the Radiation Protection (RP) Program, radioactive material authorizations (RMA), radiation work permits (RWP), facility procedures, and for high radiation areas (HRA).

The stack exhaust and constant air monitor (CAM) may be used as substitutes for the stack gas and stack particulate channels, respectively. If so used, annunciation on the control console is provided by the warning setpoint and automatic initiation of the evacuation and confinement system is provided by the alarm setpoint. The setpoints for the stack exhaust channel are based on the same criteria as that for the stack gas channel. The setpoints for the CAM are lower than that for the stack particulate.

Setpoints for Process RMS channels are described in Section 11.1.4 and listed in Table 7-9.

Table 7-9 – Setpoints for Process Radiation Monitoring System Channels

<table>
<tr><th rowspan="2">Channel</th><th colspan="2">Setpoint[a]</th><th rowspan="2">Notes</th></tr>
<tr><th>Warning</th><th>Alarm</th></tr>
<tr><td>Heat Exchanger</td><td>2 mrem/h</td><td>5 mrem/h</td><td rowspan="6">2 mrem/h is the public dose rate limit
5 mrem/h is the radiation area dose rate limit
100 mrem/h is the high radiation area dose rate limit
Refer to Section 11.1.4 for setpoint bases.</td></tr>
<tr><td>Delay Tanks</td><td>5 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>Demineralizer</td><td>5 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>PN Sample Receiver</td><td>100 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>Waste Tank Vault</td><td>0.2 mrem/h</td><td>2 mrem/h</td></tr>
<tr><td>Experiment</td><td>100 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>Recirculation Air Monitor</td><td>3 DAC</td><td>10 DAC</td><td rowspan="3">EC is the airborne concentration is given in 10 CFR Part 20 Appendix B Table 2. DAC is the airborne con centration given in 10 CFR Part 20 Appendix B Table 1.
Stack gas, stack exhaust, and recirculation air EC is for Ar-41.
Stack Particulate EC is for Co-60.
Constant Air Monitor DAC is for Co-60.Refer to Section 11.1.4 for setpoint bases.</td></tr>
<tr><td>Stack Exhaust</td><td>3000 EC</td><td>3300 EC</td></tr>
<tr><td>Constant Air Monitor</td><td>0.3 DAC</td><td>1 DAC</td></tr>
</table>

All of the Process RMS setpoints are associated with personnel safety or public dose. Access controls for entry into high radiation areas (HRA) required by 10 CFR Part 20 may be met using process RMS, e.g. by interlocks on doors, shutters, and exposure to irradiated samples. If so, these channels are displayed in the control room. Should an entry be made into a HRA that is interlocked by a RMS channel setpoint, control room annunciation occurs. The annunciator signal originates from the HRA entry door or gate being opened. HRA access controls are associated with the process RMS for the

[a] Setpoints listed are the typical setpoints for the channel. Setpoints may be set at more conservative values.

delay tanks, demineralizer, and designated experiment areas. The PN sample receiver HRA controls may also be in place as described in Section 11.

Facility status information may also be provided by process RMS. The heat exchanger, demineralizer, and delay tank channels may be used to indicate abnormal radioactivity levels in the primary coolant. Airborne activity is associated with the CAM, RCM, and stack exhaust channels. Experiment area monitors may be used to provide information on sample radioactivity or beam line radiation levels. The waste tank vault channel provides information on high activity in liquid waste. Locations of Process RMS radiation detectors are given in Table 7-10.

Table 7-10 – Locations for Process Radiation Monitoring System Channels

Channel	Location
Control Room	At the Control Room window overlooking the reactor bay
Over the Pool	Approximately at elevation [redacted] to [redacted] over the top of the reactor pool
West Wall	Approximately at elevation [redacted] to [redacted] on the West Wall of the reactor building within 20 feet of the new fuel storage rack
Stack Gas	Inside and off-line, shielded chamber located in the Ventilation Room above the Control Room
Stack Particulate	Inside and off-line, shielded chamber located in the Ventilation Room above the Control Room
Stack Exhaust	Inside, or adjacent to, the stack exhaust and downstream from the confinement filters in the Ventilation Room above the Control Room
CAM	Inside the reactor building approximately at elevation [redacted] feet within 20 feet of the top of the reactor pool
RCM	Near the ventilation system air recirculation duct intake on the East Wall of the reactor building above the Control Room
Heat Exchanger	Near the heat exchanger in the Mechanical Equipment Room.
Demineralizer	Near the primary coolant demineralizer entry point.
Delay Tanks	Near the primary coolant entry point to the first delay tank.
PN Sample Receiver	Near the PN sample return location.
Waste Tank Vault	In or above the waste tank vault.
Experiment	In close proximity to reactor experiments inside the reactor building, associated laboratories, or other areas within the site boundary.

7.7.3. Radiation Monitoring System Maintenance and Operator Response

Facility and experiment status and/or personnel radiation exposure are provided by RMS channels. Operator response to abnormal reading, annunciators, RMS channel warning or alarm, and RMS channel inoperability or malfunction are specified in reactor operating procedures. Should a required RMS channel become inoperable, replacement or substitution with a similar process RMS channel may be made. Notification of the Reactor Health Physicist and designated senior reactor operator are made as needed. Facility procedures are used to meet the technical specification requirements and channel calibrations.

7.8. References

7-1 NUREG 1537 Part 1, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors-Format and Content-Appendix 14.1: Format and Content of Technical Specification for Non-power Reactors*, February 1996, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1537/.

7-2 North Carolina State University PULSTAR Reactor, *R120 Facility License-Appendix A: Technical Specifications*, <date TBD>.

7-3 North Carolina State University PULSTAR Reactor, *Emergency Plan*, February 24, 2017.

7-4 North Carolina State University PULSTAR Reactor, *Reactor Thermal Power and SCRAM Setpoint Determination*, July 22, 2019.

8. ELECTRICAL POWER SYSTEMS

8.1. Normal Electrical Power Systems

The design of the North Carolina State University (NCSU) PULSTAR reactor is such that the reactor can be shut down and safely maintained in a shutdown condition following a complete loss of electrical power. There are no electrical power supplies that are critical for maintaining the facility in a safe shutdown condition, even for extended periods of time. In the event of a prolonged loss of normal electrical power, standard 120VAC outlets powered from the auxiliary generator are available throughout the reactor control room and reactor bay to provide power for various equipment as needed.

The electrical power for Burlington Engineering Laboratories is supplied from the university distribution system. In the event that commercial power is lost, emergency lighting is supplied by backup batteries and selected radiation monitors and reactor instrumentation are supplied by an uninterruptible power supplies (UPS). The PULSTAR reactor is equipped with an auxiliary electric generator to provide power for post shutdown monitoring and ventilation in the event that commercial power is lost.

8.1.1. Main Campus Power

Burlington Engineering Laboratories (BEL) receives its power from the campus 12 kV, three phase, 60 hertz underground electrical service. The voltage is stepped down to 460 volts by an outdoor, pad mounted, oil filled transformer located on the south side of the BEL north wing. The transformer is connected to the switchgear in room 1106A by a weatherproof bus. All feeder circuits in BEL originate from this point. The switchgear provides a 600 ampere feeder to reactor building switchboard panel SB-1 and a 400 ampere feeder to reactor building motor control center M (MCC-M). The switchgear can be used to de-energize electrical power to all reactor equipment in the event of fire or other emergency conditions. Refer to the electrical schematic diagrams shown in Figure 8-1 through Figure 8-3.

8.1.2. Electrical Distribution System

SB-1 located in the mechanical equipment room supplies 460 volt, 3 phase power to PULSTAR reactor equipment. SB-1 is a 600 ampere, three phase, circuit breaker protected switchboard panel. All pumps and fan motors supplied directly by SB-1 are 460 volt, three phase, 60 Hertz induction motors. Refer to Figure 8-2.

The electrical distribution system was updated in 2006 and again in 2013. All design work was performed by licensed engineers, reviewed by state inspectors, and was found to meet or exceed local codes.

MCC-M, located in the third floor ventilation equipment room, supplies 460 volt, 3 phase power to PULSTAR reactor equipment. Refer to Figure 8-2.

The control room distribution panel (CRDP) is located in the control room (2130) and receives power from a 460/120v stepdown transformer fed from SB-1. Refer to Figure 8-3.

The reactor bay lighting panel (RBLP) is located just outside the MER and receives power from SB-1 through a 460/120v stepdown transformer.

The primary piping vault panel (PPVP) is located in the primary piping vault (PPV) on the north wall and receives power from the RBLP. The panel supplies power for the PPV overhead monorail, vault lighting and receptacles.

The reactor bridge panel (RBP) is located on the north wall on the reactor bridge and receives power from the RBLP. The panel supplies power to the bridge experiment hut and for lighting and receptacles at the bridge.

The reactor bay experimental panel (RBEP) is located on the north wall of the MER and receives power from SB-1 through a 460/120V stepdown transformer.

8.2. Emergency Electrical Power Systems

There are no emergency electrical power supplies that are necessary for maintaining the PULSTAR reactor in a safe shutdown condition, even for extended periods of time. The PULSTAR reactor has an auxiliary generator that can temporarily supply electrical service to specific loads. Upon loss of electrical power, the reactor will enter into and remain in a safe shutdown condition even if the auxiliary generator fails to start. Therefore, there is no time constraint for regaining electrical power and no mission times associated with the auxiliary generator, UPS, or emergency lighting.

Commercial 460 volt, 3 phase electrical power for the control room distribution panel and for the two confinement fans is routed through automatic load transfer switches (LTS). Also routed to each LTS is 460 volt, three phase power from the 10 kW auxiliary generator located in Room 1106. The conduits distributing commercial power and generator power are physically separate and routed for maximum practical separation to minimize a single event rendering both sources of power inoperable.

Commercial power is routed through the normally closed set of line contacts of each LTS. Upon loss of commercial power, the generator automatically starts allowing the line contacts in each LTS to open and the generator contacts to close supplying electrical power from the generator to their respective loads. The line and generator contacts are mechanically interlocked so that only one set of contacts can be closed at a time. When commercial power is restored, the control room distribution panel LTS and confinement fan LTSs will switch back to the line contacts after a short delay. It should be noted that only one of the confinement fans can operate at a time due to electrical interlocks of the magnetic starters even though both confinement fan LTSs will switch to the generator contacts.

The auxiliary generator has its own control panel located on the housing of the unit. This panel contains the generator voltmeter with a phase selector switch, ammeters, a run time meter, and a frequency meter. The panel is also equipped with a water temperature gauge, an oil pressure gauge, and a battery charge rate ammeter. The control panel is equipped with a circuit breaker.

The auxiliary generator is connected to the auxiliary generator distribution panel through an automatic transfer switch. This switch is spring loaded to remain in the open position. As the generator comes up to speed and voltage, the generator output voltage works against the spring tension to close the switch and apply power to the auxiliary generator distribution panel. Upon securing the auxiliary generator, the output voltage is removed and the transfer switch automatically opens. Refer to Figure 8-2 for the load distribution.

In addition to automatically starting upon loss of line power, the generator may be started locally at the generator or from CRDP load transfer switch.

The generator also has an emergency latch relay with reset and indicator light. This latch relay opens the starting and/or run circuitry of the generator if low oil pressure or high water temperature trips occur. In the event that the natural gas supply is unavailable, a pressure switch mounted in the natural gas line will generate a Low Natural Gas Pressure Alarm in the control room alerting the reactor operator who then can take the appropriate action.

The auxiliary generator distribution panel (AGDP), located adjacent to the auxiliary generator, houses the feeder breakers that supply auxiliary power to the two confinement fans and the control room distribution panel. The auxiliary generator is rated at 10 kW (steady-state). The loads that are supplied from the auxiliary generator are the control room distribution panel which is rated at 30 amps at 120 VAC and the confinement fans which are each rated at 1.7 amps at 460 VAC. These loads are well within the capacity of the generator.

To assure that an auxiliary power source is available and capable of supplying power to the confinement fans, the technical specification associated with the electrical distribution system is:[8-1]

4.5.b Operability of the confinement system on auxiliary power will be checked monthly but at intervals not to exceed six (6) weeks.(1)

(1)Operation must be verified following modifications or repairs involving load changes to the auxiliary power source.

8.3. References

8-1 North Carolina State University PULSTAR Reactor, *R120 Facility License-Appendix A: Technical Specifications*, <date TBD>.

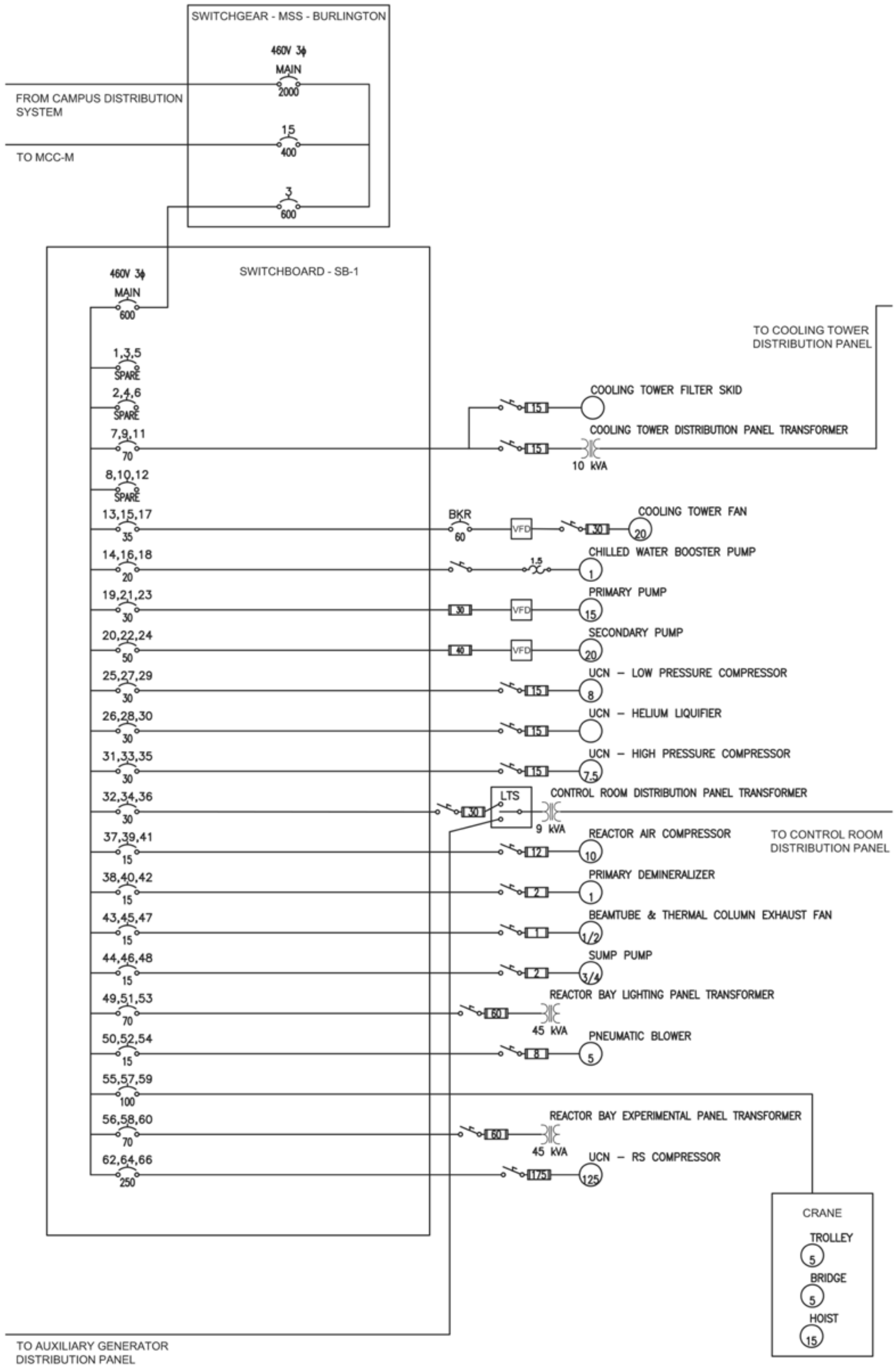


Figure 8-1 – Electrical Distribution System – Switch Board No.1 (SB1)

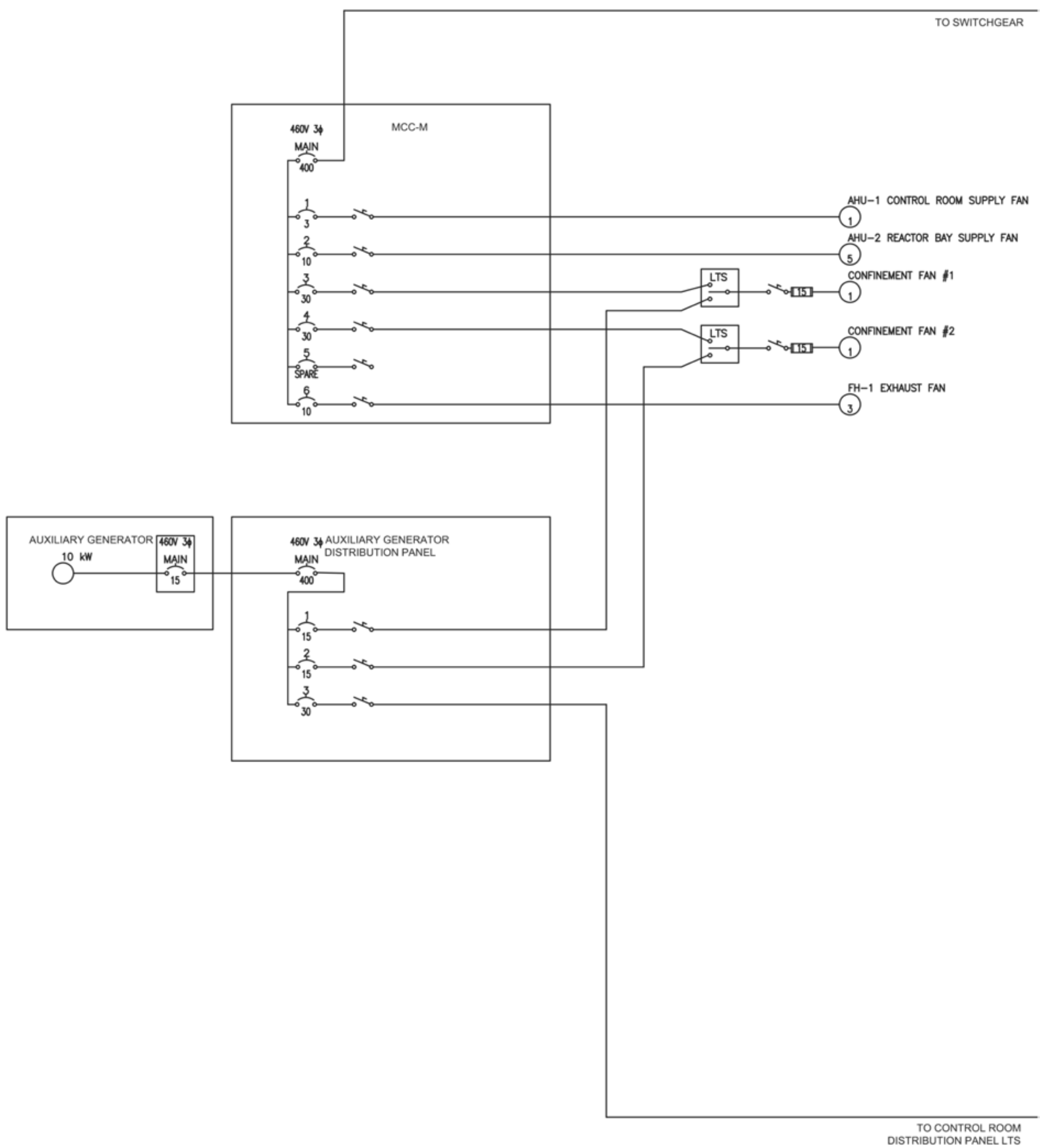


Figure 8-2 – Electrical Distribution System – Motor Control Center M (MCC-M)

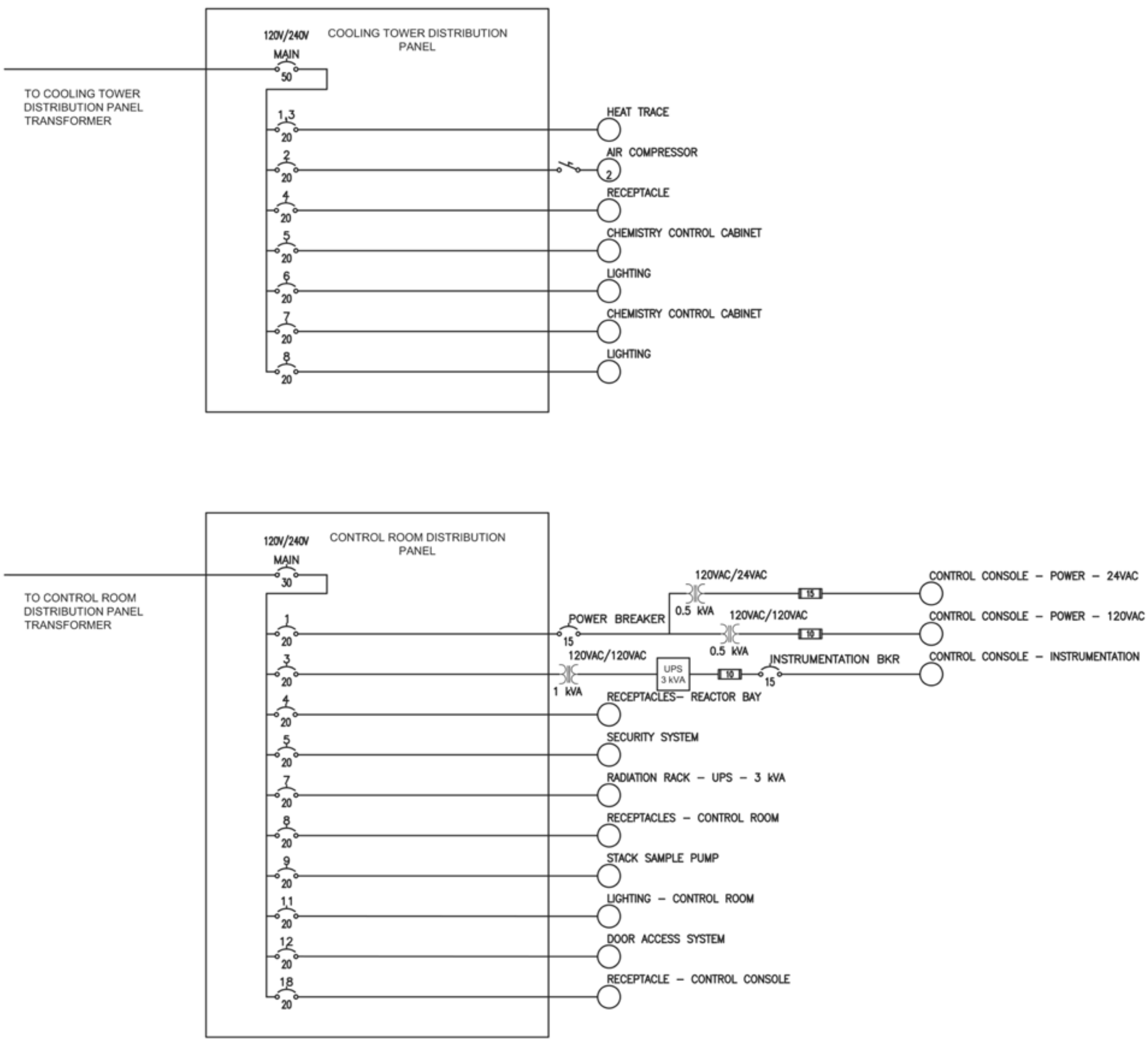


Figure 8-3 – Electrical Distribution System – Control Room Distribution Panel (CRDP)

9. AUXILIARY SYSTEMS

9.1. Heating, Ventilation, and Air Conditioning Systems

Several potential sources of radioactive gas and particulate releases exist at the North Carolina State University (NCSU) PULSTAR reactor. These vary from the production of ^{41}Ar gas in the beamtubes or similar facilities to a failed experiment or a ruptured fuel pin. The handling of these potentially radioactive effluents during both normal and confinement conditions requires an adequate ventilation system capable of minimizing uncontrolled releases to the environment and providing ventilation for personnel and equipment. The size of the ventilation system is based on the magnitude of the release of potential sources. The functional requirements for operation are discussed in Section 13.

The ventilation system is shown schematically in Figure 6-1. Outside air for the reactor building is supplied at 180 cfm through the intake located on the third floor of the south side of the reactor building. The intake is protected by a steel grating. The air is filtered, heated or cooled as necessary to maintain temperature and humidity control, and distributed throughout the reactor building. The control room receives its air through a separate duct branching from the main supply mixing box. It is separately filtered and conditioned to maintain personnel comfort and electronic equipment stability. The reactor building and control room HVAC systems are controlled via wall mounted thermostats located in the respective zones. The HVAC system was completely updated in 2006 and meets all local and state codes for the occupancy rating of the spaces. Air from the reactor bay is drawn through a pre-filter, and the main filters, monitored for radioactivity, and then discharged to the atmosphere through the 100-foot exhaust stack. There is a separate exhaust duct for the beamtubes and the thermal column utilizing a booster fan and an absolute filter. This air is discharged into the exhaust plenum prior to the pre-filter. The pneumatic transfer system also discharges air to the exhaust plenum using a booster fan which is operated as required.

Normally, all the air discharged from the reactor building passes through the exhaust plenum containing a pre-filter which is 30% efficient, a main filter with an average efficiency of 85%, the exhaust fan, and up the stack at a rate of 1870 cfm. If the confinement mode is initiated, the air being discharged is diverted from the main exhaust fan, which is automatically shut down, to one of the confinement fan filter trains. The air is then discharged at a rate of 600 cfm through a 99.97% HEPA filter and a charcoal adsorber. The two confinement fans are interlocked so only one can be operating at any given time.

There are actually two exhaust stacks. The one that is visible from the outside, discharges air from the Burlington Engineering Laboratories south wing and is 100 feet tall. The reactor building stack is located concentrically inside the original stack to within 10 feet of the top, which prevents backflow.

9.1.1. Reactor Building Heating, Ventilation and Air Conditioning

Section 6 provides a detailed description.

9.1.2. Control Room Heating, Ventilation and Air Conditioning

Section 6 provides a detailed description.

9.1.3. Beamtube and Thermal Column Exhaust Fan

The vent lines from the beamtubes and the thermal column are connected to a header through normally open valves around the perimeter of the lower level of the biological shield. This in turn is connected to a HEPA filter and the suction of the beamtube and thermal column exhaust fan (BT&TC). Just prior to the HEPA filter is a water separator and drain leg with an "S" trap. The BT&TC exhaust fan can be started locally or remotely from the control room. Upon initiation of the confinement system the BT&TC fan turns off automatically.

9.1.4. Pneumatic Blower Ventilation

The pneumatic blower is vented directly to the exhaust plenum located in the ventilation equipment room. Refer to Section 10 for more information on the pneumatic blower system.

9.1.5. Confinement System

Section 6 provides detailed description of the confinement system.

9.2. Handling and Storage of Reactor Fuel

9.2.1. Irradiated Reactor Fuel Handling and Storage

The irradiated fuel can be stored in [redacted] fuel storage pits or the linear storage racks. By design, the fuel storage pits and racks provide physical protection for the stored fuel, protecting the stored assemblies from mechanical damage. The spacing and capacity of these storage pits are designed to have a multiplication factor of less than 0.9 when loaded with new fuel assemblies[9-1]. [redacted] irradiated fuel assemblies can be placed in each fuel storage pit. [redacted], the storage pits are always filled with water to provide cooling by natural convection. The storage pits and storage racks are designed so that fuel can only be inserted into allowed locations. The fuel installed in the storage pits are positioned well below the top the of the pit to prevent accidental critically from a fuel assembly inadvertently dropped on top of the pit. [redacted] irradiated fuel storage locations are provided by [redacted] linear racks mounted along the north and south wall of the pool in a subcritical configuration[9-2]. The rack along the north wall has [redacted] locations and the rack along the south wall has [redacted]. These [redacted] racks are suspended from hangers and are approximately midway between the core and the top of the pool. Refer to Section 6 for a detailed discussion of the fuel storage racks.

Approved facility procedures are in place to assure that irradiated fuel is only stored in authorized locations. Preparation of irradiated fuel for shipment follows the requirements given in the Quality Assurance Program for Radioactive Material Packages (NRC Docket 71-0331). Procedures are in place or are developed, as required, for shipments of irradiated fuel and are approved by the Reactor Safety and Audit Committee along with the campus wide Radiation Safety Committee.

The k_{eff} for the storage pits was measured to be 0.77 when loaded with 4% enriched fuel.[9-1] Six percent fuel is not permitted in the storage pits. The k_{eff} for the linear storage racks was measured to be 0.35 when loaded with 6% enriched fuel assemblies and 0.26 for 4% enriched fuel.

Technical specifications associated with this section:[9-3]

5.3 Fuel, including fueled experiments, shall be stored in a geometrical configuration where k_{eff} is no greater than 0.9 for all conditions of moderation and reflection using light water. In cases where a fuel shipping container is used, the licensed limit for k_{eff} of the container shall apply.

9.2.2. New Reactor Fuel Handling and Storage

Receipt of new fuel meets the requirements given in the Quality Assurance Program for Radioactive Material Packages (NRC Docket 71-0331) and facility procedures for inventory, storage, handling and accountability. Procedures are in place or are developed and approved, as required, by the Reactor Safety and Audit Committee along with the campus wide Radiation Safety Committee for the receipt of new fuel.

New reactor fuel can be stored in the rack mounted on the [redacted]. The rack is designed to store non-bundled fuel pins and has a capacity to store up to [redacted] which is equivalent to [redacted] fuel assemblies and to have a k_{eff} of less than 0.9.[9-3,9-4] The rack is mounted out of the direct line of all beamtubes and is [redacted] so that the fuel will remain dry even if the entire volume of the pool was to flood the reactor bay floor.

See Section 9.2.1, above, for the technical specifications associated with this section.

9.2.3. Fuel Handling Tools

Tools are provided for handling individual fuel assemblies and for manipulating other core components such as reflectors and core grid plate plugs. All fuel handling tools have the capability of being locked and secured when not in use as required by the NRC approved facility Physical Security Plan.

9.3. Fire Protection Systems and Programs

The purpose of the fire protection system at the Burlington Engineering Laboratory (BEL) and PULSTAR reactor is to provide detection and notification capability which will mitigate loss of property and life in the event of a fire. The reactor building and BEL have smoke detectors at various locations along with pull stations. Fire extinguishers are positioned throughout the reactor building and BEL. The fire extinguishers and detection system are regularly inspected by the NCSU Fire and Life Safety Office. The system is zoned and upon the triggering of an alarm, reports automatically to the telecommunication officer at the NCSU Department of Public Safety.

NCSU Fire and Life Safety Office maintains and inspects the fire detection system along with the fire extinguishers and performs periodic safety walkthroughs. NCSU Fire and Life Safety provisions are consistent with similar provisions at NRC licensed non-power reactor facilities that minimize flammable material in the reactor bay. Upon activation of the fire alarm, audible and visible strobe alarms notify occupants of the building. NCSU Fire and Life Safety would respond along with the Raleigh Fire Department and Hazmat Unit. Training and response is addressed the Emergency Plan and Procedures.

Reactor operator response for fire alarms are controlled by abnormal response procedures. For the worst case scenario, a fire would cause a complete loss of electrical power causing the reactor to scram and enter into a safe shutdown condition. Radioactive material inventory outside the reactor pool is significantly less than the fuel inventory stored inside the pool, therefore the consequences of a catastrophic fire would be bounded by the MHA – fuel cladding failure accident as discussed in Section 13.2.6.1.

9.4. Communication Systems

Several methods of communication are available to personnel in the reactor building. Telephones are located in the control room and on the floor level of the reactor bay. An intercom connects the control room with various locations throughout the reactor building and the offices and labs in the BEL. A public address system is available in the control room that can be heard in the MER, PPV, the reactor bay, the ventilation equipment room and the basement laboratories.

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

All activities using byproduct, source, and special nuclear material covered under the reactor license take place within the reactor building. Refer to Section 11 for a detail discussion.

9.6. Cover Gas Control in Closed Primary Coolant Systems

The PULSTAR reactor does not utilize a cover gas in the primary coolant system.

9.7. Other Auxiliary Systems

9.7.1. Reactor Air

Reactor air, shown in Figure 9-1, is supplied by a standard air compressor and moisture separator located in the mechanical equipment room. There are no reactor operations systems that utilize compressed air and it is available mainly for use by experimental facilities.

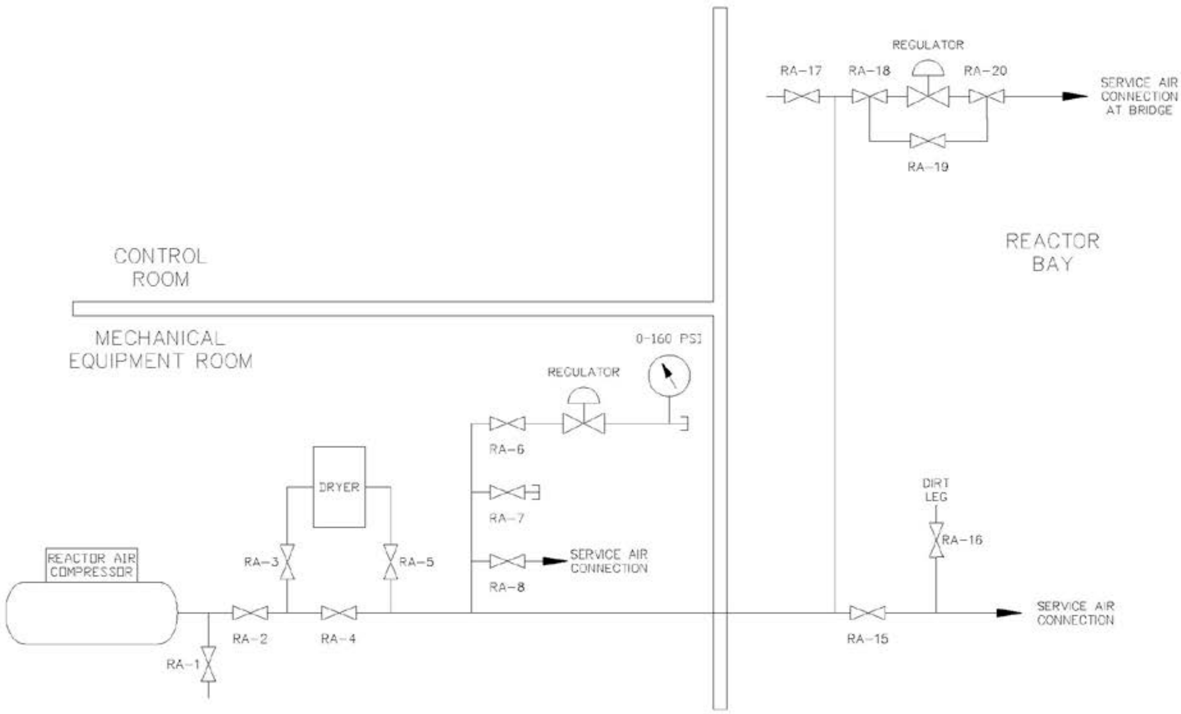


Figure 9-1 – Reactor Air System

9.7.2. Service Water

A source of high purity demineralized water is available at various locations throughout the reactor building. The service water system, shown in Figure 9-2, supplies demineralized organic free water for makeup to the primary coolant system, filling beam tubes, and for other uses requiring purified water.

The major components of the service water system are inlet and outlet filters, a charcoal bed and demineralizer resin beds. The inlet and outlet filters are identical and normally contain 1 micron filters. The charcoal filter column is constructed out of fiberglass and is used to remove organic materials from the water. The two demineralizer columns are also constructed out of fiberglass and are composed of H-OH mixed bed resin. The water is purified by an ion exchange process. Positive impurity ions are exchanged for H+ ions and negative impurity ions are exchanged for OH- ions. The process yields demineralizer effluent pure water.

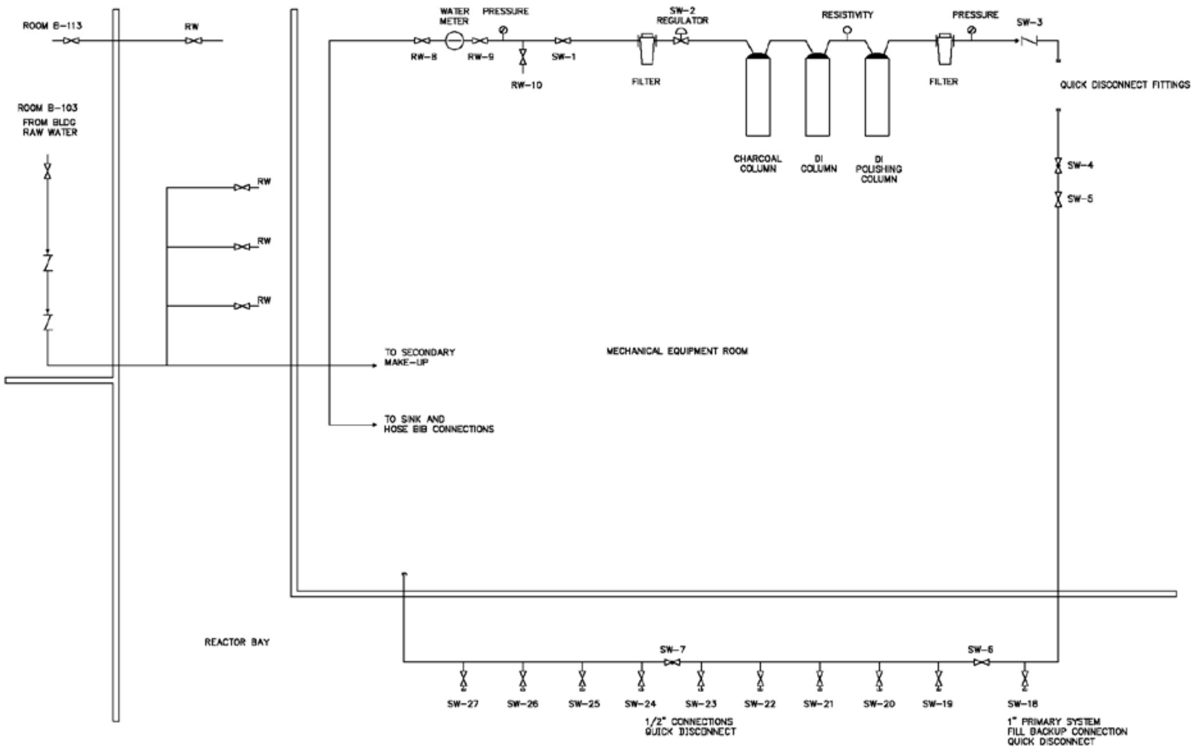


Figure 9-2 – Service and Raw Water Systems

9.7.3. Auxiliary Electrical System

Electric power is available for the control room and the two confinement fans from a 10 kW standby generator. Refer to Section 8 for a detailed description.

9.7.4. Emergency Lighting

Emergency lighting has been installed in various locations throughout the reactor building to assist personnel in the event of loss of electrical power. The lights sense the loss of power and automatically energize. The self-contained batteries are designed to last long enough to allow personnel to safety exit the building.

9.7.5. Reactor Bay Crane

The reactor bay crane with a 10-ton capacity rating is used to handle heavy equipment within the reactor bay. All drive systems are electrically powered and are equipped with positive self-locking devices to prevent motion in the event of power failure. The crane is driven by three separate motors. One motor is used to drive the bridge assembly, another motor to operate the trolley and a third motor to raise and lower the hoist. The bridge motor drives the traveling crane at a maximum speed of 175 fpm. The unit has variable speed and is equipped with travel limit switches. The trolley motor drives the hoist assembly from north to south on the bridge at a maximum speed of 125 fpm. The trolley controls are variable speed and are equipped with travel limit switches. The hoist motor has a maximum speed of 40 fpm and is also variable speed. The crane receives power from SB-1. All the controls for the crane are mounted in a RF controller and also, as a backup controller, on a pushbutton station that is suspended by cable and chain from the trolley.

The reactor bay crane power is normally kept in the off position and only operated by authorized personnel. The crane is not required to mitigate any accident scenario.

9.8. References

9-1 North Carolina State University PULSTAR Reactor, *Startup Test 2.12-Installation of Neutron Source and Fuel in the Reactor Pool*, August 1972, January 1973.

9-2 Nuclear Reactor Program, North Carolina State University, Internal Report, December 2015.

9-3 North Carolina State University PULSTAR Reactor, *R120 Facility License-Appendix A: Technical Specifications*, <date TBD>.

9-4 North Carolina State University PULSTAR Reactor, *Calculation No. NRP-98-01-Criticality Analysis for a 250 Fresh Fuel Pin Storage Rack*, December 1998.

10. EXPERIMENTAL FACILITIES AND UTILIZATION

10.1. Summary Description

The program of operating a research reactor is aimed at the ultimate goal of providing maximum use for the experimentalist. As in the case of reactor operation, planning, design, and operation of experiments requires equal skill, attention and rigor. The emphasis in these areas is dictated by the type of experiments planned by the faculty and university departments.

The North Carolina State University (NCSU) PULSTAR reactor is used for the traditional university activities of teaching and research. In addition, it provides specialized nuclear services to state and federal agencies and industry.

The PULSTAR has a selection of experimental irradiation ports located either internal to the reactor pool, or extending through the reactor bioshield into the reactor bay. See Figure 10-1 for an illustration of the various experimental facilities.

The reactor beamtubes may be utilized for a wide range of experiments harnessing the intense neutron and gamma irradiation fields emanating from the faces of the reactor core. Beamtube facility experimental capabilities include neutron diffraction, neutron radiography, positron beam generation, nuclear material irradiations, and other time of flight or radiative capture experiments. Pneumatic tubes may be used for neutron activation analysis, isotope production, radiochemistry and synthesis, and nuclear physics studies.

There are four rotating exposure ports (REP) which allow for placement of samples at peripheral locations to the reactor core. Samples are encapsulated and then loaded into irradiation containers. Rotation of the loaded irradiation container within the REP provides for uniform irradiation of encapsulated samples. Typical uses of the REP include neutron activation analysis, production of radioisotopes, and radiation damage studies.

Dry exposure ports (DEP) allow for placement of samples at peripheral locations to the reactor core or various pool locations. Samples are loaded into irradiation containers and placed in the DEP. Typically, DEPs are used for testing of various types of detectors.

The thermal column (TC) enclosure houses neutron moderating materials, a bulk irradiation space, and/or shielding. Experimental equipment or items for irradiation may be located in the TC enclosure. The TC enclosure also has a tangential (side) port and vertical access column.

With regard to the reactor proper as an object of experimental investigation, undergraduate and graduate laboratory classes have convenient access for reactor operating parameter studies. Unique or exclusive use type experiments by students and researchers may also be easily accommodated. To illustrate the areas of application, work is ongoing in the areas of neutron diffraction, neutron radiography, positron annihilation spectroscopy, fueled experiment studies, radiation damage in materials, reactor kinetics, neutron activation analysis, nuclear instrumentation testing, isotope production, health physics and dosimetry, and nuclear medicine research.

10.2. Experimental Facilities

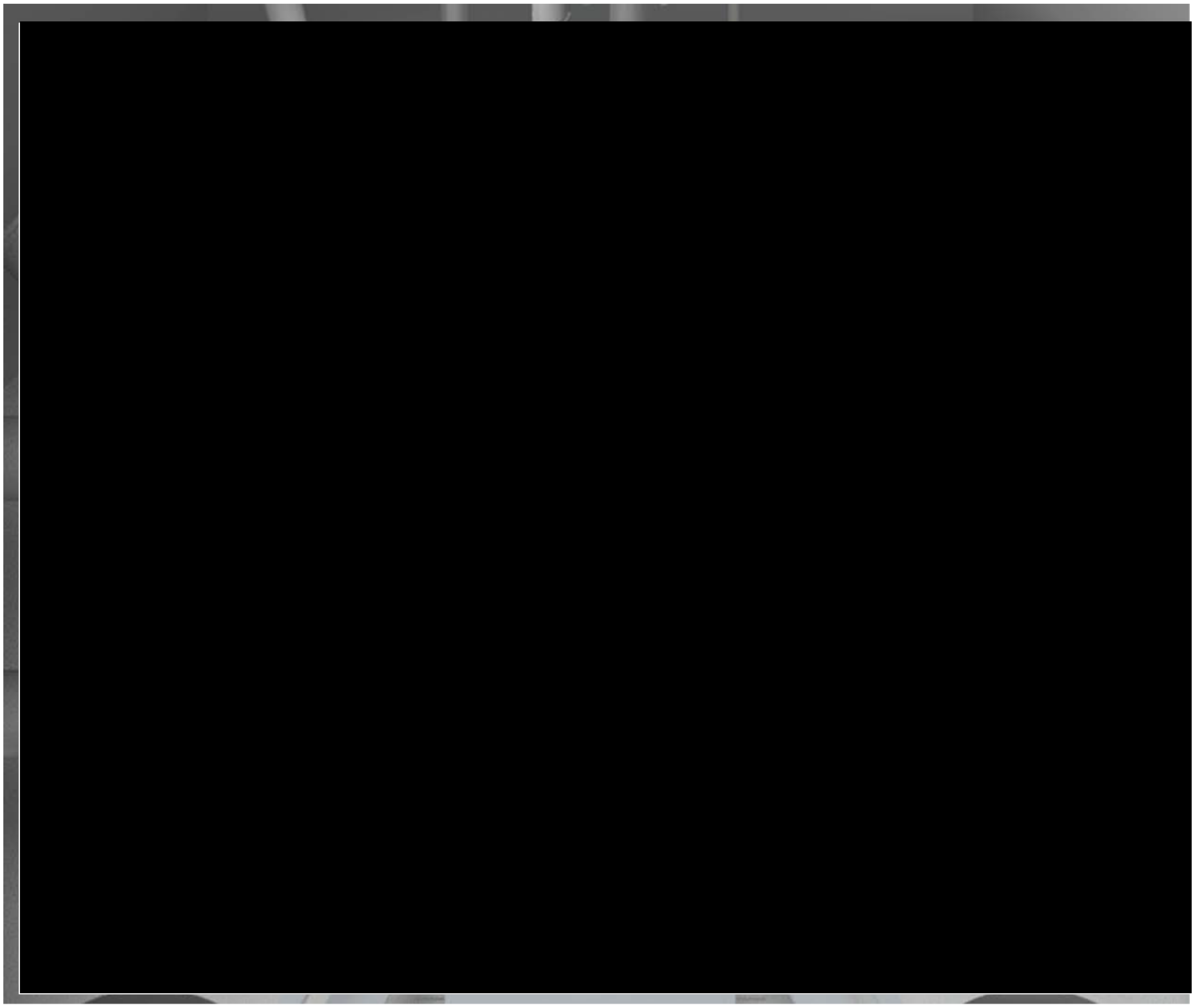


Figure 10-1 – Experimental Facilities

1	6 inch Beamtube (BT#1)
2	6 inch Tangential Through Beamtube (BT#2)
3	8 inch Beamtube (BT#3)
4	6 inch Beamtube (BT#4)
5	6 inch Beamtube (BT#5)
6	12 inch square Beamtube (BT#6)
7	Dry Exposure Ports (DEP)
8	Rotating Exposure Ports (REP)
9	Pneumatic Transfer Tube (PN)

10.2.1. In-core Facilities

At present there are no in-core experimental facilities.

10.2.2. In-reflector Facilities

Several in-pool irradiation locations are available adjacent to the reactor core to take advantage of the high thermal neutron flux region. These facilities may be either vertical, water filled exposure tubes, or the pneumatic transfer system. The thermal neutron flux in these exposure ports is approximately 1×10^{13} n/cm^2·sec.

10.2.2.1. Rotating Exposure Ports (REP)

Four 2.625-inch diameter REPs are provided for irradiation of samples at peripheral locations to the reactor core. Each port is capable of irradiating one container holding encapsulated samples. The port, irradiation container, and encapsulated samples are submersed in the reactor pool.

Standard irradiation containers are maintained by reactor operations but an experimenter may construct an irradiation container specific for an approved experiment if needed. Irradiation containers are identified by a unique number, handled using nylon string, fishing line or aluminum wire, and secured in the reactor pool during use and storage. These containers are only briefly removed from the reactor pool during loading and unloading of samples. Construction of the irradiation containers includes weights at the bottom to ensure placement is maintained during use and storage and holes in the body to allow for filling and draining of reactor pool water.

Samples loaded into the irradiation container are uniquely identified by the experimenter. Loading and unloading of samples is performed by qualified personnel who are knowledgeable in radiation protection and physical reactor controls. All sample materials, capsules, and associated items (markings, string, etc.) to be irradiated must meet approved experimental protocol requirements.

10.2.2.2. Dry Exposure Ports (DEP)

Dry Exposure Ports (DEP) are generally constructed of semi-rigid, curved aluminum or plastic tubing with a water tight end cap. During use the water tight end of the DEP is secured in position in the reactor pool and the open end is secured above the pool surface. Radiation streaming from the reactor core is prevented by virtue of the DEP tube curvature. Detectors or encapsulated samples are lowered inside the DEP to the desired position for the experiment or measurement. Radiation surveys designated by the reactor health physicist are performed as necessary while the DEP is in use. All material and associated items to be irradiated must meet approved experimental protocol requirements. All irradiated materials are surveyed upon removal from the pool.

10.2.3. Pneumatic Transfer System (PN)

A 2-inch pneumatic tube is provided for the rapid transfer of samples to and from the reactor. This tube is used for the production of isotopes and for neutron activation analysis in a wide variety of engineering and scientific areas. The routing of the piping and blower for the pneumatic system has been chosen such that siphoning of the pool water and subsequent uncovering of the core cannot occur. The pneumatic transfer system can utilize either air or nitrogen as the transfer gas. To minimize the production of ^{41}Ar, nitrogen gas is preferred. The pneumatic transfer system, shown in Figure 10-2, has one loading/unloading terminus located in an associated laboratory.

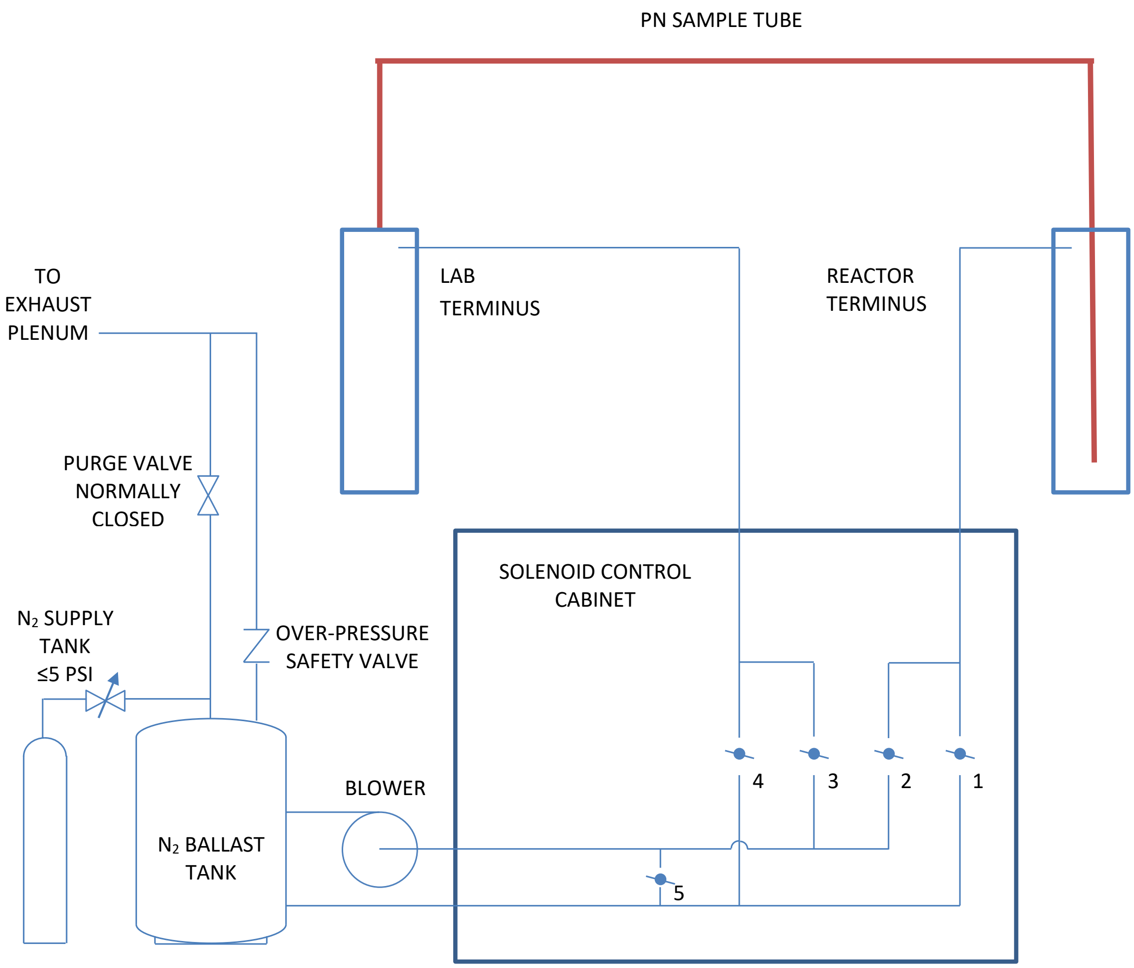


Figure 10-2 – Pneumatic Transfer System Schematic

10.2.4. Beamtubes

The beamtubes, shown in Figure 10-1, are provided to extract neutrons from the core for use in diffraction studies, spectroscopy, radiography, and time-of-flight measurements. The basic tube assembly consists of an embedded aluminum sleeve, a concentric closed-end aluminum chamber, and a set of interior shielding plugs of canned borated barites concrete and lead. The tubes can also be used as dry irradiation chambers for small samples in radiation effect studies. Samples can thus be placed at the core face and easily monitored in a dry environment. At 2 MW, thermal fluxes up to approximately 2×10^{12} n/cm^2·sec are available at the core end of these beamtubes.

Loading and unloading beamtubes is performed with the support of the reactor health physicist. Beamtube plug storage ports are provided in the west wall of the reactor building.

10.2.5. Thermal Column Enclosure

The thermal column (TC) enclosure houses neutron moderating materials, a bulk irradiation space, and/or shielding. Experimental equipment or items for irradiation may be located in the TC Enclosure. The TC enclosure also has a tangential (side) port and vertical access column.

The TC enclosure has two sections. The innermost section located adjacent to the reactor is 4 feet wide x 4 feet high x 5 feet long. The outermost section is 5 feet wide x 5.4 feet high x 4.3 feet long. Access to experimental equipment and items inside the TC enclosure may be included in the experimental design or may be achieved by removal of shielding and internal materials and equipment with appropriate radiation safety precautions using approved facility procedures Radiation detectors may be installed in the TC enclosure as needed for experimental or safety reasons.

10.3. Experimental Review

At the reactor facility, several methods of performing experiments are employed. The experimenter is encouraged to operate his own experiment, with conditions imposed only for reasons of safety. Careful definition of safety limits and a close relationship between the reactor operation staff and the experimenter is necessary. Usually, operations personnel insert and remove experiments from the irradiation facilities. An experimenter, however, can be certified by the operations personnel to use a routine facility such as the pneumatic tube.

All experiments performed in or with use of the reactor must be evaluated and approved by the appropriate reactor staff and meet experiment requirements approved by the University Radiation Safety Committee (RSC) and Reactor Safety and Audit Committee (RSAC). Refer to Section 12 for details on the safety committees. The reactor utilization request (RUR) defines the format, requirements, and conditions for reactor use. Should there be a safety issue related to a reactor experiment or should a new, untried experiment be proposed, the RSC and RSAC must review and approve the requested experiment prior to actual performance.

For a proposed experiment, a member of the operations staff, or in the case of a student, his particular faculty sponsor, is assigned the responsibility of following and assisting with the planning of the experiment.

Administrative procedures and controls are developed which give sufficient definition of, and checks on, the performance of experiments. The reactor staff review ensures that procedures are written by the experimenter that will provide detailed and specific controls of an experiment should that be deemed necessary. No changes are made in experimental procedures without prior review by the reactor staff.

There are many limitations associated with experiments in the reactor facility. Irradiation of explosives, unstable mixtures, and dynamic devices which could compromise the reactor integrity are limited by the technical specifications. Refer to the approved technical specifications for limitation associated with reactivity, chemical and/or physical integrity, radiation hazards, fissionable material, etc. Refer to Section 12 for details on experimental reviews.

Refer to Sections 11 and 13 for details on experiment and experiment failure analysis and along with the report *Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor*.[10-5]

10.3.1. Request for Experimental Approval

The request for reactor experiments is made through the PULSTAR staff who maintain custody of all necessary forms and essential records. An experimenter must furnish in standardized format a description and purpose of the reactor use desired. Consideration must also be given to the irradiation facility required, necessary flux, total exposure, target activation, and core reactivity effects of the experiment. Questions of safety for untried experiments require that thermodynamic unknowns be addressed, target chemical form, flammability and toxicity be discussed and special emergency procedures related to the foregoing be detailed.

On completion of the document and approval by the Manager of Engineering and Operations or his designee, the requested experiment protocol is reviewed and approved by the Radiation Safety Committee and the Reactor Safety and Audit Committee as required. On approval, an experimental protocol number is assigned. Special conditions or constraints may be dictated by the RSC and/or RSAC as deemed necessary.

10.3.2. Review Criteria

The leading criteria to be used in determining the potential hazards and effects associated with an experiment are as follows:

a. Chemical

The following materials if used within the reactor water system (in other than trace quantities) may result in hazardous conditions; therefore, they are avoided or treated with special precautions:

[redacted]

b. Encapsulation

All experiments shall be contained unless it can be shown that the absence of such does not create a hazard. Any system which operated under positive pressure (excluding typical nuclear detectors such as an ion chamber), or may develop pressure due to an accident, and contains or is expected to contain amounts of releasable radioactivity or toxins which would jeopardize personnel safety, must have double encapsulation.

c. Heat Transfer

Systems transferring heat generated by gamma-heating, exothermic reactions, or fission within the experiment, must be designed for thermal stress generated by the reactor operation at 120 percent of nominal power as well as stresses induced by fast startup rates or sudden shutdowns.

d. Mechanical Integrity

The choice of materials must be of the type which are structurally and chemically suitable within the test environment, and resulting levels of radioactivity post-irradiation must be considered in view of handling facilities. Finally, the selection is always sensitive to corrosion problems, either of the material or effects induced in reactor components.

e. Radiation

No experiment, during normal operation, shall result in a direct radiation level in accessible work areas which cannot be reasonably controlled to insure compliance with 10 CFR Part 20. If a credible failure scenario would result in higher levels, special radiation monitoring shall be provided.

f. Instrumentation and Control

Sufficient instrumentation must be included to measure all parameters which may relate to a potential hazard, and automatically control the experiment if needed and practical. This would include such items as status lights for beam shutter positions in semi-permanent facilities.

g. Experiment Operation

All experiments will be operated in accordance with the conditions and limitations specified in the RUR form and in the approved experiment protocol. PULSTAR operations personnel render assistance as required to scientists and engineers in planning and executing experiments on the PULSTAR facility. Close contact between experimenter and operations personnel ensures a workable and safe plan for each experiment.

h. Interference with Reactor

Experiments should be arranged so that they will cause little nuclear or physical interference with operation of the reactor. Installation and removal should normally be possible within a reasonable time, even in the case of experiment failure. It is necessary to make sure that reactivity effects of moving experiments do not exceed technical specifications and/or administrative limits.

The total worth of experiments placed in the reactor core is limited to a maximum 3000 pcm (absolute worth). The worth of each non-secured experiment will not exceed 1000 pcm (absolute value) and the worth of each movable experiment will not exceed 300 pcm or 200 pcm/sec, whichever is more limiting.

i. Manning of Experiments

In most cases the manning of experiments is not required. However, those experiments and operations which do require direct supervision or monitoring shall be specified in the approved experiment protocol and/or RUR.

10.3.3. Technical Specifications for Experiments

10.3.3.1. Limitations of Experiments

To prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure the limitations specified in the technical specifications for experiments are as follows (Fueled experiments must also meet the requirements of Specification 3.8.):[10-1]

<u>Technical Specification 3.7</u>

The reactor shall not be operated unless the following conditions governing experiments are satisfied:

a. All materials to be irradiated shall be either corrosion resistant or encapsulated within a corrosion resistant container to prevent interaction with reactor components or pool water. Corrosive materials, liquids, and gases shall be doubly encapsulated.

b. Irradiation containers to be used in the reactor, in which a static pressure will exist or in which a pressure buildup is predicted, shall be designed and tested for a pressure exceeding the maximum expected by a factor of 2. Pressure buildup inside any container shall be limited to 200 psi.

c. Cooling shall be provided to prevent the surface temperature of an experiment to be irradiated from exceeding the saturation temperature of the reactor pool water.

d. Experimental apparatus, material or equipment to be inserted in the reactor shall be positioned so as to not cause shadowing of the nuclear instrumentation, interference with control rods, or other perturbations which may interfere with safe operation of the reactor.

e. Concerning the material content of experiments, the following will apply:

   i. No experiment will be performed unless the major constituent of the material to be irradiated is known and a reasonable effort has been made to identify trace elements and impurities whose activation may pose the dominant radiological hazard. When a reasonable effort does not give conclusive information, one or more short irradiations of small quantities of material may be performed in order to identify the activated products.

   ii. Attempts will be made to identify and limit the quantities of elements having very large thermal neutron absorption cross sections, in order to quantify reactivity effects.

   iii. Experiments involving material that is considered to be explosive (1), either while contained, or if it leaks from the container, shall be designed to maintain seal integrity even if detonated, to prevent damage to the reactor core or to the control rods or instrumentation and to prevent any change in reactivity.

   iv. Each experiment will be evaluated with respect to radiation induced physical and/or chemical changes in the irradiated material, such as decomposition effects in polymers.

   v. Experiments involving cryogenic materials(1) within the biological shield, flammable(1), or highly toxic materials(1) require specific procedures for handling and shall be limited in quantity and approved as specified in Specification 6.2.3.

f. Credible failure of any experiment shall not result in releases or exposures in excess of the annual limits established in 10 CFR Part 20.

(1) Defined as follows (reference - Handbook of Laboratory Safety - Chemical Rubber Company, 5th Ed., 2000,[10-2] unless otherwise noted):

Toxic: A substance that has the ability to cause damage to living tissue when inhaled, ingested, injected, or absorbed through the skin (Safety in Academic Chemistry Laboratories - The American Chemical Society, 7th Ed., 2003.[10-3])

Flammable: Having a flash point below 73°F and a boiling point below 100°F. The flash point is defined as the minimum temperature at which a liquid forms a vapor above its surface in sufficient concentrations that it may be ignited as determined by appropriate test procedures and apparatus as specified.

Explosive: Any chemical compound, mixture, or device, where the primary or common purpose of which is to function by explosion with substantially simultaneous release of gas and heat, the resultant pressure being capable of destructive effects. The term includes, but is not limited to, dynamite, black powder, pellet powder, initiating explosives, detonators, safety fuses, squibs, detonating cord, igniter cord, and igniters.

Cryogenic: Cryogenic material is material with a normal boiling point below -243°F (reference - National Bureau of Standards Handbook 44.[10-4])

Technical Specification 3.8

Fueled experiments may be performed in experimental facilities of the reactor with the following conditions and limitations:

a. Specification 3.2 pertaining to experiment reactivity worth shall be met.

b. Specifications 3.5 and 3.6 pertaining to operation of the radiation monitoring system and ventilation system shall be met during reactor operation or if moving or handling an irradiated fueled experiment.

c. Specification 3.7 pertaining to limitations on experiments shall be met, with the exception that containers used for vented fueled experiment shall meet specification 3.8.d.iv.1.

d. Fissionable materials used in fueled experiments shall meet the following:

   i. Fissionable material physical form shall be solid, powder, or liquid.

   ii. Fission rate less than or equal to $9.6x10^9$ fissions per second.

   iii. Total number of fissions less than or equal to $1.8x10^{16}$.

   iv. Vented fueled experiments shall meet the following:

      1. Fission gases and halogens may be released. All other materials shall be contained.
      2. Monitoring of the exhaust flow rate. Maximum flow rate shall be less than or equal to three (3) liters per minute (lpm).
      3. Minimum decay time of thirty (30) minutes before being exhausted by the reactor building ventilation system.
      4. Filtration of exhaust for particulates and halogens.(1)
      5. Monitoring of the experiment exhaust gas for radioactivity.
      6. Monitoring for halogens in the stack particulate radiation monitoring channel.

e. Specification 5.3 pertaining to criticality control for fueled experiments in storage shall be met.

f. Specifications 6.2.3 and 6.5 pertaining to the review of experiments shall be met.

(1) Filter removal efficiency shall be certified by the supplier to be 0.95 or greater at flow rates at 3 lpm or less.

Bases

The limitations given in Specification 3.8 ensure that:

a. Fueled experiments performed in experimental facilities at the reactor prevent damage to the reactor or excessive release of radioactive materials in the event of an experiment failure.

b. Radiation doses from accidental or planned releases of airborne activity do not exceed three percent (3%) of the annual limits given in 10 CFR Part 20.

Specification 3.8.a provides reactivity control during irradiation.

Specification 3.8.b provides for radiation monitoring and ventilation system operation, including actuation of the confinement mode of operation should an accidental release occur during irradiation and handling of a fueled experiment.

Specification 3.8.c and 3.8.d.i provide for experimental controls to prevent release of fissionable materials and fission products.

Specification 3.8.d.ii limits radiation dose from the release of fission products to a Total Effective Dose-Equivalent (TEDE) of 0.003 rem in public areas outside the reactor building, a TEDE of 0.15 rem inside the reactor building, and a Total Organ Dose-Equivalent to the thyroid (TODE) of 1.5 rem inside the reactor building.

Specification 3.8.d.iii limits the production of long-lived fission products for safety and security concerns to levels below those given for Category 2 Quantities of Concern in 10 CFR Part 37.

Specification 3.8.d.iv provides controls for planned releases from vented experiments needed to ensure that radiation dose does not exceed three percent of the annual radiation dose limits given in 10 CFR Part 20.

Specification 3.8.e ensures that fueled experiments are stored in sub-critical configurations.

Specification 3.8.f ensures that fueled experiments are reviewed, approved, and documented as required by Specifications 6.2.3 and 6.5.

10.3.3.2. Experiment Review

The technical specifications requirements for experiment reviews are as follows:[10-1]

Technical Specification 6.5

6.5.1 New (untried) Experiments

All new experiments or class of experiments, referred to as “untried” experiments, shall be reviewed and approved by the RSC, the RSAC, the Director of the Nuclear Reactor Program, Manager of Engineering and Operations, and the Reactor Health Physicist, prior to initiation of the experiment.

The review of new experiments shall be based on the limitations prescribed by the facility license and Technical Specifications and other Nuclear Regulatory Commission regulations, as applicable.

6.5.2 Tried Experiments

All proposed experiments are reviewed by the Manager of Engineering and Operations and the Reactor Health Physicist (or their designated alternates). Either of these individuals may deem that the proposed experiment is not adequately covered by the documentation and/or analysis associated with an existing approved experiment and therefore constitutes an untried experiment that will require the approval process detailed under Specification 6.5.1.

If the Manager of Engineering and Operations and the Reactor Health Physicist concur that the experiment is a tried experiment, then the request may be approved.

Substantive changes to previously approved experiments will require the approval process detailed under Specification 6.5.1.

10.4. References

10-1 North Carolina State University PULSTAR Reactor, *R120 Facility License-Appendix A: Technical Specifications*, <date TBD>.

10-2 A. Keith Furr, *Chemical Rubber Company-CRC Handbook of Laboratory Safety*, 5th Ed., 2000.

10-3 American Chemical Society, *Safety in Academic Chemistry Laboratories, Volume 1-Accident Prevention for College and University Students*, 7th Edition, 2003. https://www.acs.org/content/dam/acsorg/about/governance/committees/chemicalsafety/publications/safety-in-academic-chemistry-laboratories-students.pdf.

10-4 National Bureau of Standards, *Handbook 44*, 2017. https://www.nist.gov/sites/default/files/documents/2016/11/10/hb44-2017-web_final.pdf.

10-5 North Carolina State University PULSTAR Reactor, *Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor*, June 2019.

11. RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

11.1. Radiation Protection

Radiation protection (RP) is accomplished by administrative and engineering controls to meet requirements given in applicable regulations, license conditions, and North Carolina State University (NCSU) policies and procedures. These controls are documented in the PULSTAR Reactor Radiation Protection (RP) program.[11-1] The RP Program meets requirements given in 10 CFR Part 20 *Standards for Radiation Protection*[11-2] and is based on guidance given in ANSI/ANS 15.11 *Radiation Protection at Research Reactor Facilities.*[11-3]

This section describes and analyzes radiation protection at the reactor facility. Other sections of the safety analysis report that contain radiation protection related information are noted. Radioactive sources and radiation doses associated with the reactor facility are included. Also included are the programs and other controls used to keep radiation dose to personnel and the public As Low As Reasonably Achievable (ALARA).

11.1.1. Radiation Sources

Radiation sources are used, produced, shipped, released, and disposed. Radiation sources associated with reactor operation are listed on the reactor license,[11-4] while others are listed on the NCSU broad scope license.[11-5] Appropriate controls and limits for these sources are provided in the radiation protection program detailed in the health physics procedures and the NCSU Radiation Safety Manual.[11-6] This includes radiation surveys, labeling, posting, transfer, storage, release and disposal, inventory, leak testing, and handling.

Radiation sources include special nuclear material (SNM) and byproduct materials from fission and activation. Examples include new and irradiated low-enriched reactor fuel, startup source(s), fission detectors, foils and wires, calibration and check sources, experimental samples and equipment, and reactor components. Experimental samples and equipment include byproduct materials and SNM, including fissionable materials used in fueled experiments.

SNM inventory of radioactive materials at the reactor facility is specifically listed on the reactor license and includes reactor fuel, fissionable materials used in fueled experiments, detectors, foils, wires, and startup sources. Under the reactor license, SNM may be possessed but not separated. Reactor fuel is stored in the reactor core or in approved, subcritical locations for all conditions of moderation and reflection outside the reactor core. SNM is stored within the reactor facility inside the controlled access area and protected as specified in the facility security plan. Fuel shipment containers approved by the US Nuclear Regulatory Commission are used for fuel transport.

Pursuant to 10 CFR Part 30 *Rules of General Applicability to Domestic Licensing of Byproduct Material*,[11-7] the facility is licensed to:

1. Possess, use, but not separate, except for byproduct material produced in non-fueled experiments, such byproduct material as may be produced by operation of the facility
2. Receive, possess, and use in operation of the facility any amount of byproduct material in the form of reactor components or otherwise integral to the reactor or reactor experimental facility
3. Receive, possess, and use in operation of the facility byproduct material which is to be irradiated in the reactor within 31 days of receipt.

Pursuant to Section 104c of the Atomic Energy Act and 10 CFR Part 50 *Domestic Licensing of Production and Utilization Facilities*[11-8] the reactor facility may possess, use, and operate the facility in accordance with the facility license. This includes the operation of the reactor and associated radiation and radioactivity.

Radiation sources used for calibration and testing of radiation monitoring equipment used at the reactor facility are included on the NCSU broad scope radioactive materials license issued by the State of North Carolina. These sources include, but are not limited to, ^{241}Am, ^{133}Ba, ^{252}Cf, ^{137}Cs/^{137m}Ba, ^{60}Co, ^{36}Cl, ^{3}H, ^{90}Sr/^{90}Y, ^{99}Tc, ^{239}Pu, and Pu(Be). Pu(Be) is ^{238}Pu. ^{252}Cf sources often contain other isotopes of Cf.

Occasionally other radionuclides are purposely made at the reactor for use as test sources, such as ^{24}Na, ^{41}Ar, ^{38}Cl, and activation foils of various metals (e.g. Mg, Mn, Ni, Cd, Au, In, Sn, and Sb). Activities range from less than 1 µCi to several mCi. Physical forms are usually solid, but liquids or gases are sometimes used.

Typical radionuclides produced by reactor operation include activation products of air, coolant activation, and corrosion. Corrosion activation products are associated with materials used in reactor components in the reactor pool and primary coolant system. Radiation and radioactivity from operations are monitored.

Radionuclides produced by reactor experiments include activation products, and fission products in the case of a fueled experiment. The specific radionuclides produced depend on the materials used in the experiment. Radiation and radioactivity from experiments are monitored. Radionuclides and activity produced by experiments are limited to that specified in an approved experiment authorization. Several factors are used to establish experiment limitations on radioactive sources. These factors include physical form, encapsulation, half-life, radiation type and energy, handling requirements, where the source will be used, contamination levels, radiation dose rates, potential generation of airborne activity, potential accidents, unintended release of material, and waste disposal.

Radioactive sources are located in the reactor pool, reactor experimental facilities (e.g. beam tubes), and designated storage locations within the reactor building. Other radioactive sources are located in State of NC regulated laboratories which are located adjacent to the reactor building in the Burlington Engineering Laboratory (BEL). Controls required by the RP Program and regulations for storage, posting and labeling are followed. Sources and storage areas are shielded if reasonable. The proximity of the reactor facility to the BEL is shown in Section 1.

Sealed radioactive sources are inventoried and leak tested as required by license conditions and regulations. SNM is inventoried as required by license conditions and 10 CFR Part 74 *Material Control and Accounting of Special Nuclear Material*.[11-9] Experiment samples are listed on an experiment log as specified in facility procedures.

Radioactive waste is produced by the operation of the reactor and performance of experiments that use the reactor. Radioactive wastes include irradiated items, airborne effluent, and liquid effluent. Irradiated items include ion exchange resins used to purify the reactor pool and reactor components and experimental materials and equipment that have become activated and/or contaminated. Radionuclides include various byproducts, e.g. activation products and fission products from fueled experiments. Activation products are commonly encountered and fission products are rarely encountered. The activities encountered range from approximately the picoCurie (pCi) to Curie (Ci) level with most being less than the milliCurie (mCi) level at the time of handling by personnel. Items are isolated, shielded, and held for decay if reasonable.

Radioactive waste generation and minimization are included in the review and approval of facility design changes and experiment authorizations.

11.1.1.1. Airborne Radiation Sources

Potential airborne radioactive sources include activation products of air and dust, coolant activation products in the form of a gas or vapor, activated materials from experiments, and fission products in the case of a fueled experiment or fuel cladding failure.

Airborne activity of fission products is not likely due to the robust nature of the fuel cladding, that fuel is submersed in water, and the strict encapsulation requirements for fueled experiment. Evidence of released activity from fuel or experiments would be indicated by the presence of radioactive materials in reactor coolant, primary demineralizer resins, or air samples.

Radionuclides produced may become airborne depending on the physical form and release pathway, e.g. loss of encapsulation of an experiment sample, reactor maintenance activity, or reactor component malfunction.

Airborne activity concentration and dose are affected by the following:

1. Ventilation system operation
2. Maintaining reactor building negative differential pressure
3. Dilution prior to release from the stack
4. Filter retention
5. Atmospheric dispersion

Radionuclides present in air that have been observed from reactor operations are listed below:

1. Noble gas: ^{41}Ar and activation products of stable Kr and Xe
2. Halogen and Vapor: ^{3}H, ^{82}Br
3. Particulate: ^{24}Na, Rb and Cs as decay products of Kr and Xe

The major radionuclide present in air is ^{41}Ar. Other radionuclides listed above have been detected, but in minor amounts having insignificant or negligible radiation dose. Releases from experiments are non-existent to negligible due to sample size and encapsulation.

Fueled experiments may be encapsulated or vented. In the case of vented fueled experiments, fission gases (Kr, Xe) and halogens (I, Br) may be released. These releases are limited, controlled, and monitored as required by Technical Specifications to be low in activity and dose. Analysis of fueled experiment releases is detailed in the report *Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor*.[11-33]

^{41}Argon

^{41}Ar is produced by neutron activation of stable ^{40}Ar. Argon and other components of air are shown below in Table 11-1.[11-10]

^{41}Ar production is associated with air in the reactor pneumatic sample system (PN), experimental beam tubes and thermal column enclosure (BT & TC), and in-pool irradiation facilities. Measures taken to minimize the amount of ^{41}Ar produced in these facilities include purging with nitrogen gas and use of shielding and/or water to reduce air volume.

Air exhausted from the PN transfer system blower and beam tube exhaust blower is emptied directly to the reactor exhaust using separate exhaust ducts. The reactor exhaust duct vents to the R-120 (PULSTAR) inner stack, which is located inside the R-63 outer stack. The ventilation system and stack are described later in this section and in Section 6.

Table 11-1 – Chemical Composition of Air

Element/Compound	Symbol	% by volume
Nitrogen	N_2	78.084
Oxygen	O_2	20.9476
Argon	Ar	0.934
Carbon Dioxide	CO_2	0.0314
Neon	Ne	0.001818
Methane	CH_4	0.0002
Helium	He	0.000524
Krypton	Kr	0.000114
Hydrogen	H_2	0.00005
Xenon	Xe	0.0000087

Leakage into the reactor building is minimized by maintaining and sealing the exhaust ducts. Air leakage into the reactor building directly from the beam tubes is minimized by sealing the beam tube openings and allowing for sufficient decay before opening the beam tubes.

Air is also present in the primary coolant system and released into the primary coolant by off-gassing of experiments placed in the reactor pool, e.g. the rotating exposure ports (REP). Samples irradiated in the REP are encapsulated and filled with water which minimizes off-gassing. ^{41}Ar may off-gas and easily migrates to the pool surface where it enters the reactor building air space.

^{41}Ar production and releases are primarily associated with operation of the pneumatic sample transfer system, which is air operated, and the overall number of MW-h of operation for the reactor. Historically, approximately 80 percent of ^{41}Ar is associated with the PN system with the remainder being associated with the BT&TC blower and off-gassing of in-pool irradiations. ^{41}Ar is also produced by activation of air entrained in the primary coolant system and may escape from the reactor pool to air.

^{41}Ar is detected and measured by radiation monitoring equipment located in the reactor building ventilation ducts. Upon exceeding an alarm setpoint, the ventilation system switches from normal to confinement mode. For ^{41}Ar, the slower exhaust rate in confinement promotes decay within the reactor building prior to release.

The ventilation system was modified in 2006 to allow for air conditioning of the reactor building. Annual ^{41}Ar activity and concentration released from 1 January 2007 to 31 December 2014 are summarized in Table 11-2.[11-11]

The concentration of ^{41}Ar is dependent upon the activity released and the ventilation system status. Exhaust rates vary with the ventilation fans in operation. Even though ^{41}Ar concentration may exceed the unrestricted area concentration limit given in 10 CFR Part 20, the regulatory requirement on public dose is met using atmospheric dispersion as described later in this section and shown in Table 11-14 through Table 11-17 and Figure 11-10.

Table 11-2 – Annual ^{41}Ar Releases

Year	^{41}Ar Activity	^{41}Ar Concentration
	Ci	µCi/ml
2017	8.125	3.9×10^{-8}
2016	13.852	6.7×10^{-8}
2015	3.34	1.6×10^{-8}
2014	7.736	3.7×10^{-8}
2013	12.717	8.7×10^{-8}
2012	16.669	6.3×10^{-8}
2011	17.658	6.6×10^{-8}
2010	10.069	3.8×10^{-8}
2009	8.048	3.0×10^{-8}
2008	5.345	2.6×10^{-8}
2007	3.131	1.1×10^{-7}
Annual Average	**9.702**	**5.26×10^{-8}**

Fueled Experiments

Fueled experiments are either encapsulated or vented. Encapsulated fueled experiments are analyzed for a postulated accidental release due to an experiment encapsulation failure. Vented fueled experiments are analyzed for intentional, routine releases and an accidental release. In both cases, fission gases and halogens are released. Particulates and fissionable materials are not released by experimental controls.

Technical Specifications limit the fission rate to control the fission product inventory available for release. Fission gases and halogens are assumed to be present in the experimental system at saturation activities. Vented fueled experiments assume the release rate of fission gases and halogens is equal to the production rate.

TS also limits the total number of fissions to prevent the buildup of longer lived fission products in the fissionable material target to levels that are safe to handle and well below 10 CFR Part 37 Category 2 limits.

TS limitations on vented experiments additionally include confining the release to the experiment system and ventilation ducts with delay (decay), filtration for particulates and halogens, and dilution prior to release. Vented experiments are monitored for the release flow rate and radioactivity at the experiment location and at the reactor stack. Automatic isolation of the release from the fueled experiment occurs at a designated alarm setpoint.

Fueled experiments are limited by Technical Specification (TS) 3.8 and performed only as needed. TS 1.2.9.3.e, 3.5, and 4.4 also apply to fueled experiments.

Fueled experiments containing ^{235}U or ^{239}Pu are analyzed in detail in the report Safety Analysis in Support of Fueled Experiments and the NCSU PULSTAR Reactor[11-33], which includes an inventory of fission gases and halogens produced and released and the release pathway.

For vented fueled experiments the source term is taken as fission gases and halogens at saturation activity in the free volume of the experimental system. A continuous, controlled release occurs during the vented fueled experiment irradiation time. The experiment exhaust is directly discharged to the reactor building exhaust using the beam tube exhaust duct.

Sections 1 through 9 and 14 in reference 11-33 provide information on vented fueled experiments. Section 2 of reference 11-33 provides information on the saturation activity calculation. Section 5 in reference 11-33 provides information on calculations used for released activity and release rates for vented fueled experiments. Section 8 in reference 11-33 provides information on calculation of beam tube exhaust activity for vented fueled experiments.

Calculation 1 and Table 14-1 and Calculation 2 and Table 14-2 in reference 11-33 give the source term, time integrated exposures, and TEDE results for occupied public areas outside the reactor building from vented fueled experiments using ^{235}U and ^{239}Pu, respectively. Fission rates of $9.6x10^9$ f/s for ^{235}U and $1.4x10^{10}$ f/s for ^{239}Pu were calculated to give a maximum TEDE of 3 mrem to occupied areas outside the reactor building under fumigation conditions for an exposure period of 24 hours.

Calculation 8, Table 14-21 and Table 14-22 in reference 11-33 give the beam tube exhaust duct activity from vented fueled experiments at a fission rate of 9.6×10^9 f/s for vented fueled experiments using ^{235}U. Calculation 8, Table 14-23 and Table 14-24 in reference 11-33 give the beam tube exhaust duct activity from vented fueled experiments at a fission rate of 9.6×10^9 f/s for vented fueled experiments using ^{239}Pu. The total activity in the beam tube exhaust duct for a vented experiment was calculated as 274 µCi for ^{235}U and 265 µCi for ^{239}Pu. External dose rates for occupied areas (3 m or greater) from the beam tube exhaust duct are $7.7x10^{-6}$ rem/h or less for ^{235}U and $7.3x10^{-6}$ rem/h or less for ^{239}Pu. The total dose for 520 h exposure is $4.0x10^{-3}$ rem or less for ^{235}U and for ^{239}Pu.

Activity in the beam tube exhaust duct is diluted by the normal reactor building exhaust and discharged from the reactor stack. Calculated doses to members of the public outside the reactor building do not exceed 3 mrem for vented fueled experiments, or 3 percent of the annual limit and below the dose constraint of 10 mrem at the fission rate of 9.6×10^9 f/s.

Calculation 9 in reference 11-33 is used to define fueled experiments for samples containing uranium. For experiments containing uranium, a fission rate of $1.9x10^6$ f/s has a TEDE of $1.0x10^{-3}$ rem or less to personnel within the reactor building and a TEDE of $1.0x10^{-5}$ rem or less to members of the public outside the reactor building. Experiments with uranium equal to or greater than $1.9x10^6$ f/s or $1.6x10^{11}$ fissions are therefore defined as a fueled experiment. The total number of fissions is $1.6x10^{11}$ based on an accidental 24 hour release that may occur at any time during an irradiation time of 520 hours. This calculation was used in TS 1.2.9.e to define fueled experiments.

Calculation 10 in reference 11-33 is used to calculate radiation monitoring system alarm setpoints.

Setpoints are based on released activity that exceed TS 3.8 limits for vented fueled experiments and fueled experiment accidents. Monitor response is based on the release concentration at the vented fueled experiment exhaust for vented fueled experiments and at the reactor stack for encapsulated or vented fueled experiments. This calculation was used for the monitor alarm setpoints given in TS 3.5.

TS 4.4 gives surveillance requirements for fueled experiments. These requirements are discussed in reference 11-33.

Accidental releases from vented fueled experiments and encapsulated fueled experiments are analyzed in reference 11-33 as described in Section 13 of the FSAR.

Tritium

Tritium, ^{3}H (or T) is produced by the neutron irradiation of deuterium, ^{2}H (or D). ^{2}H is a small component of normal hydrogen. Due to evaporation of primary coolant at the reactor pool surface, ^{3}H in the form of tritiated water vapor (HTO) enters the reactor building air space.

Production of ^{3}H from activation of airborne water vapor is considered negligible due to the low amount of water vapor present in experiments and reactor facilities and the low abundance of ^{2}H. ^{3}H is more likely to be produced as an activation product of the primary coolant.

^{3}H production and release from tertiary fission with diffusion through the fuel clad is considered negligible as compared to activation of deuterium in water. The yield for ^{3}H production by tertiary fission is low and diffusion is low due to the low reactor fuel temperature. ^{3}H is assumed to be in the form of HTO, as this is the more limiting radiation hazard. Some experiments may involve the use of heavy water. To minimize leakage or evaporation, heavy water is contained as an experimental condition. Upon entering the reactor air space the HTO vapor is exhausted from the ventilation system.

The amount of ^{3}H present in air is estimated by assuming an air temperature of 30°C and relative humidity of 50 percent. The reactor building is air conditioned and is typically is below 30°C and 50 percent relative humidity. These assumptions conservatively estimate the water content in air. At 30°C and 50 percent relative humidity, approximately 15 g water per kg of air is present. Using a density of 1 g/ml for water and 1.2×10^{-3} g/ml of room temperature air gives 5.6×10^{4} ml of air per ml of water:[11-10]

$$5.6 \times 10^4\ ml\ of\ air\ per\ ml\ of\ water = \frac{(1000\ g\ of\ air)\left(1\ ml/1.2\times10^{-3}\ g\right)}{(15\ g\ of\ water)(1\ ml/g)} \qquad \textit{Equation 11-1}$$

Measurements of condensate from the reactor building air-conditioning system or a dehumidifier located in the reactor building for ^{3}H have been performed. These measurements indicate an average of 6.5×10^{-5} µCi/ml in condensate and are used to estimate ^{3}H concentration in air:

$$H3\ \mu Ci/ml\ \ in\ air = \frac{6.5\times10^{-5}\ \mu Ci/ml\ of\ water}{5.6\times10^{4}\ ml\ of\ air\ per\ ml\ of\ water} \qquad \textit{Equation 11-2}$$

$$H3\ \mu Ci/ml\ \ in\ air = 1.2\times10^{-9}\ \mu Ci/ml\ in\ air\ at\ 1\ MW \qquad \textit{Equation 11-3}$$

$$H3\ \mu Ci/ml\ \ in\ air\ at\ 2\ MW = 2*1.2\times10^{-9}\ \mu Ci/ml\ in\ air \qquad \textit{Equation 11-4}$$

$$H3\ \mu Ci/ml\ \ in\ air\ at\ 2\ MW = 2.4 \times 10^{-9}\ \mu Ci/ml\ in\ air \qquad \textit{Equation 11-5}$$

Doubling was assumed at 2 MW since the fluence rate at 2 MW is doubled from that at 1 MW, which in turn doubles the production rate. 2×10^{-5} µCi/ml is given in 10 CFR Part 20 Appendix B for an annual dose of 5 rem. The annual committed effective dose-equivalent for 2000 hours exposure is calculated to be less than 1 mrem, [(2.4×10^{-9} /2×10^{-5}) (5000 mrem)]. For the public, the dose is lower yet due to atmospheric dispersion.

^{3}H present is well below the regulatory dose limits for occupationally exposed personnel and members of the public. ^{3}H airborne activity concentration is well below the recommended air sampling limit of 0.01 DAC given in NUREG 1400 *Air Sampling in the Workplace*, [2.4×10^{-9}/2×10^{-5} = 1.2E$\times10^{-4}$]. Given this result, ^{3}H measurements are not routinely performed.[11-12]

^{16}Nitrogen

^{16}Nitrogen is a coolant activation product produced by a (n,p) fast neutron reaction with ^{16}Oxygen. ^{16}N has a half-life of 7.13 seconds and emits high energy gamma photons and beta particles. ^{16}N is produced by activation of water flowing through the reactor core and water present at the periphery of the reactor core. The fast neutron fluence rate is at its highest near the center of the reactor core.

The reactor may be operated in forced convection and natural convection cooling. In forced convection flow, ^{16}N is carried downward through the core region, core plenum, and outlet piping assembly to the primary cooling system. In natural convection cooling, ^{16}N migrates to the pool surface as a result of the warmer primary coolant in the pool rising to the pool surface. Natural convection cooling is more likely to have an airborne release of ^{16}N.

Reactor pool heights of 14 feet, 17 feet, and 20 feet were analyzed for ^{16}N airborne activity from natural convection cooling at a reactor power of 0.1 MW and 0.25 MW. Operating history indicates a normal pool height of greater than 17 feet, infrequent operation in natural convection, and infrequent continuous operation.

The ^{16}N concentration at the reactor core outlet in natural convection cooling is estimated from the historical value of 25 mR/h measured above the primary coolant outlet piping located in the valve pit in forced convection at 1 MW.[11-37] Based on a Microshield 5[11-15] calculation described Section 11.1.5 and Equation 11-76, the concentration in the valve pit piping is 13.9 µCi/ml at 1 MW[11-37]. Adjustments for transit time from the valve pit to the reactor core give a reactor core outlet concentration of 41.5 µCi/ml at 1 MW.[11-37]

Reactor core flow rate in natural convection cooling at the power limit of 0.1 MW and 0.25 MW was estimated from data given in reference 11-13. Flowrate is reported as 34 gpm at 0.1 MW and estimated to be 47.8 gpm at 0.25 MW from reported data.[11-13]

The core water volume is approximately 10.7 gallons[11-13] and core height is 2 feet giving an estimated linear flow rate of 6.36 ft/min [2 feet / (10.7 gal / 34 gpm)] at 0.1 MW and 8.94 ft/min at 0.25 MW.

In natural convection irradiation time is 18.882 seconds at 0.1 MW and 13.431 seconds at 0.25 MW. At 1 MW in forced flow operation, irradiation time is 1.284 seconds.

The water volume in the reactor core is fixed, the concentration [C(rx)] as described below is a function of reactor power and coolant flow rate and is given by:

$$C(rx) = \frac{\sigma\varphi N}{kV}\left[1 - e^{-\lambda t}\right]$$ *Equation 11-6*

where,

$C(rx)$ is the reactor core concentration of ^{16}N in µCi/ml

k is equal to $3.7x10^4$ dps per µCi

σ is the reaction cross section in cm^3

ϕ is the fluence rate in cm^{-2} s^{-1}

λ is the decay constant in s^{-1}

t is irradiation time in s

N is the number of atoms

V Is the primary coolant volume in the reactor core equal to 10.7 gallons, or $4.05x10^4$ ml

V and N are constant during the irradiation time, t.
σ is constant for the nuclear reaction $^{16}O(n,p)^{16}N$.
ϕ is proportional to reactor power.
t varies with the primary coolant flow rate.

^{16}N concentration of 41.5 µCi/ml at 1 MW was adjusted for irradiation time and reactor power to give C(rx) in natural circulation. C(rx) was calculated to be 29.7 µCi/ml at 0.1 MW and 64.6 µCi/ml at 0.25 MW:

$$C(rx) = 29.7\ \mu Ci/ml = [41.5\ \mu Ci/ml]\frac{(0.1\ MW)\left[1 - e^{(-0.0972/s)(18.882\ s)}\right]}{(1\ MW)\left[1 - e^{(-0.0972/s)(1.284\ s)}\right]}$$ *Equation 11-7*

Table 11-3 – ^{16}N Activity for Natural Convection Mode of Operation

MW	Flow Rate gpm	Linear Flow Rate Feet per minute	Irradiation Time seconds	C(rx) Core µCi/ml
0.1	34	6.36	18.882	29.7
0.25	47.8	8.94	13.431	64.6
1			1.284	41.5

^{16}N concentration at the reactor pool surface concentration [C(rps)] was calculated by decay correction of C(rx) for the transit time by assuming the natural convection cooling flow rate continues to the reactor pool surface.

$$C(rps) = C(rx)e^{(-\lambda t)}$$ *Equation 11-8*

where,

t is time to reach the reactor pool surface

Assuming the linear flow rate extends to the reactor pool surface, the time to reach the reactor pool surface is estimated to be 132 seconds, 161 seconds and 189 seconds for reactor pool heights of 14, 17, and 20 feet respectively at 0.1 MW.[11-13] and 94 seconds, 114 seconds, and 134 seconds for reactor pool heights of 14, 17, and 20 feet respectively at 0.25 MW.[11-13]

Concentration in air was assumed to be released completely to the reactor building air space at the natural convection cooling flow rate. The concentration in the reactor building [C(rb)] is calculated as follows:

$$C(rb) = \frac{P}{kV} = C(rps)\frac{R}{kV}$$ ***Equation 11-9***

where,

$C(rb)$ $= \mu Ci/ml$

P $= C(rps)R \text{ in } \mu Ci/s$

$C(rps)$ is the ^{16}N reactor pool surface concentration in µCi/ml from Equation 11-8

R is the reactor pool convection flow rate of 2145 ml/s (or 34 gpm) at 0.1 MW and 3015 ml/s (or 47.8 gpm) at 0.25 MW

k is the effective removal rate constant, $\lambda + {}^{F}/_{V} = 0.0976\ s^{-1}$

V is the reactor building minimum free volume, 2.4×10^9 ml

F is the reactor building exhaust rate $8.83\text{x}10^5$ ml/s (or 1870 cfm)

Annual dose from submersion to airborne ^{16}N in the reactor building was calculated using C(rb) and a reported dose conversion factor (DCF) given in US Department of Energy Report DOE/EH--0070 DE88 014691 External Dose-Rate Conversion Factors for Calculation of Dose to the Public.[11-34] The DCF provides applies to immersion in contaminated air from semi-infinite (hemispherical) clouds for an exposure for an entire calendar year at ground level (2π geometry).

Using the reported DCF for ^{16}N[11-34], the dose inside the reactor building is adjusted for an occupational year of 2000 h and the reactor building dimensions to give an annual occupational dose using equation 11-10.

$$\text{Annual mrem} = (\text{DCF mrem/y per } \mu Ci/m^3)C(rb)\mu Ci/ml(1\text{x}10^6 ml/m^3)(2)(1-e^{-\mu enR})(2000/8760)$$ ***Equation 11-10***

where,

$C(rb)$	Reactor building concentration from Equation 11-9 in μCi/ml
$[1 - e^{-\mu_{en} r}]$	accounts for fraction of energy equilibrium based on dimensions of the reactor building for submersion dose.[11-36] At 6.2 MeV, this term is equal to 0.0187 with r = 944 cm and μ_{en} in air = 2.0×10^{-5} cm^{-1}
2	accounts for the 4π geometry (sphere vs. hemi-sphere)
r	is the radius of the equivalent sphere for the reactor building total volume of 3.5×10^{9} ml, or 944 cm
DCF	Reported DCF is $2.93\text{x}10^{4}$ mrem/y per μCi/ml for ^{16}N[11-34] from a semi-infinite cloud (hemispherical, or 2π geometry, at ground level) for an entire calendar year
2000/8760	ratio of occupational work hours to total hours in a calendar year
$1\text{x}10^{6}$	Converts cubic meters to ml

Table 11-4 – ^{16}N Dose Rate Relative to Pool Level at 0.1 MW

Pool Level ft	Transit Time sec	C(0)exp(-λt)	C(rb) μCi/ml	Annual Dose mrem
20	1.89×10^{2}	3.18×10^{-7}	2.91×10^{-12}	7.5×10^{-4}
17	1.61×10^{2}	4.99×10^{-6}	4.57×10^{-11}	1.2×10^{-2}
14	1.32×10^{2}	7.83×10^{-5}	7.17×10^{-10}	1.8×10^{-1}

Table 11-5 – ^{16}N Dose Rate Relative to Pool Level at 0.25 MW

Pool Level ft	Transit Time sec	C(0)exp(-λt)	C(rb) μCi/ml	Annual Dose mrem
20	1.34×10^{2}	1.38×10^{-4}	$1.78\text{x}10^{-9}$	4.5×10^{-1}
17	1.14×10^{2}	9.79×10^{-4}	1.26×10^{-8}	3.2
14	$9.40\text{x}10^{1}$	6.93×10^{-3}	8.93×10^{-8}	22.3

For example, at 0.1 MW and a pool height of 14 feet, the annual dose calculation is:

$$18\ mrem/y = (2.93\text{x}10^{4}\ mrem/y\ per\ \mu Ci/m^{3})(7.17\text{x}10^{-10} \mu Ci/ml)(1\text{x}10^{6} ml/m^{3})(2)(0.0187)(2000/8760) \qquad \textit{Equation 11-11}$$

Calculated airborne concentrations and annual doses from ^{16}N are low for operation of the reactor with normal reactor pool heights from 17 to 20 feet.

Dose to members of the public is negligible due to atmospheric dispersion and additional decay time before being exhausted from the reactor building. The air exchange time in the reactor building is approximately 45 minutes which provides sufficient time for the ^{16}N to decay to negligible levels. Average atmospheric dispersion at the normal ventilation flow rate provides an additional dose reduction factor in excess of 10,000 as shown in Table 11-13. For continuous operation, the public dose would be increased by a factor of 4.38 [8760 h per year vs 2000 h per year].

In forced convection flow, ^{16}N is carried downward through the core region, core plenum, and outlet piping assembly to the primary cooling system. For ^{16}N removed by the primary coolant system flow, sufficient decay occurs resulting in negligible ^{16}N levels upon return to the reactor pool. ^{16}N produced at the periphery of the reactor core is not drawn into the core and the primary coolant system. ^{16}N from the reactor periphery migrates near the reactor core and slowly rises. The time to reach the reactor pool surface is sufficient for significant decay to occur resulting in a trivial release at the pool surface. Upon reaching the reactor pool surface, any ^{16}N released is mixed with air in the reactor building and exhausted to the environment. The air exchange rate depends on the ventilation system mode of operation; 45 minutes in normal mode and 140 minutes in confinement. Decay, mixing in the reactor air volume, and the air exchange rate reduces ^{16}N activity. Levels are negligible and non-detectable in the reactor building and the environment. From this discussion, it is concluded that airborne ^{16}N activity is insignificant from reactor operation with the reactor in forced convection cooling.

Other Airborne Radionuclides

Activation products of stable Kr and Xe present in air and associated decay products of Rb and Cs are produced in minor amounts due to low abundance in air. These radionuclides have been detected at the reactor facility with a total below one percent of the effluent limits given in 10 CFR Part 20 Appendix B.

Dust includes fibers, hairs, pollen, meteorite particles, skin cells, soil, wind borne aerosols, and other pollutants. Radionuclides associated with dust activation have been reported in the literature (Health Physics Journal Volume 11 Issue 6).[11-16] ^{24}Na and ^{82}Br have been occasionally detected at the reactor facility with a total below one percent of the effluent limits given in 10 CFR Part 20 Appendix B.

Radon and associated decay products are present in low quantities from naturally occurring materials. Radionuclides include ^{220}Rn, ^{214}Pb, ^{214}Bi, ^{212}Pb, and ^{212}Bi. These radionuclides have been detected, but because of the high air exchange rate and intake of fresh air, Rn and the associated decay products are not present in significant amounts. ^{222}Rn concentration is less than 4 pCi/l.

Experiment failures and reactor maintenance may also release airborne activity. Potential airborne release is considered in the review of the experiments or maintenance. Various controls to prevent and/or mitigate airborne releases are specified in radioactive material authorizations, facility procedures, and radiation work permits. Radiation monitoring and design features of the reactor facility are used to limit personnel dose from airborne activity releases.

Release of Airborne Activity

Airborne particulate activity release is unlikely from routine operation due to the low volatility of particulates, low release fractions, corrosion control processes (e.g. water chemistry and purification), sample encapsulation requirements, and/or water submersion, and other experimental controls including radiation monitoring and air sampling. Encapsulation is required for all experiments. Water retains solid material and non-volatile material completely and up to 97 percent of vapors.[11-17]

NUREG 1400 *Air Sampling in the Workplace* provides release fractions that are suitable for air sampling in the work environment. The release fractions are 0 for encapsulated materials, 0.01 for non-volatile powders and liquids, 0.001 for solids, and 1 for gases or volatile materials. These release fractions are considered conservative and a simplified version of those given in 10 CFR Part 30.72 Schedule C. The release fractions given in NUREG 1400 for materials other than gases or volatile materials are applicable to particulates with an aerodynamic median aerosol diameter (AMAD) up to 10 microns. 10 CFR Part 20 assumes 1 micron AMAD particles. Release fractions as given in NUREG 1400 may be applied to experiments and operational work activities to determine the need for air sampling.[11-12]

Reactor Ventilation System

The ventilation system is described in Section 6 and depicted Figure 6-1. Normal ventilation includes 180 cfm of air intake, 5575 cfm of recirculated air, and 1870 cfm of exhausted air. The PN blower and BT&TC exhaust fan discharge into the reactor exhaust plenum prior to both the normal exhaust fan and the confinement fans. Therefore exhaust air flow rates in normal ventilation mode or confinement mode are not affected by the operation of the PN blower or BT&TC fan. In confinement mode, the air intake is closed and the exhaust is routed through particulate and charcoal filters prior to being released to the environment at a rate of 600 cfm.. On loss of offsite power, the auxiliary generator automatically starts and ventilation switches from normal mode to confinement mode. There are two confinement system filter trains, but only one train is used at any given time.

Both the normal and confinement system produce a negative pressure on the reactor building with respect to the atmosphere. Failure to maintain a negative pressure for more than five minutes activates an annunciator in the Control Room. The only release point of airborne effluent with negative pressure maintained is the reactor building (R-120) exhaust stack.

The BEL South Wing (R-63) ventilation exhaust is rated at 12,500 cfm. This portion of the building exhaust is a source of clean process air, which may be used for dilution of the reactor exhaust if it is in operation. The reactor building exhaust stack (R-120) is located concentrically inside of the BEL South Wing (R-63) exhaust stack. Stack diameters are 0.5 m for the R-120 stack and 1.2 m for the R-63 stack. The outer BEL R-63 exhaust stack is 10 feet higher than the inner reactor R-120 exhaust stack (100 ft vs 90 ft), which prevents backflow. Exhaust flow velocities for the two stacks are; 4.5 m/s (4 to 5 m/s) in normal mode and 1.4 m/s in confinement mode for the R-120 exhaust and 5.1 m/s for the R-63 exhaust. Refer to Table 11-6.

Flow for both the R-120 inner stack and outer R-63 stack is considered turbulent based on Reynolds numbers being greater than 4000. Thorough mixing of air exhausted by the R-120 stack into the R-63 stack is assumed to occur based on the exhaust flow velocities and turbulent flow.

The R-63 exhaust uses the entire 30 m outer stack. The R-120 exhaust duct connects to the R-120 inner stack at the third floor, or at approximately 10 m elevation. Only the upper part of the inner R-120 stack is in use; from 40 feet (10m) to 90 feet. The inner R-120 stack is 10 feet below the exhaust point of the 100 feet (30m) high R-63 outer stack.

Reviews on changes to the ventilation system, confinement system, exhaust stack, and surrounding buildings that may affect radiation dose from airborne effluent are performed following 10 CFR Part 50.59, 10 CFR Part 50.54, and the RP Program.

Table 11-6 – Stack Parameters

Parameter	R-63	R-120 Normal	R-120 Confinement	Units
Exhaust Rate	12500	1870	600	cfm
Exhaust Rate	5.899	0.883	0.283	m^3/s
Flow Velocity	5.056	4.497	1.443	m/s
Area	1.167	0.196	0.196	m^2
Radius	0.610	0.250	0.250	m
Circumference	3.828	1.570	1.570	m
Hydraulic diameter	1.914	0.785	0.785	m
Reynolds Number	292694	213625	68548	

Stack Dilution

Nominal exhaust rates and stack dilution factors are shown in Table 11-7.

Table 11-7 – Stack Dilution Factors

Exhaust Fan	Nominal cfm	Stack Dilution Factor with R-63
R-120 Confinement	600	1
R-120 Normal	1870	1
R-63	12,500	N/A
Confinement with R-63	13,100	20
Normal with R-63	14,370	7

The Stack Dilution Factor (SDF) is calculated as follows:

$$SDF = \frac{Nominal\ cfm\ with\ R63\ fan}{Nominal\ cfm\ without\ R63\ fan}$$ *Equation 11-12*

SDF is 1 if the R-63 fan is off. Nominal cfm with the R-63 fan running is the sum of R-120 and R-63. Minimal SDF are listed and are reported at one significant figure to account for flow rate variances.

The R-63 fan is not required for reactor operation, but credit for the SDF may be taken if it is operable. Indication of R-63 fan operation is provided in the Control Room. Building differential pressure (dp) is related to the exhaust rate. Building dp is displayed in the reactor facility control room. Differential pressure measurements are made using surveillance procedures.

Filter Retention

Filter retention[11-18,11-19] (R) for the ventilation system filters and radionuclides are summarized below:

Table 11-8 – Filter Retention

Exhaust Filter	Noble Gas Retention	Particulate Retention (> 0.3 microns)	Iodine Retention
Normal	0	0	0
Confinement HEPA	0	0.9997	0
Confinement Charcoal	0	0	0.9

High efficiency particulate air (HEPA) filters and charcoal beds are used in the confinement mode of ventilation. Filter removal is given by the product (1-R), where R is the retention factor.

Acceptance criteria are 99.97 percent for HEPA tested with a 0.3 micron aerosols and 99 percent for charcoal tested with Freon R-11. Test methods follow ASME N510-1989 *Testing of Nuclear Air Treatment Systems*. A filter retention factor of 0.9 is used for iodine for conservatism.

Atmospheric Dispersion

Atmospheric dispersion calculations are made for the Gaussian Plume Model (GPM) at distances from 30 m to 5000 m for all exposure times for occupied locations. In addition, this analysis considered fumigation (i.e. trapping) conditions caused by an inversion and calm winds for periods up to 24 hours.

Atmospheric dispersion is assessed using methods taken from ANSI/ANS-15.7 *Research Reactor Site Evaluation*[11-20] and NRC 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 Appendix I*[11-21] and NRC Regulatory Guide 1.111 *Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Cooled Reactors*[11-22] to determine the airborne activity concentration at locations from 30 m to 5000 m away from the reactor stack.

The following affect atmospheric dispersion:

1. Variations in wind direction (i.e. wind rose data)
2. Wind speed
3. Weather stability (Classes A through F)

NOTES:

1. Specific details on the above items are provided in this Section and Section 2.
2. The effective stack height and building wake corrections were evaluated but neither is considered to have a significant effect on downwind airborne concentrations.
3. Changes to the ventilation system, confinement system, exhaust stack, and surrounding buildings may affect radiation dose from airborne effluent.

Atmospheric dispersion is defined by parameter [X/Q]. [X/Q] is the ratio of the airborne activity concentration at a given location to the activity exhaust rate. [X/Q] equations presented below are considered applicable to locations that are at or beyond 100 m from the reactor stack. [X\Q] values are used to calculate submersion and inhalation radiation doses.

For distances within 100 m of the reactor stack, external dose from an overhead plume and the reactor stack to ground and elevated locations were calculated using line sources. Conical and cylindrical shaped sources were reviewed and were determined to have narrow dimensions giving a shape similar to a line. The assumption of line geometry is conservative since it concentrates dispersed activity into a line with no credit taken for shielding. Details are provided later in this section.

The general equation for [X/Q] is as follows:

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z u}[e^{-\frac{y^2}{2\sigma_y^2}}][e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \quad \textit{Equation 11-13}$$

where,

$[X/Q]x,y,z$	is the atmospheric dispersion parameter for location (x,y,z), in $[s/m^3]$
X	is in $[Ci/m^3]$
Q	is in $[Ci/s]$
x	is the downwind distance from the stack to receptor in m
y	is the lateral distance from the plume centerline in m
z	is the receptor elevation in meters
σ_y	is the lateral dispersion parameter in meters
σ_z	is the vertical dispersion parameter in meters
h	is the physical stack height in m, or 30 meters
u	is the wind speed in m/s

NOTES:

1. z and h are relative to the ground elevation of 0 m
2. Radioactive decay during transport is neglected.
3. The real (z-h) and a totally reflected plume (z+h) from the ground surface are included in the [X/Q] general equation

Q is calculated from the measured concentration in the reactor exhaust duct at a given R-120 stack exhaust rate, either in confinement or normal ventilation mode of operation, and the SDF. If the R-63 exhaust is not present, SDF has a value of 1. If the R-63 exhaust is present, then the value of Q decreases by the appropriate SDF value; either 7 or 20:

$$Q = \frac{CF}{SDF} \quad \textit{Equation 11-14}$$

where,

C is in $[Ci/m^3]$

F is in $[m^3/s]$

For conservativism, the R-63 exhaust is not assumed to be present in this analysis. If the R-63 exhaust is present, Q decrease which in turn decreases the downwind concentration X by the SDF.

The [X/Q] general Equation 11-13 given above may be modified under certain conditions. These include sector averaging, wind direction distribution, average wind speed, and weather stability class distribution.

ANSI/ANS-15.7 provides modified equations for different time frames. Upon inspection of these equations and noting that there are elevated receptors near the reactor facility, the equations are further modified.

Releases Less Than 2 Hours

For a release of less than 2 hours it is assumed that the weather stability class, wind speed, and wind direction remain constant.

Assumptions made are as follows:

1. The assumed wind speed from ANSI/ANS-15.7 is 1 m/s
2. The most restrictive weather stability class for the given location is used
3. The receptor is assumed to be on the plume centerline, i.e. y = 0 m

With the noted assumptions, Equation 11-13 becomes:

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z u}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}]$$ ***Equation 11-15***

The plume centerline equation above accounts for a receptor location at any elevation relative to the ground level. If the receptor is at ground level, i.e. z = 0 m, then Equation 11-15 becomes:

$$[X/Q]_{x,y,z} = \frac{1}{\pi\sigma_y\sigma_z u}[e^{(-\frac{h^2}{2\sigma_z^2})}]$$ ***Equation 11-16***

Where the real and totally reflected plume are summed.

Releases 2 Hours or Longer

Over time, the direction and speed of the wind varies. An estimate of the average atmospheric dispersion over a period that is very long compared to that over which the mean wind speed and direction are calculated is obtained by:

1. Integrating the [X/Q] general Equation 11-13 with respect to “y”
2. Multiplying by the wind frequency (f) for a given wind speed and direction for a given weather stability class (F) for a given sector
3. Dividing by the width of the sector at the downwind distance of interest

These adjustments are used in the sector averaged model. Sector averaging applies if the wind direction deviates sufficiently across the sector. For a given sector, the arc length (S) of the sector is used as the sector width:

$$S = \frac{2\pi x}{n} \qquad \textit{Equation 11-17}$$

where, n is the number of sectors, and the dimension "y" ranges from -πx/n to +πx/n, i.e. "y" has a total width equal to S.

Sector averaging is considered at downwind distances (x) if $\pi x/n > 2\sigma y$ and for periods greater than 2 hours using the weather parameters given in ANSI/ANS-15.7 and Section 2. If $x\pi/n \geq 2\sigma y$, Equation 11-13 is integrated with respect to y for values of y from ±∞ .

$$\overline{[X/Q]_{x,y,z}} = \frac{nfF}{2\pi x}\int_{-\infty}^{+\infty}\frac{1}{2\pi\sigma_y\sigma_z u}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}][e^{-\frac{y^2}{2\sigma_y^2}}] \qquad \textit{Equation 11-18}$$

where,

f is the frequency fraction for a specified wind direction and wind speed as given in Section 2 and ANSI/ANS-15.7 and F is the weather stability class frequency.

The reported solution gives:

$$\overline{[X/Q]_{x,y,z}} = \sqrt{2/\pi}\,\frac{n}{2\pi x}\,\frac{f}{2\sigma_z u}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \qquad \textit{Equation 11-19}$$

Noting that if n = 16, then (16 / 2π) [2 /π]½ = 2.032 and z = 0 m for ground level receptors, then the sector averaged Equation 11-20 below is in agreement with that given in ANSI/ANS- 15.7.

$$\overline{[X/Q]_{x,y,z}} = \frac{2.032\,fF}{\sigma_z u x}[e^{(-\frac{h^2}{2\sigma_z^2})}] \qquad \textit{Equation 11-20}$$

Where the real and totally reflected plume are summed and u, f, and F are taken from Section 2 and ANSI/ANS-15.7 for the specified release period.

For elevated receptors, Equation 11-19 becomes the following for 16 sectors (n = 16):

$$\overline{[X/Q]_{x,y,z}} = \frac{2.032\,fF}{2\sigma_z u x}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}] \qquad \textit{Equation 11-21}$$

On inspection of h, the reactor facility stack height, where the relationship $\pi x/n > 2\sigma y$ is valid gives the following minimum distances for sector averaging for weather stability classes A through F:

Table 11-9 – Distances for Stability Classes

Stability Class	Minimum Distance (m)
A	3500

B	2500
C	1700
D, E, F	100

From the table above, the following simplifications are made regarding sector averaging:

1. Stability classes D, E, and F were sector averaged at distances greater than 100 m.
2. Stability classes A, B, and C were not sector averaged at any distance greater than 100 m for conservatism.

Releases from 2 to 24 hours

For stability classes A, B, and C it is assumed that the weather stability class frequency (F), wind direction (f), and wind speed (u) remain constant. From ANSI/ANS-15.7, F is set at 1, f is set at 1, and u is 1 m/s. The most restrictive weather stability class was used for a given downwind location (x,y,z). The following equation is used for all receptor elevations (z):

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z u}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}]$$ **_Equation 11-22_**

For stability classes D, E, and F averaging over a sector width is used to represent a meandering plume. From ANSI/ANS-15.7, F is set at 1, f is set at 1, and a constant wind speed (u) is set 1 m/s. The most restrictive weather stability class was used for a given downwind location (x,y,z). The following equation is used for all receptor elevations (z):

$$\overline{[X/Q]_{x,y,z}} = \frac{2.032}{2\sigma_z ux}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}]$$ **_Equation 11-23_**

Releases from 1 to 4 days

For releases from 1 to 4 days, sector averaged [X/Q] values for stability class D and F were calculated using the stability class frequencies and wind speeds listed in ANSI/ANS-15.7 and Equation 11-24.

Table 11-10 – Stability Class Variables

Stability Class	Stability Frequency (F)	Wind Speed (u)	Wind Frequency (f)
D	0.4	3 m/s	1
F	0.6	2 m/s	1

$$\overline{[X/Q]_{x,y,z}} = \sum_{ClassD}^{ClassF} \frac{2.032}{2\sigma_z ux}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}]Ff$$ **_Equation 11-24_**

Releases greater than 4 days

For release periods greater than 4 days [X/Q] values were calculated using the stability class frequencies and wind speeds listed in ANSI/ANS-15.7 and Section 2.

For stability classes A, B, and C, [X/Q] was calculated with y = 0 m (centerline of plume):

$$[X/Q]_{x,y,z} = \frac{1}{2\pi\sigma_y\sigma_z u}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}]Ff \qquad \textbf{Equation 11-25}$$

For stability classes D, E, and F, [X/Q] was calculated using the sector average:

$$\overline{[X/Q]_{x,y,z}} = \sum_{ClassD}^{ClassF} \frac{2.032}{2\sigma_z ux}[e^{[-\frac{(z-h)^2}{2\sigma_z^2}]} + e^{[-\frac{(z+h)^2}{2\sigma_z^2}]}]Ff \qquad \textbf{Equation 11-26}$$

Fumigation (trapping during an inversion)

In fumigation conditions, vertical dispersion is uniform from ground level to the stack height, h. σz is replaced by "h". [X/Q] for fumigation conditions for the plume centerline (y = 0 m) were calculated at a wind speed of 1 m/s for periods up to 24 hours using Equation 11-27 taken from NRC Regulatory Guide 1.145[11-38]:

$$[X/Q] = \frac{1}{\sqrt{2\pi}h\mu\sigma_y} \qquad \textbf{Equation 11-27}$$

where, h is the physical stack height of 30 m and replaces σ_z

Temperature inversions in the lapse rate are associated with fumigation (i.e. trapping) conditions. Inversion frequency is given in SAR Table 2-18 as ranging from 32 to 43 percent in Greensboro, NC. Fumigation period of 24 hours at a wind speed of 1 m/s [11-37] is used in this analysis. Ground based inversions occur by rapid cooling of the ground on cloudless nights with light winds. With warming of the ground during the day by the sun, the inversion ends. It is noted that the reactor building is surrounded by significant paved surface areas and that the reactor typically operates during daytime hours.[11-25]

Inversions may also occur during periods of air stagnation. In North Carolina, approximately 15 air stagnation days per year are reported.[11-33] Air stagnation is defined as a mean wind speed of 4 m/s and a period of 4 days or more.[11-33]. This gives an annual frequency of less than five percent (15/365) for stagnant air.

Maximum fumigation [X/Q] values are used to assess potential radiation dose in occupied locations near the reactor facility for periods up to 24 hours.

Calm winds

Calm winds were assumed to exist for periods up to 24 hours. Calm winds have reported wind speeds less than 0.5 m/s. The straight-line GPM is not applicable and becomes undefined if the wind speed becomes zero. NUREG 1887[11-23] gives a model for calm winds that uses horizontal and vertical turbulence velocities (m/s) rather than normal dispersion parameters. For calm winds default turbulence velocities (σ) of 0.13 m/s are used for the wind, cross wind, and vertical turbulence.

For the plume centerline (y = 0 m), the [X/Q] equation for calm winds for periods up to 24 hours given in NUREG 1887[11-23] is simplified to Equation 11-28:

$$[X/Q] = \frac{1}{(2\pi^{3/2})(x^2+h^2)\sigma} \qquad \textbf{Equation 11-28}$$

A review of weather patterns is given in Section 2. Wind speed data for Jordan Hall at a height of 30 m on the university campus indicates that light winds (i.e. 2 m/s or less) occur approximately three percent of the time. Periods of calm winds (i.e. 0.5 m/s or less) for Jordan Hall would occur less frequently.

[X/Q] Applications

[X/Q] values are used to assess off-site dose from emergency and routine releases of airborne effluent. Release times up to 4 days are associated with emergencies. Release times greater than 4 days are used to demonstrate compliance with 10 CFR Part 20 for routine releases.

Stack Height

Site specific corrections and data for effective stack height, wind speed and wind frequency (wind rose), and building wake were evaluated.

Stack height is affected by momentum effects due to the velocity of the exhausted air, buoyancy effects due to the temperature of the exhausted air, and the exhaust velocity relative to the wind speed.

ANSI/ANS-15.7 guidance for the effective stack height is not applicable since the exhaust velocity is less than 10 m/s in all ventilation modes and the temperature difference of the air exhaust and ambient air is well below 50 °C.

NRC Regulatory Guide 1.111 states that the effective stack height for effluents exhausted from release points more than twice the height of surrounding solid structures is determined as follows:

$$h_e = h + h_{pr} - h_t - c$$ ***Equation 11-29***

where,

h_e is the effective stack height

h is the physical stack height

h_{pr} is the plume rise above the release point due to buoyancy effects and momentum

h_t is the difference in terrain height between the release point and the location of interest which must be greater than 0 meters.

c is the downwash correction factor

It is noted that for the reactor facility:

1. h_e occurs at distances away from the stack release point and is assumed to be reached at a distance equal to 10 times the stack height, or 300 m. For distances less than 300 m the change is the physical stack height, Δh, is taken at 0 m giving h_e = h, or 30 m.
2. h is 30 m (100 feet) and it is noted that the stack slightly exceeds 2 times the height of surrounding structures and buildings
3. h_t is taken as 0 m, since there are no major valleys or hills nearby

4. h_{pr} is based on the following:
    a. momentum effects due to the exhaust velocity
    b. heat emission rate is taken as 0 since the temperature difference of the air exhaust and ambient air is well below 50 degrees C
    c. the equations given in the references from Regulatory Guide 1.111 or ANSI/ANS 15.7
5. c is applicable if the exit velocity, V, is less than 1.5 times the horizontal wind speed, μ, and is calculated using the equation given in Regulatory Guide 1.111. c is 0 m if V is equal to or exceeds 1.5μ.

From the references given in Regulatory Guide 1.111, plume rise is calculated by the following for neutral and unstable weather conditions (Classes A, B, C, and D):

$$h_{pr} = 1.44d[V/u]^{2/3}[x/d]^{1/3}$$ ***Equation 11-30***

where,

d is the stack diameter

V is the exhaust velocity

u is the horizontal wind speed

c if applicable as indicated below, is subtracted from this equation for h_{pr}

Downwash correction factor, c, is applicable if V is less than 1.5μ, and is given by the following

$$c = 3d[1.5 - V/u]$$ ***Equation 11-31***

Where c is 0 m if V is equal to or exceeds 1.5μ, i.e. if c is not positive.

h_{pr} increases with distance, x, but is recognized to have a maximum value based on the atmospheric mixing height. This value of h_{pr}, as corrected by c if applicable, is compared to the following:

$$h_{pr} = 3d[V/u]$$ ***Equation 11-32***

From the references given for Regulatory Guide 1.111, plume rise is calculated by the following for stable conditions (Classes E and F):

$$h_{pr} = 4[F_m/S]^{1/4}$$ ***Equation 11-33***

$$h_{pr} = 1.4[F_m/u]^{1/3}[S]^{-1/6}$$ *Equation 11-34*

Where F_m is the momentum flux and is equal to:

$$F_m = V^2[d/2]^2 = [Vd]^2/4$$ *Equation 11-35*

S is the stability parameter having values of 8.7×10^{-4} for Class E and 1.75×10^{-3} for Class F weather stability classes.

Plume rise is taken as the smaller of the h_{pr} values calculated for distances at or beyond 300 m.

From the references given in Regulatory Guide 1.111 and the discussion above, the following equation for effective stack height is used:

$$h_e = h + h_{pr}$$ *Equation 11-36*

Where c is accounted for in the equations used for h_{pr} if c is applicable.

From ANSI/ANS 15.7, the effective stack height is given by the following for exhaust with temperatures less than 50 °C:

$$h_e = h + \Delta h$$ *Equation 11-37*

$$\Delta h = d[V/u]^{1.4}$$ *Equation 11-38*

Effective stack heights for routine releases are assessed for weather conditions and wind speeds given in ANSI/ANS-15.7 and Section 2 were calculated as described above.

Effective stack heights calculated exceed the physical stack height, or $h_e > h$, for all ventilation modes and weather stability classes at wind speeds from 1 to 4 m/s. h_e ranged from 30.8 m to 40.7m.

It is observed that all effective stack heights are calculated to exceed the physical stack height, or he > h, for all ventilation modes and weather stability classes at wind speeds from 1 to 4 m/s.

It is noted that stack heights greater than 30 m give lower [X/Q] values. For simplification and conservatism, the actual stack height of 30 m is used to calculate [X/Q], i.e. effective stack height is not used for [X/Q] calculations.

Wind Rose Data

Weather data in the form of a wind rose indicating wind speed and direction for several weather stations within 50 miles of the reactor site are given in Section 2. Section 2 data indicates that the peak wind frequency ranges from 0.12 to 0.17 for any of the 16 sectors analyzed for wind direction.

A wind frequency of 0.15 is used for times greater than 4 days for [X/Q] calculations. For time less than less than or equal to 4 days, a wind frequency of 1 is used in [X/Q] calculations.

Building Wake

The building height relative to the outer stack is approximately 2.5 (100 ft vs. 42 ft gives a ratio of 2.38). Modification of σ_y and σ_z as discussed in ANSI/ANS-15.7 for the building wake increases or has no effect on the parameters used in the above equation; e.g. if σ is 2 m, Σ is > 2.6 m and if σ is 100 m, Σ is 100 m:

$$\Sigma = \sigma^2 + [0.5A/\pi]^{1/2} \qquad \textit{Equation 11-39}$$

Where *A* has a maximum value of 54 m for the BEL.

Larger deviations would decrease the [X/Q] value at a given receptor location.

NRC Regulatory Guide 1.111 states that building wake corrections apply to ground level releases (i.e. stack height of 0 m). ANSI/ANS-15.7 notes that building wake effects shall not be used for exposure times greater than 8 hours.

The stack height is 30 m and both routine and accident releases are analyzed for exposure periods in excess of 8 hours. As a result, no wake correction is made.

Dispersion Parameters

Dispersion parameters σ_y and σ_z were calculated using fitting data from NUREG 1887 *RASCAL 3.0.5: Description of Models and Methods*[11-23] and are within 8% of those given in TID 24190 *Meteorology and Atomic Energy 1968*.[11-24]

The calculated dispersion parameter values were generally lower than those listed in TID 24190[11-24] giving a higher, or more conservative, [X/Q] value. Data for PG stability classes A through G is given in NUREG 1887[11-23] and was used to determine [X/Q] values.

[X/Q] Values

Equations used to calculate [X/Q] for distances (x) from 30 m to 5000 m for different release times are as follows:

1. < 2 hours using Equation 11-15 (GPM)
2. 2 to 24 hours using Equation 11-22 (GPM) and Equation 11-23 (GPM)
3. Up to 24 hours for fumigation using Equation 11-27
4. Up to 24 hours for calm winds using Equation 11-28
5. 1 to 4 days using Equation 11-24 (GPM)
6. Greater than 4 days using Equation 11-25 (GPM) and Equation 11-26 (GPM)

Both the ANSI/ANS-15.7 and Section 2 weather conditions were evaluated using the GPM to determine the limiting case for a given offsite location. For elevated releases the maximum downwind concentration may occur for different PG stability classes at different elevations and downwind distances. Maximum [X/Q] values vary with the location (x,y,z) being considered.

Weather parameters used to calculate [X/Q] for different release times are given in Table 11-11.

GPM Calculations

[X/Q] values were calculated using the physical stack height (h) of 30 m for receptor heights (z) ranging from 0 m to 60 m for each stability class. Occupied areas currently exist at heights up to 30 m above ground level within 5000 m of the reactor stack.. The highest value of [X/Q] at each location (x,y,z) for all stability classes is used for conservatism. Results are shown on the following pages.

1. [X/Q] for periods greater than 4 days is conservatively estimated using the ANSI/ANS-15.7 approximation and realistically estimated using Section 2 data.
2. [X/Q] for releases from 2 to 24 hours is applicable to accidents or abnormalities since the reactor building would undergo more than 10 air exchanges in 24 hours. X/Q for 1 to 4 days may be used for longer accident or abnormal releases.
3. [X/Q] for releases less than 2 hours is most conservative. These may be used in determination of radiation monitor setpoints.

Figures 11-1 through 11-7 graph the GPM equation results from 100 m to 5,000 m from the reactor facility for receptors at elevations from ground level (0 m) to 60 m.

Table 11-11 – Weather Parameters to Calculate [X/Q]

Duration	PG Stability Class	PG Stability Frequency	Wind Frequency, f	Wind (m/s)	Lateral Dimension
< 2 h	A through F	1	1	1	0, centerline
2 h to 24 h	A, B, C	1	1	1	0, centerline
2 h to 24 h	D, E, F	1	1	1	Sector Averaged
Up to 24 h	Fumigation	1	1	1	0, centerline
Up to 24 h	Calm Wind	1	1	0.5	0, centerline
1 to 4 days	D	0.4	1	3	Sector Averaged
1 to 4 days	F	0.6	1	2	Sector Averaged
>4 days	C	0.333	0.15	3	0, centerline
>4 days	D	0.333	0.15	2	Sector Averaged
>4 days	F	0.333	0.15	2	Sector Averaged
>4 days	A	0.02	0.15	2	0, centerline
>4 days	B	0.04	0.15	2	0, centerline
>4 days	C	0.06	0.15	3	0, centerline

>4 days	D	0.28	0.15	4	Sector Averaged
>4 days	E	0.18	0.15	3	Sector Averaged
>4 days	F	0.42	0.15	2	Sector Averaged

Note: Class A through F applies to GPM calculations.

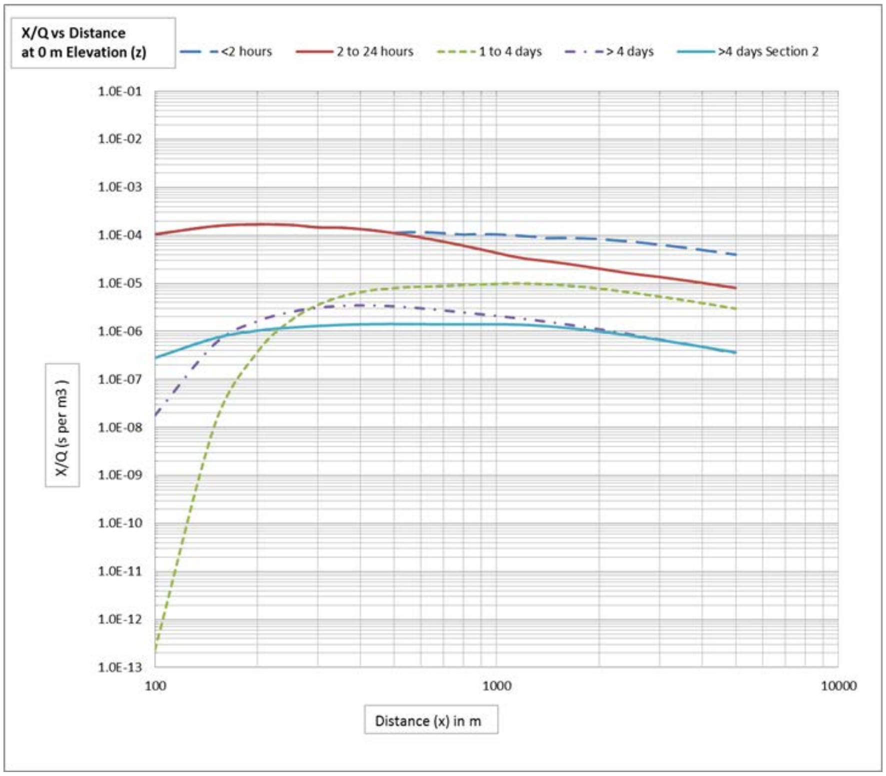


Figure 11-1 – GPM X/Q vs. Distance at 0 m Elevation (z)

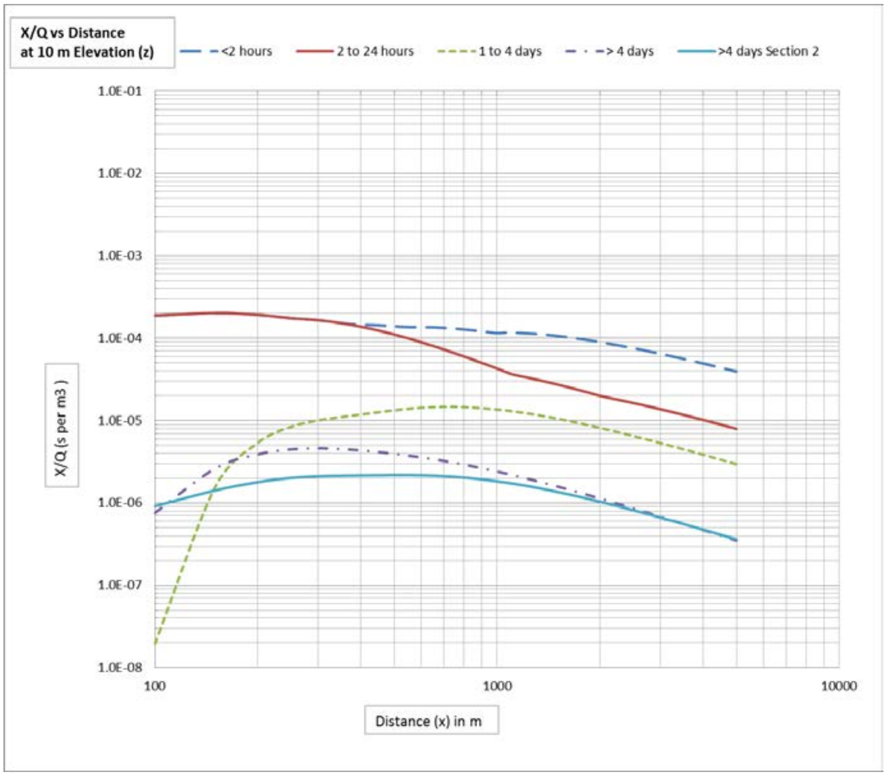


Figure 11-2 – GPM X/Q vs. Distance at 10 m Elevation (z)

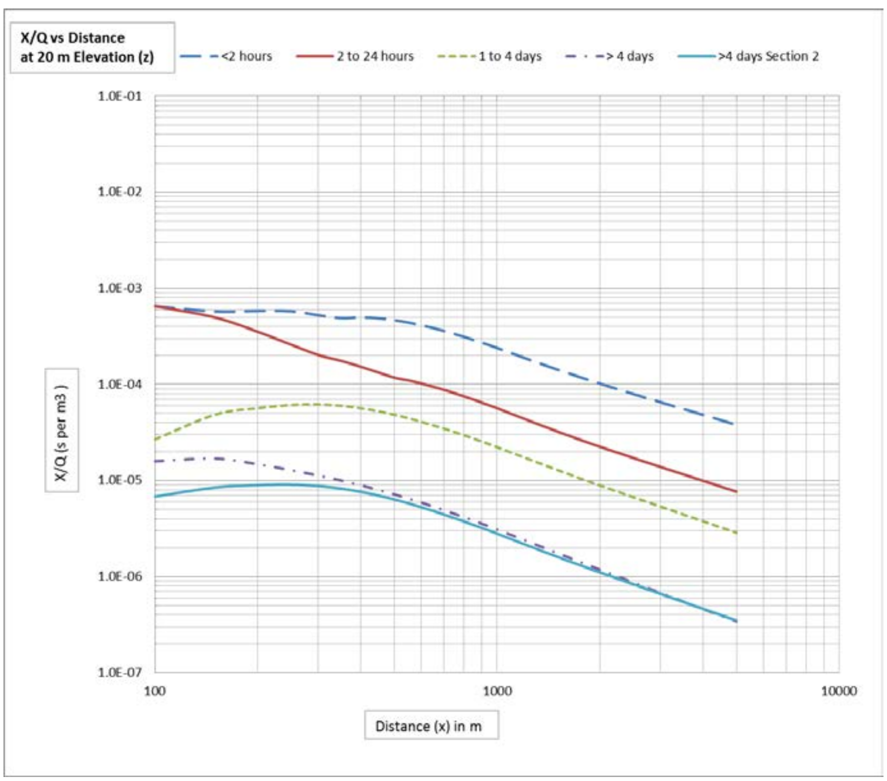


Figure 11-3 – GPM X/Q vs. Distance at 20 m Elevation (z)

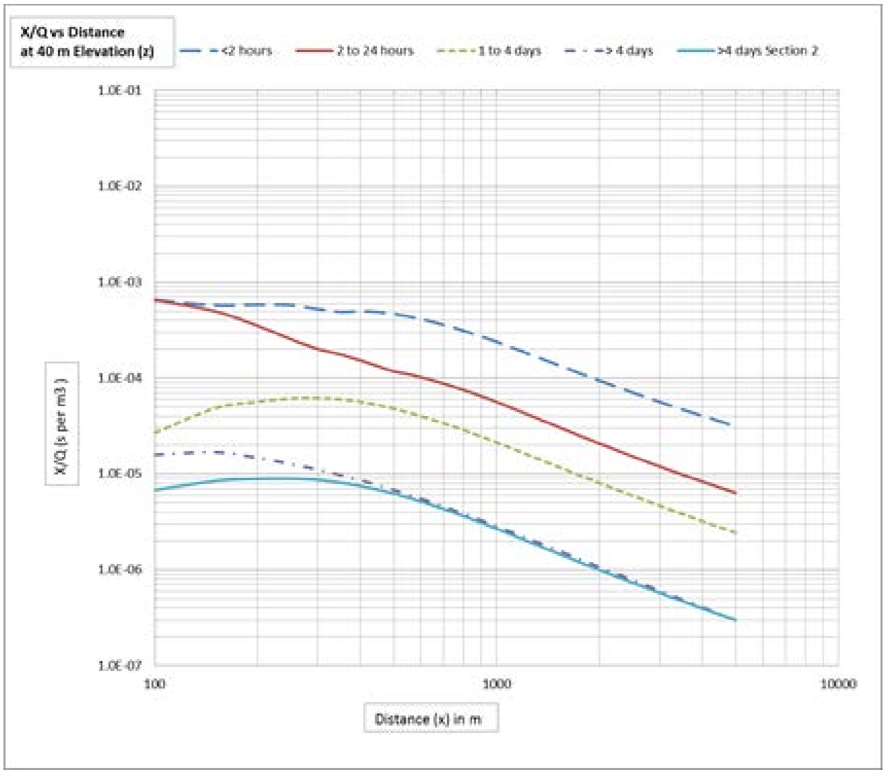


Figure 11-4 – GPM X/Q vs. Distance at 40 m Elevation (z)

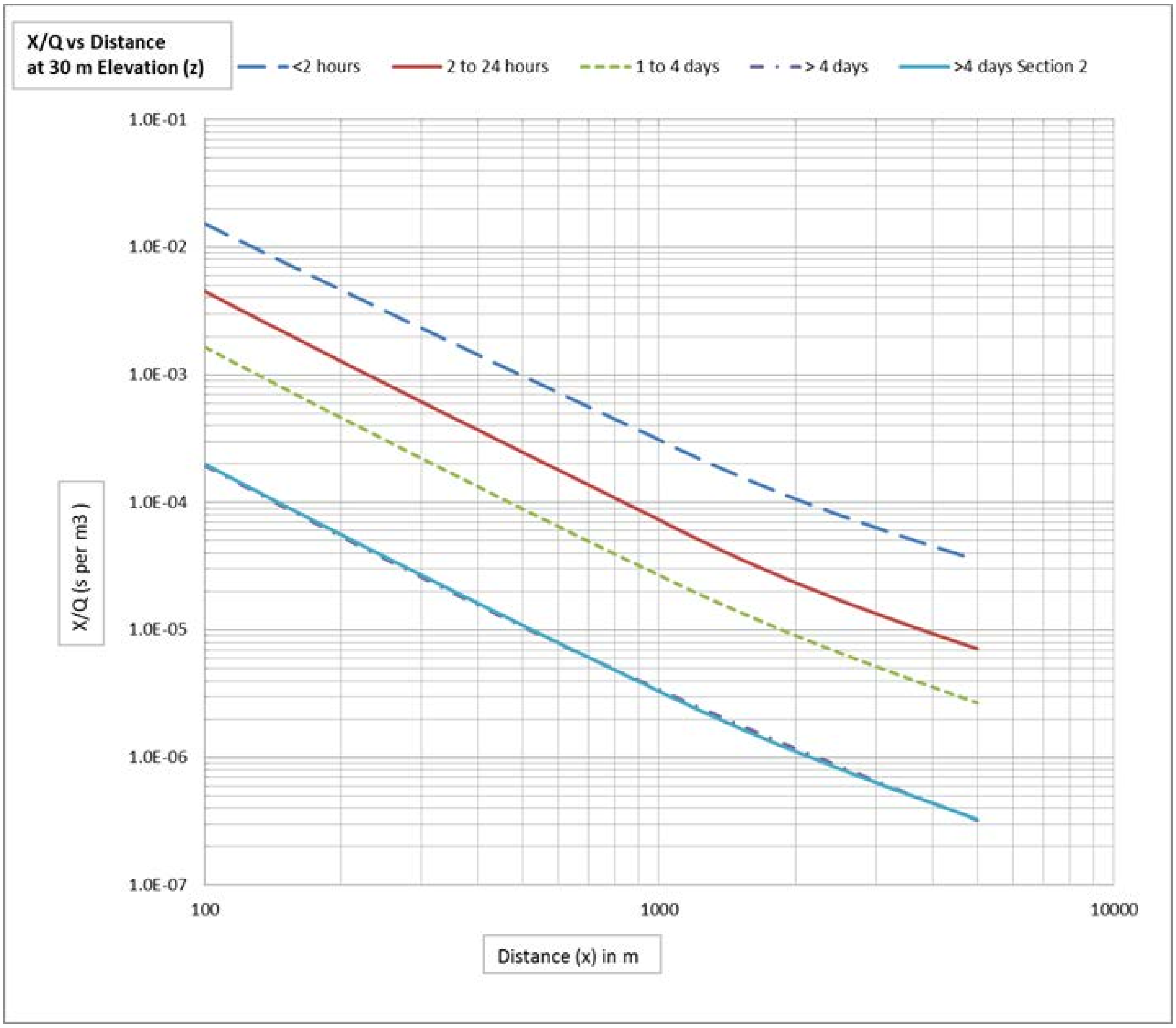


Figure 11-5 – Maximum GPM X/Q vs. Distance at 30 m Elevation (z)

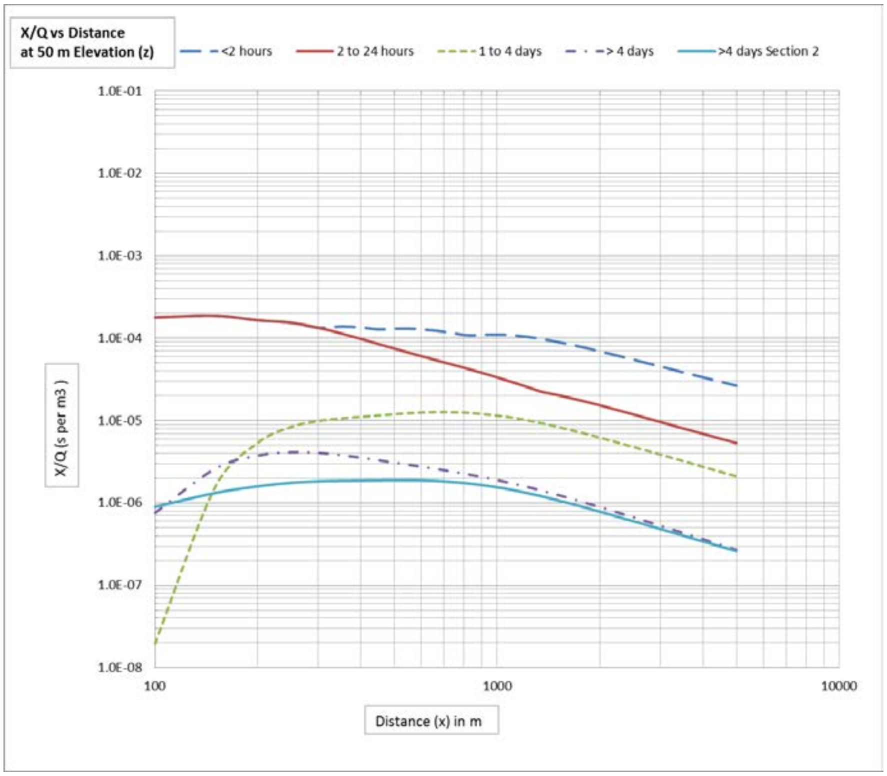


Figure 11-6 – GPM X/Q vs. Distance at 50 m Elevation (z)

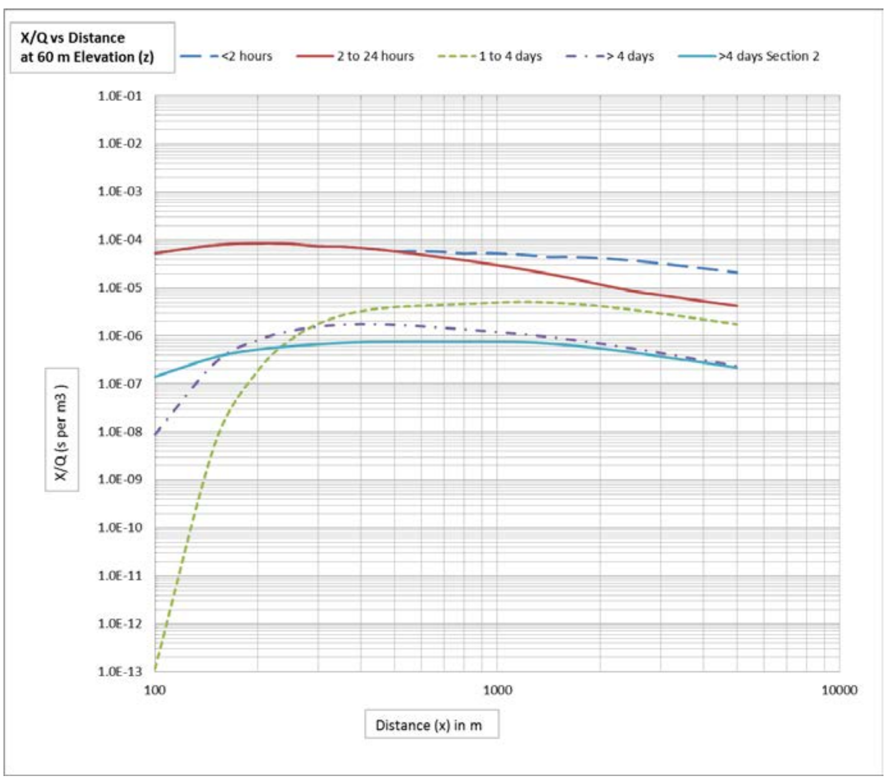


Figure 11-7 – GPM X/Q vs. Distance at 60 m Elevation (z)

Table 11-12 – Maximum GPM X/Q at Distances from 100 – 5000 m for Various Times and Elevations (z)

Time	Elevation	Maximum X/Q (s/m³)
< 2 hours	0	1.69E-04
	10	2.02E-04
	20	6.53E-04
	25	2.44E-03
	30	1.53E-02
	40	6.53E-04
	50	1.87E-04
	60	8.47E-05
2 – 24 hours	0	1.69E-04
	10	2.02E-04
	20	6.53E-04
	25	1.27E-03
	30	4.52E-03
	40	6.53E-04
	50	1.87E-04
	60	8.47E-05
1 – 4 days ANSI/ANS 15.7	0	1.00E-05
	10	1.47E-05
	20	6.19E-05
	25	2.86E-04
	30	1.65E-03
	40	6.19E-05
	50	1.27E-05
	60	5.12E-06
> 4 days ANSI/ANS 15.7	0	3.49E-06
	10	4.61E-06
	20	1.69E-05
	25	6.12E-05
	30	1.95E-04
	40	1.69E-05
	50	4.13E-06
	60	1.75E-06
> 4 days Section 2	0	1.42E-06
	10	2.18E-06
	20	9.06E-06
	25	4.11E-05
	30	2.00E-04
	40	8.96E-06
	50	1.89E-06
	60	7.63E-07

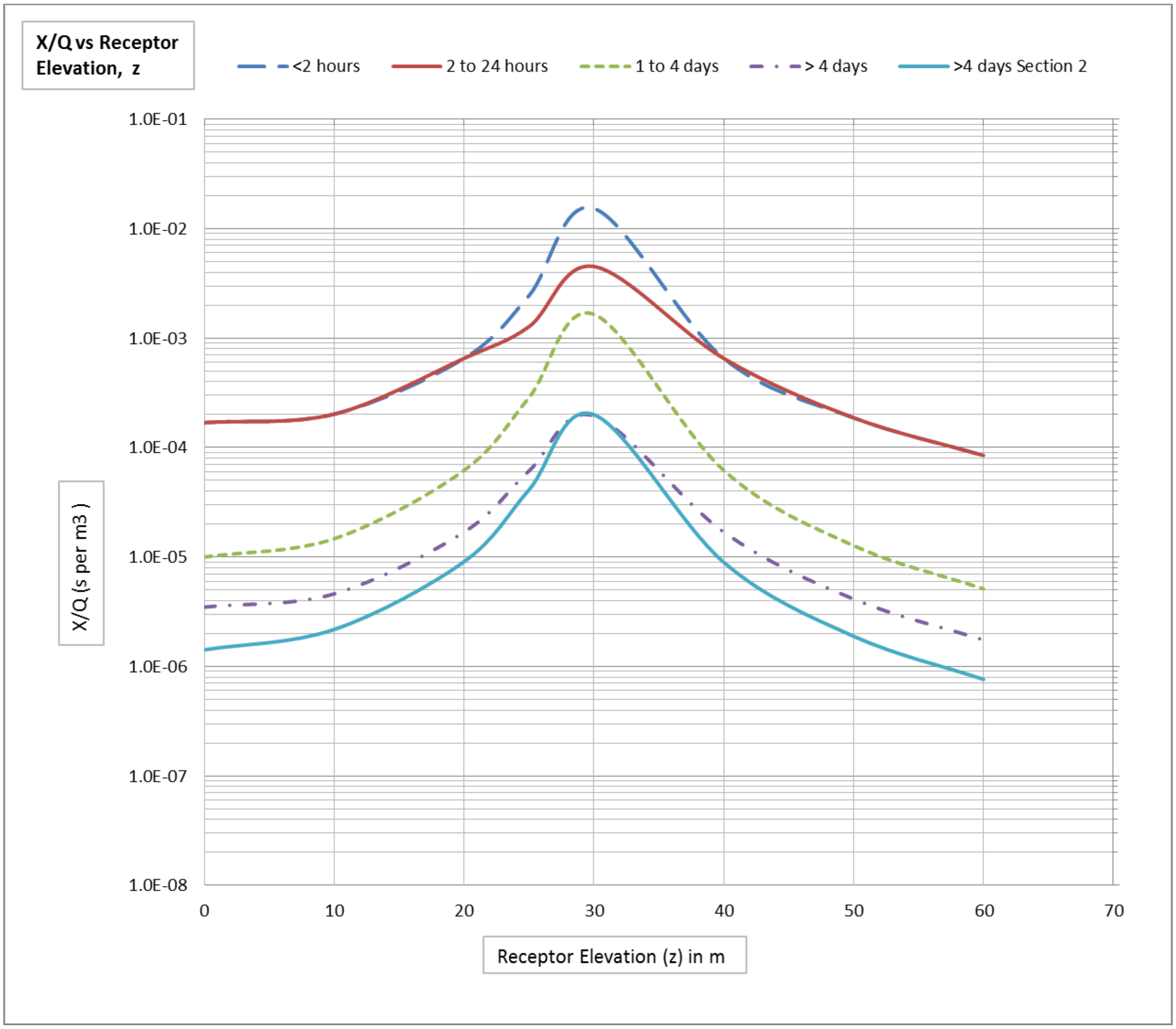


Figure 11-8 – Maximum GPM X/Q vs. Elevation (z) for Distances 100 – 5000 m

[X/Q] Results

From Table 11-12 and Figure 11-8 for the GPM, the ANSI-ANS 15.7[11-20] weather data gives higher [X/Q] values than those using weather data from Section 2. From Figure 11-8, the elevation with the maximum [X/Q] for the GPM is 30 m, or the stack height, for any release period.

For dose assessment, maximum [X/Q] values for occupied locations calculated using the GPM, fumigation model, and calm wind model were used. Results are given in Table 11-13.

For occupied locations, the following are noted:

- Site boundary is located approximately 30 m away from the exhaust stack.
- Closest buildings outside the site boundary are 50 m away (Withers, Mann).
- Closest residential areas are 200 m away (Hillsborough St).
- Student dormitories are 325 m away (Carroll, Syme, North).
- Most buildings are three stories in height.
- DH Hill library is the tallest building near the facility at 150 m away and 30 m high.
- Maximum [X/Q] values are associated with occupied locations that are elevated or closer

to the release point from the 30 m reactor stack. Ground level [X/Q] have lower values.

- There are no occupied areas at distances, x, less than 150 m at a height, z, greater than 12 m.

Table 11-13 – [X/Q] Values for Occupied Locations

Building or Location	Distance x (m)	Height z (m)	Eq. 11-27 [X/Q] Fumigate Up to 24 h (s/m^3)	Eq.11-28 [X/Q] Calm Wind Up to 24 h (s/m^3)	Eq.11-15 [X/Q] GPM 2 h (s/m^3)	Eq.11-22 Eq.11-23 [X/Q] GPM 2-24 h (s/m^3)	Eq. 11-24 [X/Q] GPM 96 h (s/m^3)	Eq. 11-25 Eq. 11-26 [X/Q] GPM > 96 h (s/m^3)
All	30 to 100	up to 12	8.54E-03	3.99E-04	2.31E-04	2.31E-04	1.39E-07	1.56E-06
All	100 to 150	up to 12	2.46E-04	4.73E-05	2.39E-04	2.39E-04	2.73E-06	3.72E-06
All	150 to 5000	up to 30	2.00E-03	4.89E-05	7.57E-03	2.15E-03	7.79E-04	9.15E-05
Withers, Mann	50	12	5.38E-03	1.73E-04	9.73E-05	9.73E-05	4.20E-14	6.80E-22
Broughton, Riddick	70	12	3.97E-03	9.36E-05	2.26E-04	2.26E-04	3.56E-08	1.03E-06
Patterson, Ricks	90	12	3.17E-03	5.80E-05	2.41E-04	2.41E-04	1.08E-06	2.86E-06
DH Hill	150	30	2.00E-03	2.17E-05	7.57E-03	2.15E-03	7.79E-04	9.15E-05
Cox	175	12	1.73E-03	1.58E-05	2.17E-04	2.17E-04	5.68E-06	4.49E-06
Dabney	200	24	1.54E-03	1.22E-05	1.49E-03	5.12E-04	1.85E-04	2.87E-05
Hillsborough St.	200	15	1.54E-03	1.22E-05	2.59E-04	2.59E-04	1.77E-05	7.60E-06
Talley, Reynolds	200	12	1.54E-03	1.21E-05	2.05E-04	2.05E-04	8.97E-06	5.10E-06
Carroll, Syme	325	12	9.93E-04	4.61E-06	1.75E-04	1.64E-04	1.51E-05	5.39E-06
North	350	20	8.23E-04	3.99E-06	4.90E-04	1.76E-04	6.00E-05	1.00E-05
MAXIMUM			***8.54E-03***	***3.99E-04***	***7.57E-03***	***2.15E-03***	***7.79E-04***	***9.15E-05***

The following maximum [X/Q] values were used in this analysis to calculate annual radiation dose in occupied public areas at and beyond the site boundary:

- 8.54×10^{-3} s/m^3 for a release times of 2 hours and 24 hours based on fumigation
- 3.99×10^{-4} s/m^3 for a release times of 2 hours and 24 hours based on calm winds
- 7.79×10^{-4} s/m^3 for a release time from 24 to 96 hours (1 to 4 days)

- 9.15×10^{-5} s/m^3 for a release time from greater than 96 hours (4 days)

Off-Site Dose Assessment for Downwind Distances From 30 m to 5000 m

Annual dose from airborne effluent to members of the public located 30 m to 5000 m away from the reactor stack is calculated using the following equation for a given radionuclide:

$$\mathrm{H}_i = (0.1)(C_i T F_V [X/Q]/[(EC_i)(SDF)]) \quad \textit{Equation 11-40}$$

where,

H is the annual effective dose equivalent

C is the annual average concentration measured in the reactor exhaust duct

F_v is the reactor stack exhaust rate

EC is the effluent concentration given in 10 CFR Part 20 Appendix B or Federal Guidance Report 11 (EPA-520-/1-88-020) Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion.[11-24]

c is the downwash correction factor

X/Q Atmospheric dispersion parameter.

SDF is 7 for normal ventilation with the R-63 fan operating, 20 for confinement with the R-63 fan operating and 1 if the R-63 fan is off

i is the ith radionuclide

0.1 is the effective dose-equivalent rate limit of 0.1 rem (100 mrem) in rem per year. 0.1 rem is associated with continuous exposure for 1 year at the EC given in 10 CFR Part 20 Appendix B.[11-2,11-24]

T is the exposure period of 1 year.

NOTES:

- Units of the parameters vary to give H in rem, mrem, or Sv
- EC for radionuclides other than noble gases is based on 0.05 rem per year to an adult. Children have an age factor of 2 resulting in 0.1 rem per year. Children are assumed to be present in public areas.
- EC for noble gas radionuclides is based on 0.1 rem per year to all age groups.
- No credit is taken for occupancy time or shielding from buildings
- Q is determined using the location of the reactor exhaust sampling system which is after the confinement filters, i.e. the filter retention factor is not needed. Q = C F_v, which makes the product C F_v [X/Q] the downwind concentration, X

- Air effluent is sampled and analyzed as described in Section 11.1.4

The annual average concentration, C(average), is calculated as follows over a calendar year:

$$C_{average} = \frac{\int C(t)dt}{\int dt} \qquad \textit{Equation 11-41}$$

Total effective dose-equivalent is given by the sum of H_i for all radionuclides released;

$$H = \sum H_i \qquad \textit{Equation 11-42}$$

Off-Site External Dose Assessment for Distances Within 100 m

For distances within 100 m of the reactor stack, external dose to ground and elevated locations were calculated by modeling the plume as line sources.

For locations within 100 m from the exhaust stack, the projected external dose rate may be estimated using line sources to represent the stack and an overhead plume. This assumes the plume from the exhaust stack does not reach the ground elevation or intersect a surrounding building at an elevated location for any weather stability class within 100 m from the exhaust stack. This method of external dose rate assessment is conservative since it concentrates dispersed activity into a line and no credit is taken for shielding.

Two locations at heights of 0 m and 20 m at 50 m away from the stack were evaluated. These locations were used since the maximum exposure rate from a line with uniformly distributed activity occurs opposite the midpoint. The elevated location is at the intersection of the midpoint of the 20 m stack line and midpoint of the 100 m overhead line. The ground location is at the midpoint of the overhead line. These lines and locations are illustrated in Figure 11-9.

Figure 11-9 – Ventilation Stack

The reactor building exhaust is connected to the reactor stack at a height of 33 feet (10 m) which leaves the upper 67 feet (20 m) as the line source for the stack.

The stack source term is assumed to be constant until discharged. The stack activity is given by the product of the release concentration and stack volume. The stack volume is based on stack dimensions of 0.25 m radius and length of 20 m, giving a stack volume of 3.93×10^6 ml.

The overhead line source is 100 m long. The overhead line source strength (S) in Ci is given by the release rate (Q in Ci/s) divided by the wind speed (u in m/s) multiplied by 100 m. Q is a product of the stack concentration (C) and stack exhaust rate (Fv). Q is reduced by the Stack Dilution Factor (SDF). S is at a maximum for calm winds which are taken as having a wind speed of 1 mph, or ~ 0.5 m/s.

Exposure rate (X') from the overhead line and stack line were calculated using the following equations and Microshield 5 computer code:[11-15,11-25]

$$X' = \varphi E[\mu_{en}/p]k \qquad \textit{Equation 11-43}$$

For the elevated midpoint and ground midpoint from the overhead line the fluence rate (ϕ) is given by the following:

$$\varphi = \left[(S/(1\times 10^4 * 4\pi h))[\tan^{-1}(5\times 10^3/h) + \tan^{-1}(5\times 10^3/h)]\right] \qquad \textit{Equation 11-44}$$

For the elevated midpoint from the stack line, ϕ is given by the following:

$$\varphi = \left[(S/(2\times 10^3 * 4\pi h))[\tan^{-1}(1\times 10^3/h) + \tan^{-1}(1\times 10^3/h)]\right] \qquad \textit{Equation 11-45}$$

For the ground midpoint from the stack line, ϕ is given by the following:

$$\varphi = \left[(S/(2\times 10^3 * 4\pi h))[\tan^{-1}(3\times 10^3/h) + \tan^{-1}(1\times 10^3/h)]\right] \qquad \textit{Equation 11-46}$$

where,

- X' is the exposure rate in R/h
- E is the gamma photon energy in MeV
- $[u_{en}/\rho]$ is the mass energy absorption coefficient in cm^2/g at energy E
- k is a conversion constant and equal to 6.606×10^{-5} to obtain X' in R/h
- c is the downwash correction factor
- S is the source strength in photons per second, given by the product of activity (A) in decays per second (dps) and radiation yield (Y) in photons or particles per decay (d)
- h is the distance to the receptor, in cm, equal to 5×10^3 cm
- φ photon fluence rate in $cm^{-2}s^{-1}$

1×10^4 cm is the length of the overhead line; 5×10^3 cm is the midpoint of the overhead line; 3×10^3 cm is the full height of the stack; 2×10^3 cm is the length of the stack line; 1×10^3 cm is the midpoint of the stack line and the length of the stack height that has no activity present.

^{41}Argon Dose Assessment

Submersion dose using Equation 11-40 and external dose from the reactor stack and an overhead plume using Equations 11-43 through 11-46 for normal and abnormal releases of Ar-41 were calculated. Concentration and activity limits were determined based on the 10 CFR Part 20 annual constraint dose of 1×10^{-2} rem and dose limit of 1×10^{-1} rem in occupied locations.

The following are noted regarding Ar-41 submersion dose calculations:

- For ^{41}Ar, the dose conversion factor in EPA-520/1-88-020 is 2.17×10^{-10} Sv/h per Bq/m^3 which converts to give 1.42×10^{-8} µCi/ml for 0.1 rem per year.
- Historical data from 2007 to 2014 at 1 MW given in Table 11-2 was doubled for 2 MW.
- Fumigation conditions are estimated to occur up to 4 days per occurrence and 15 days a year, or 360 hours.
- Calm wind conditions are estimated to occur infrequently over 11 days a year (264 hours).
- Occupancy times are not taken into account, i.e. fraction of time in the occupied area getting the maximum dose is taken as 1.

Results are given in Table 11-14 for submersion doses:

Table 11-14 – ^{41}Ar Submersion Dose and Release Limits for Occupied Locations from 30 m to 5000 m

Atmospheric Dispersion Model	[X/Q] s/m^3	Exposure Time, h	Normal Vent Annual Ave µCi/ml	Maximum Dose, rem	Normal Vent µCi/ml for 0.01 rem	Normal Vent Ci for 0.01 rem	Normal Vent Ci for 0.1 rem	Normal Vent with R63 Ci for 0.1 rem
Fumigation	8.54E-03	24	1.14E-07	1.66E-05	6.87E-05	5.24E+00	5.24E+01	4.03E+02
Calm Wind	3.99E-04	24	1.14E-07	7.75E-07	1.47E-03	1.12E+02	1.12E+03	8.62E+03
GPM	2.15E-03	24	1.14E-07	4.18E-06	2.73E-04	2.08E+01	2.08E+02	1.60E+03
Fumigation	8.54E-03	360	1.14E-07	2.49E-04	4.58E-06	5.24E+00	5.24E+01	4.03E+02
Calm Wind	3.99E-04	264	1.14E-07	8.52E-06	1.34E-04	1.12E+02	1.12E+03	8.62E+03
GPM	7.79E-04	96	1.14E-07	6.05E-06	1.88E-04	5.75E+01	5.75E+02	4.42E+03
GPM	9.15E-05	8760	1.14E-07	6.49E-05	1.76E-05	4.89E+02	4.89E+03	3.76E+04

At the annual average concentration for Ar-41, the maximum annual public dose is estimated to be 0.25 mrem due to fumigation conditions occurring over several intervals throughout the year totaling 360 hours (15 days) from normal ventilation without R63 dilution.

A release concentration of 1.8×10^{-5} µCi/ml averaged over one calendar year with an exhaust activity of 489 Ci for normal ventilation without R63 dilution is estimated to give the annual constraint dose of 1×10^{-2} rem in occupied public areas.

External dose for locations within 100 m of the reactor stack were calculated using Equation 11-43 through Equation 11-46. The stack release rate and overhead line source strength varies with ventilation mode. The SDF reduces the stack release rate and overhead line source strength. Results for ^{41}Ar are given in Table 11-15 for the summation of external dose rates from the stack line and overhead line midpoint locations normalized to a release concentration of 1 µCi/ml:

Table 11-15 – Release Rate, Stack and Overhead Line Activity, and Dose Rates from a Release of 1 µCi/ml

Ventilation Mode	Release Rate Q Ci/s	Stack Line Ci	Overhead Line Ci	SDF
Normal	9.72×10^{-1}	3.93×10^{0}	2.17×10^{2}	1
Confinement	2.83×10^{-1}	3.93×10^{0}	6.33×10^{1}	1
Normal	1.39×10^{-1}	3.93×10^{0}	3.11×10^{1}	7
Confinement	1.42×10^{-2}	3.93×10^{0}	3.17×10^{0}	20

Location	Without R-63 Dilution		With R-63 Dilution	
	Normal rem/h at 1 mph	Confinement rem/h at 1 mph	Normal rem/h at 1 mph	Confinement rem/h at 1 mph
Ground	9.28×10^{-2}	2.76×10^{-2}	1.39×10^{-2}	2.14×10^{-3}
Elevated	3.90×10^{-1}	1.14×10^{-1}	5.66×10^{-2}	6.60×10^{-3}

The maximum value is 3.9×10^{-1} rem/h per 1 µCi/ml of ^{41}Ar released under calm winds in normal ventilation without R-63 dilution. Calm winds have a wind speed of 1 mph, or approximately 0.5 m/s. Stronger winds give lower source strength (S) for the overhead line. S values are fixed for the stack line based on ventilation mode.

The previously reported data and Equation 11-43 through Equation 11-46 were used to calculate the concentration in µCi/ml and activity in Ci for ^{41}Ar released that gives the 10 CFR Part 20 annual constraint dose of 1×10^{-2} rem and annual dose limit of 1×10^{-1} rem for locations within 100 m of the stack. Results are given in Table 11-16 below.:

Table 11-16 – ^{41}Ar Release Limits from External Dose for Elevated Locations within 100 m

^{41}Ar Release Limits, µCi/ml				
Annual Dose rem	Normal without R-63	Normal With R-63	Confinement without R-63	Confinement with R-63
1×10^{-2}	2.9×10^{-6}	2.0×10^{-5}	2.9×10^{-6}	5.8×10^{-5}
1×10^{-1}	2.9×10^{-5}	2.0×10^{-4}	2.9×10^{-5}	5.8×10^{-4}

^{41}Ar Release Limits, Ci				
Annual Dose rem	Normal without R-63	Normal With R-63	Confinement without R-63	Confinement with R-63
1×10^{-2}	8.1×10^{1}	5.6×10^{2}	2.6×10^{1}	5.2×10^{2}
1×10^{-1}	8.1×10^{2}	5.6×10^{3}	2.6×10^{2}	5.2×10^{3}

For distances less than 100 m from the stack with calm winds of 0.5 m/s, the minimal average ^{41}Ar concentration allowed for an entire year for normal ventilation without R-63 dilution is calculated at the annual dose limit for an elevated receptor location (x,y,z = 50m, 0m, 20m) based on external dose from an overhead plume:

$$2.9 \times 10^{-5}\ \mu Ci/ml = \left(\frac{0.1\, rem/y}{8760\, h/y}\right) \bigg/ \frac{0.39\, rem/h}{\mu Ci/ml} \qquad \textit{Equation 11-47}$$

An annual average concentration in the reactor stack exhaust of 2.9×10^{-6} µCi/ml with an exhaust activity of 81 Ci gives the annual constraint dose of 1×10^{-2} rem to the public at an elevated location for normal ventilation without R63 dilution from exposure to an overhead plume of Ar-41.

For 2MW, the annual average concentration in the reactor stack exhaust of 1.14×10^{-7} µCi/ml is estimated by doubling the average release given in Table 11-2. This gives a maximum annual dose of 4×10^{-4} rem to the public at an elevated location for normal ventilation without R-63 dilution from exposure to the reactor stack and an overhead plume of Ar-41.

The maximum annual dose from submersion or external exposure to Ar-41 released from normal operation the reactor at 2 MW is estimated approximately 0.4 mrem.

In addition to routine releases, three ^{41}Ar accident release scenarios are considered:

Case 1: A bolus of air from the PN system is removed with ^{41}Ar activity saturated followed by reactor operation at 2 MW with the PN system operating.

Case 2: Same conditions as Case 1 except the PN blower releases activity into the reactor bay via a PN blower rupture preceded by PN system usage for 24 hours at 2 MW.

Case 3: A drained BT is opened immediately after the reactor is operated at 2 MW with the PN system used for 24 hours prior to reactor shut down releasing a bolus of air with ^{41}Ar activity saturated. ^{41}Ar activity is assumed to be released into the reactor bay rather than being released directly to the reactor stack.

Case 3 was calculated to be the limiting accident with a 24 hour average concentration of

4.3×10^{-5} µCi/ml:

Saturation activity of Ar-41 in BT: $A(\infty) = [\sigma\phi N][1\ \mu Ci / 3.7\times10^{4}\ dps] = 2.0\times10^{6}\ \mu Ci$ of Ar-41

Where, σ is the thermal cross section for the (n,γ) with Ar-40 equal to $0.5\times10^{-24} cm^2$

ϕ is the thermal neutron fluence rate at 2 MW in a BT an is equal to $2\times10^{12} cm^{-2}s^{-1}$

based on measured data at 1 MW

N is number of Ar-40 atoms in the maximum BT air volume of 2.95×10^{5} ml

$N = 7.33\times10^{22}$ atoms of Ar-40

24 hour average concentration, [C]: $[C] = [Cp + Cb]\ [1 - \exp(-k \times 24h)] / (k \times 24h)$

$[C] = (8.8\times10^{-4} + 3.3\times10^{-5})\mu Ci/ml[1 - \exp(-0.803\ h^{-1} \times 24h)]/(0.803\ h^{-1} \times 24h)$

$[C] = 4.7\times10^{-5}\ \mu Ci/ml$

Where, Peak concentration $Cp = 2.0\times10^{6}\ \mu Ci / 2.4\times10^{9}\ ml = 8.8\times10^{-4}\ \mu Ci/ml$

Cb is the maximum Ar-41 concentration in the ventilation system prior to opening of the BT, equal to 3300 EC of Ar-41, or 3300 x 1×10^{-8} µCi/ml k is the total removal rate constant = air exchange rate and decay constant

$k = v + \lambda = 803\ h^{-1}$, $v = 0.424\ h^{-1}$, $\lambda = 0.378\ h^{-1}$

Accident scenarios are considered unlikely due to administrative controls on experiments, design features of the reactor, setpoints of the radiation monitoring system, and operational and radiation safety procedures. Therefore, the 24 hour dose from a single accident is analyzed.

For the accidental releases of Ar-41 given in all cases, the confinement system is activated.

Submersion doses were calculated using Equation 11-40 for different atmospheric conditions for occupied locations. Results are given in Table 11-17:

Table 11-17 – ^{41}Ar Submersion Dose from an Accidental Release

Atmospheric Dispersion Model	[X/Q] s/m3	Exposure Time, h	Confinement 24h Ave µCi/ml	Max Dose rem
Fumigation	8.54E-03	24	4.7E-05	2.2E-03
Calm Wind	3.99E-04	24	4.7E-05	1.0E-04
GPM	2.15E-03	24	4.7E-05	5.6E-04

If fumigation conditions are present for 24 hours following an accidental release of Ar-41, a maximum submersion dose of approximately 2.2 mrem is calculated at distances from 30m to 5000m:

$$\mathbf{2.2 \times 10^{-3}}\, \boldsymbol{rem} = \frac{(4.7\times10^{-5}\,\mu Ci/ml)}{(1.42\times10^{-8}\,\mu Ci/ml)}\frac{(0.1\,rem)(24\,h)}{(8760h/y)}(\mathbf{8.54\times10^{-3}}\ \boldsymbol{s/m^3})(\mathbf{0.283}\,\boldsymbol{m^3/s}) \qquad \textit{Equation 11-48}$$

If submersion does not occur, then dose is received from the reactor stack and an overhead plume. The maximum external dose within 100 m of the reactor stack at an elevated location from an accidental release of Ar-41 is calculated to be 0.44 mrem:

$$4.4\times10^{-4}\,rem = \frac{4.7\times10^{-5}\ \mu Ci/ml}{1\,\mu Ci/ml}(0.39\,rem/h)(24\,h) \qquad \textit{Equation 11-49}$$

Maximum dose from an accidental release of Ar-41 at a power of 2 MW is estimated to be approximately 2 mrem.

Vented Fueled Experiment Dose Assessment

Time integrated exposures and dose assessment of vented fueled experiments were analyzed in Sections 7 and 8 of reference 11-33, respectively. Calculated results are given in reference 10-5 Calculation 1 Tables 14-1 for ^{235}U and Calculation 2 Table 14-2 for ^{239}Pu. Radiation dose to members of the public outside the reactor building was calculated as not exceeding 3 mrem from a vented fueled experiment. This dose is accrued and continuously monitored over the irradiation time of the vented fueled experiment, i.e. the exposure period. Multiple vented fueled experiments may be performed over time if the actual dose is lower.

Dose to occupants inside the reactor building during the irradiation time of the vented fueled experiment was made in Section 8 and Calculation 8 Tables 14-22 and 14-24 of reference 11-33 for the exposure time of 520 hours. 520 hours at the allowed fission rate gives the limit for the total number of fissions allowed in the vented fueled experiment. Dose rates and total doses are low and within 10 CFR Part 20 limits.

Airborne Radioactive Effluent

Airborne radioactivity is monitored using the Radiation Monitoring System as described in Section 11.1.4 and released to the environment by an exhaust stack. Filtration of the exhaust by the confinement ventilation system high efficiency particulate air (HEPA) filters and charcoal filters occurs automatically upon exceeding a monitor setpoint or manually by operator action.

Particulate activity is monitored in the exhaust air and reactor building. Submersion dose is measured by the area radiation monitors and personnel dosimeters. Air sampling for other radionuclides may be performed using in-line equipment from the reactor exhaust stack and portable equipment used inside the reactor building, as necessary. Any abnormal air sample analysis requires an investigation as described in the RP Program.

An objective of the RP Program is to limit public dose from airborne effluent to the constraint dose of 10 mrem per year as stated in 10 CFR Part 20. This dose is accrued and tracked over the year allowing time for releases from experiments and operations to be investigated, assessed, and controlled.

Measured ^{41}Ar concentrations and calculated doses from 1996 to 2015 are shown in Figure 11-10. Dose from airborne activity is historically below 10 percent of the applicable limits. Environmental monitoring, as described in Section 11.1.7, is performed and indicates radiation dose and airborne activity are at non-detectable to low levels.

Radiation dose to members of the public from ^{41}Ar from routine or accident releases were calculated to be below the annual constraint dose of $1{\times}10^{-2}$ rem given in 10 CFR Part 20.

Airborne releases from fueled experiments have not occurred in the history of reactor operations as of the date of this analysis. In the future, planned releases of fission gases and halogens may occur from fueled experiments with calculated radiation doses that do not exceed 3 mrem in public locations.

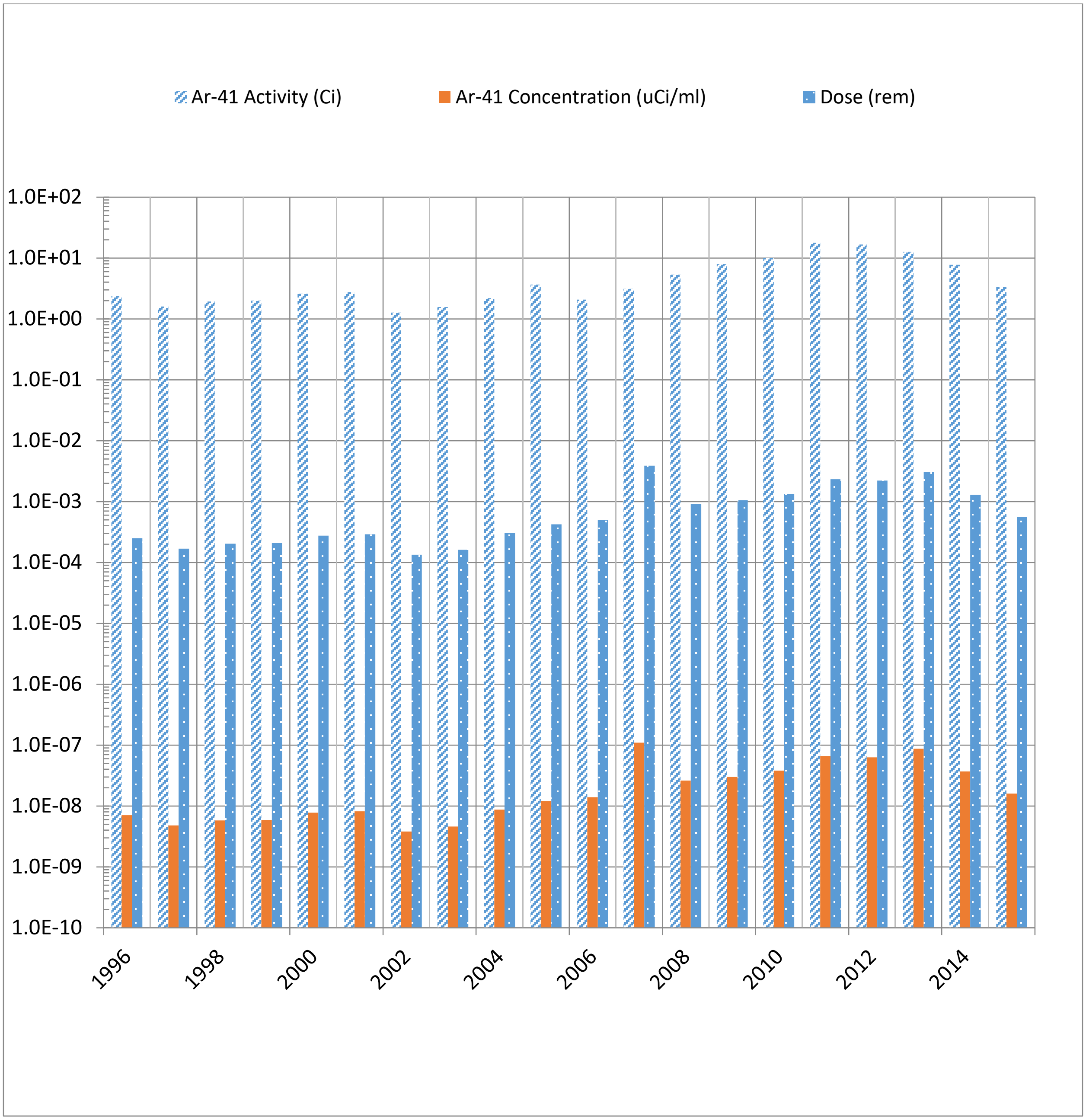


Figure 11-10 – ^{41}Ar Airborne Effluent Date – 1996 to 2015

The following are noted regarding airborne releases:

1. R-63 fans and normal ventilation is in use most times, making the SDF of 7 applicable.
2. No occupied location presently exists at the stack height within 100 m of the stack.
3. Historic data from 1996 to 2015 has given annual average concentration of approximately 1×10^{-7} µCi/ml and a release activity of 20 Ci, which are well below the annual average values calculated that result in the annual constraint dose of 1×10^{-2} rem to the public.
4. Releases of tritium and ^{16}N were analyzed to give negligible dose.
5. Other radionuclides from reactor operation and experiments are not routinely released.

6. Radiation monitoring is in place to detect airborne activity releases.

11.1.1.2. Liquid Radioactive Sources

Liquid sources are associated with reactor operations, maintenance, and experiments. Activity is present is the primary coolant system, liquid samples used in experiments and associated laboratories, condensate from the reactor air handling system, and liquid waste.

Primary Coolant System

[redacted]

Figure 11-11 – Primary Coolant System

Materials present in the primary coolant system include:

1. De-ionized water (> 0.5 MΩ-cm)
2. Reactor fuel (Zircaloy clad, Zircaloy 2 and 4 composition)
3. Control Rods (Ag, In, Cd with Sn coating)
4. Stainless steel (SS 304, SS 316 composition)
5. Aluminum (6061 composition)

The primary coolant system is limited to 120 degrees Fahrenheit.

Nominal flow rate is 1000 gpm at a 2 MW power limit.

Radionuclides present in the primary coolant from reactor operation are produced as activation products of water and trace impurities from corrosion. These radionuclides include, but are not limited to, ^{16}N, ^{41}Ar, ^{19}O, ^{3}H, ^{24}Na, ^{28}Al, ^{38}Cl, ^{56}Mn, ^{110m}Ag, ^{60}Co, ^{65}Zn, ^{54}Mn, ^{59}Fe, and ^{124}Sb.

1. ^{16}N is a coolant activation product produced by a (n,p) fast neutron reaction with ^{16}O. ^{16}N has a short half-life and emits high energy gamma photons. ^{16}N decays significantly during transit in the primary coolant system.
2. ^{41}Ar is produced as an activation product of air. Air can be dissolved or entrained in water or may off-gas from in-pool experiments. Abundance of Argon in air is low but because of its short half-life ^{41}Ar levels in primary coolant during operation are detectable.
3. ^{19}O is produced by neutron activation of ^{18}O. Oxygen is a component of water. ^{18}O is in low abundance and ^{19}O is short-lived. Due to the short half-life ^{19}O activity is produced at detectable levels in the primary coolant. ^{19}O decays significantly during transit in the primary coolant system.
4. Tritium, ^{3}H, is produced by the neutron irradiation of deuterium (^{2}H) in water. ^{2}H abundance in water is low. Diffusion of tritium produced by tertiary fission is negligible due to the low yield, low power, and low fuel temperature.
5. ^{24}Na and ^{28}Al are activation products of aluminum, Al, produced by the (n,α) and (n,γ) reactions, respectively. The reactor pool liner and several components in the reactor pool are made of aluminum.
6. ^{38}Cl production is associated with activation of impurities in graphite, including stable ^{37}Cl. Due to the short half-life ^{38}Cl activity produced is significant. Graphite reflectors and experimental equipment have been used and are stored in the reactor pool. ^{38}Cl levels have decreased in recent years since the graphite reflectors were replaced with beryllium reflectors.
7. ^{110m}Ag is present in reactor coolant produced by activation of the control rods.
8. ^{60}Co is present in reactor coolant and is produced by neutron activation of ^{59}Co contained in stainless steel. Several components used in the reactor primary coolant system are made of stainless steel. Other common activation products of stainless steel include ^{54}Mn, ^{56}Mn, ^{65}Zn, and ^{59}Fe.
9. ^{124}Sb is associated with activation of lead. Sb is used as a hardening agent in lead. Lead is used in experimental equipment used in the reactor pool.

Corrosion and activation products are present in the reactor primary coolant at low activity concentrations. The concentrations are stable under steady-state reactor operation. If the reactor is shut down, short-lived radionuclides decay significantly.

Short-lived radionuclide activities, e.g. ^{24}Na, ^{38}Cl, and ^{56}Mn, vary with reactor operating and shut down times. The purification system demineralizer reduces the concentration of these radionuclides in the primary coolant. For typical daily operating cycles, concentrations of these radionuclides are in the range of 1×10^{-6} µCi/ml to 1×10^{-4} µCi/ml. For prolonged operating times, e.g. one or more continuous days, the concentrations of these radionuclides reaches an equilibrium value. Following reactor shut down, these short-lived radionuclides decay and are removed by the purification system demineralizer to non-detectable levels (less than 10 percent of unrestricted area limits given in 10 CFR Part 20 Appendix B).

Longer lived radionuclide concentrations are low and do not produce measurable radiation levels in the primary piping, delay tanks, primary pump or heat exchanger. These radionuclides, primarily ^{110m}Ag, ^{60}Co, ^{65}Zn, and ^{54}Mn, are removed from the primary coolant and build up in the primary purification system demineralizer.

Concentrations of longer-lived radionuclides are estimated below at 2 MW based on measurements made at 1 MW. The 1 MW values were doubled since the production rates are doubled at 2 MW.

Table 11-18 – Primary Coolant Radionuclides

Nuclide	**Primary Coolant Concentration at 2 MW µCi/ml**
^{110m}Ag	6×10^{-6}
^{60}Co	4×10^{-6}
^{59}Fe	6×10^{-6}
^{54}Mn	2×10^{-6}
^{24}Na	1×10^{-4}
^{65}Zn	6×10^{-6}
^{3}H	2×10^{-4}

Fission product activity is normally not present in the primary coolant due to fuel clad integrity. Fission product activity in the primary coolant would be detected by abnormal readings from the radiation monitor at the purification system demineralizer and during technical specification surveillance on primary coolant. If present, the leaking fuel element(s) would be identified and contained.

To decrease the generation of radioactive materials associated with water impurities, a service water demineralizer is used to treat tap (potable) water before it is added to the pool. Impurities in the primary coolant are controlled by a bypass demineralizer where a portion of the primary coolant flow is filtered and passed through a mixed bed ion exchange column. Corrosion control is accomplished by this demineralizer system, which maintains the resistivity of the water greater than 500 kΩ·cm. Replacement of the primary demineralizer resin is scheduled when the pool resistivity meter indicates between 0.5 to 1 MΩ·cm.

With adequate demineralization, the principal activity in the primary water leaving the core is caused by ^{16}N. In the forced convection mode of cooling, the delay tanks in the primary system allows for decay of ^{16}N. As analyzed in Section 11.1.1.1 and Section 11.1.5, ^{16}N concentration leaving the reactor core is estimated to be 43 µCi/ml in forced convection cooling at 2 MW and 440 µCi/ml in natural convection cooling. Very low levels of activity normally exist downstream from the delay tanks in the primary coolant system.

As required by the technical specifications, primary coolant is sampled and analyzed for resistivity and gross and isotopic radioactivity. This monitoring is used to indicate the effectiveness of corrosion control and existence of unusual radionuclides.

Experiments and Laboratories

Liquid sources from experiments typically have activities on the order of less than 1 µCi to mCi. Liquid sources used in experiments are limited and monitored as specified in approved experiment authorizations. Liquid waste from the laboratories consists of residues and dissolved materials from cleaning contaminated glassware or similar items. After use, liquids are collected for disposal or disposed of into the liquid waste system. Refer to Section 11.2 for details on liquid waste systems.

Condensate

Tritium in the reactor pool evaporates and enters the air space in the reactor building. The levels of ^{3}H in air are low, but become concentrated in the reactor building air conditioning condensate. The condensate from the air conditioner is collected into a holding tank located in the primary piping vault (PPV). The condensate is then processed and sampled as liquid radioactive waste. Refer to Section 11.2 for details on liquid waste systems.

Liquid Waste

Sources of aqueous liquid waste are associated with reactor operations, maintenance, and experiments, e.g. air conditioning system condensate, draining of beam tubes, change-out of resins in the primary coolant purification demineralizer system, and water associated with the primary coolant system (e.g. equipment leakage) and in the associated laboratories.

High activity, chemical, biological, non-aqueous, or non-radioactive hazardous wastes are not placed in the liquid waste system. These wastes are collected and transferred for disposal by NCSU Environmental Health and Safety Center in accordance with applicable regulations.

Liquid waste is treated as described in Section 11.2.2. A representative sample of each liquid waste tank is collected for analysis of gross beta, tritium, and gamma isotopic activity and concentration. Each liquid waste sample is tested to meet the definition of being readily dispersible as required by 10 CFR Part 20. This test is made by analyzing waste water passed through a 0.45 micron rated filter. If the activity on the filter is non-detectable, then the liquid waste is considered readily dispersible. Non-detectable is defined in facility procedures as being less than 5×10^{-8} µCi/ml for beta activity (i.e. 50 pCi/l, the environmental limit for beta activity).

The RHP reviews liquid waste analysis prior to release of each waste tank using approved procedures. If discharge requirements to the sanitary sewer given in 10 CFR Part 20 and other applicable local ordinances and NCSU policies are met, then the waste tanks are released to the local sanitary sewer system. If discharge requirements to the sanitary sewer given in 10 CFR Part 20 and other applicable local ordinances and NCSU policies are not met, then the waste tanks are pumped to containers or drums and either treated a second time by the liquid waste system or transferred to NCSU Environmental Health and Safety Center for disposal in accordance with applicable regulations. Liquid waste sampling, analysis, and discharge are controlled by the RHP using approved procedures.

Historically, radioactive liquid wastes from the reactor facility are low in activity, low in concentration, and low in volume. Concentrations are typically below the sanitary sewer limits at the point of discharge from the reactor facility. If needed, dilution from the NCSU campus at a minimum of 670,000 gallons per day is claimed. This provides adequate dilution if averaged over one month to meet 10 CFR Part 20 limits.

A summary of sanitary sewer discharges from 1996 to 2015 given in Figure 11-12.[11-11]

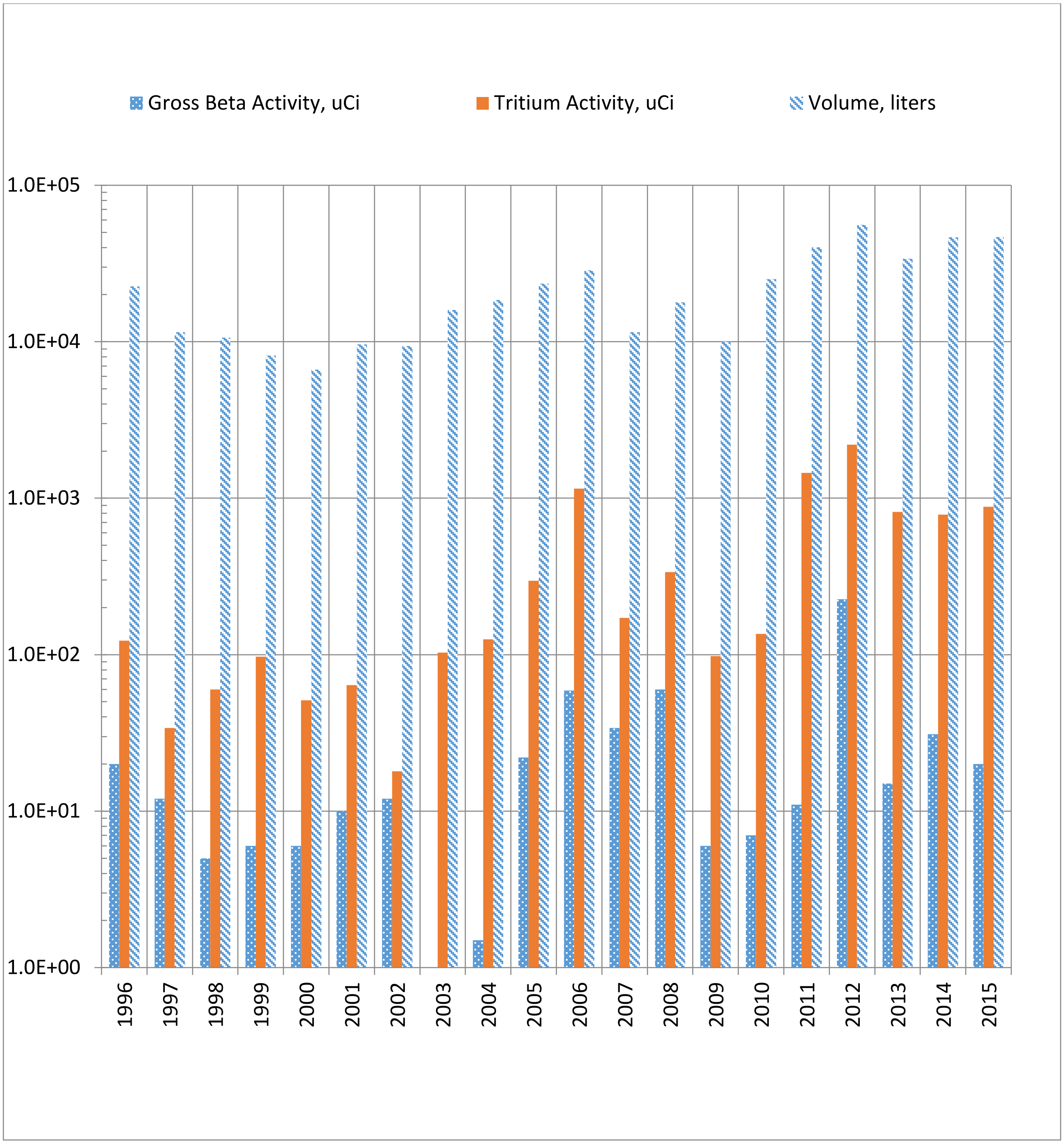


Figure 11-12 – Sanitary Sewer Discharges – 1996 to 2015

11.1.1.3. Solid Radioactive Sources

Solid sources include reactor fuel, irradiated reactor and experiment equipment/components, calibration sources and fission sources, experiment samples, and solid waste. Radioactive source use and access are restricted by physical and administrative controls.

The reactor facility security plan, RP Program, and NCSU Radiation Safety Manual limit unescorted access to authorized and trained personnel. Only authorized and trained individuals work with radiation sources. All experiments are reviewed and approved as described in Section 11.1.2 and Section 12.

The reactor building is a controlled access area and has a security plan as required by 10 CFR Part 73. Currently, total activity of sealed radioactive sources does not exceed Category 2 limits listed in 10 CFR Part 37. Experiment activity limits are well below those given in 10 CFR Part 37 for either in a single source or total for all sources.

Radioactive sources are located in the reactor pool, reactor experimental facilities (e.g. beam tubes), and designated storage locations within the reactor building. Other radioactive sources are located in State of NC regulated laboratories which are located adjacent to the reactor building in the Burlington Engineering Laboratory (BEL). Controls required by the RP Program and regulations for storage, posting and labeling are followed. Sources and storage areas are shielded if reasonable. The proximity of the reactor facility to the BEL is shown in Section 1.

Sealed radioactive sources are inventoried and leak tested as required by license conditions and regulations using facility procedures. SNM is inventoried as required by license conditions and 10 CFR Part 74 *Material Control and Accounting of Special Nuclear Material*. Experiment samples are listed on an experiment log as specified in facility procedures.

Reactor Fuel

New reactor fuel is stored in an approved storage rack in the reactor facility that meets technical specification requirements for criticality control. This storage area is monitored for radiation levels and is located within the reactor facility controlled access area.

Reactor fuel in current use in the reactor core is located in the reactor pool. Currently, no reactor fuel is spent. Used fuel in the reactor pool is stored in approved storage racks or the fuel storage pits that meet technical specification requirements for criticality control. The reactor pool is monitored for radiation levels and pool level and is located within the reactor facility controlled access area.[11-14]

Fuel handling tools and equipment are secured when not in use by authorized personnel. Only authorized personnel are allowed to handle fuel. Fuel is inventoried and accounted for by the Reactor Health Physicist (RHP) as required by 10 CFR Part 74 and facility procedures.

If reactor fuel is to be disposed, arrangements are made with the US Department of Energy. Handling, packing, packaging, and shipment is made in accordance with applicable regulations, facility procedures, and the following documents:[11-1]

1. Quality Assurance Program for Packaging and Transportation of Radioactive Material, required by 10 CFR Part 71 and as approved by the US NRC, approval number 71-0331.[11-27]

2. Security Plan for Shipment of Hazardous Material at the NCSU PULSTAR Nuclear Reactor, required by 49 CFR Part 172.[11-28]

Irradiated components / equipment

Components and equipment used in reactor operations and experiments become radioactive by activation. Examples include in-pool irradiation containers and beam tube devices and collimators. Materials used are selected that minimize activity and volume while meeting performance objectives and specifications, e.g. Aluminum is often used in place of steel. Materials are held for decay prior to handling as needed. Longer lived, high activity sources are few and are stored in the reactor pool until these items are considered radioactive waste and disposal is arranged.

Calibration sources and detectors

Calibration sources include a 5 Ci Pu(Be) startup source that is used to check reactor power monitors located in the reactor pool. This source is long-lived and needed for reactor operation. It is stored in the reactor pool and only removed from the reactor pool for maintenance or other approved testing.

Detectors include fission chambers, foils, and wires that are used in power monitors and for fluence rate or fluence measurements for experiments. These typically are of low mass and allowed to decay prior to handling. Longer-lived, high activity sources are few and are stored in approved locations in the reactor facility or associated laboratories until these items are considered radioactive waste and disposal is arranged.

Samples

Solid samples from experiments become radioactive by activation. Only materials approved in the experiment authorization and reactor facility procedures are used. These are used for experimental measurements and are recorded in an experiment log. Activities are normally short-lived or longer-lived radionuclides in the pCi to µCi range for each sample. Collective activity ranges up to several µCi to less than a mCi. Samples are stored in approved locations or collected and disposed as solid waste.

Samples used in fueled experiments are inventoried and accounted for by the Reactor Health Physicist (RHP) as required by 10 CFR Part 74 and facility procedures.

Solid waste

Solid wastes consist mostly of spent dewatered resins, dry active waste (trash and disposable items), and experiment samples (e.g. residues, solids, and sample containers). Spent resins typically contain up to a few mCi of longer-lived activation products, e.g. ^{110m}Ag, ^{60}Co, ^{54}Mn, and ^{65}Zn. Dry active waste typically contains activation products from reactor operations and experiments of a few µCi to 1 mCi. Occasionally, activated or contaminated equipment is collected. Activity varies from a few µCi to mCi levels.

All radioactive solids from the reactor building and associated laboratories are collected, monitored, and stored for transfer to the NCSU Environmental Health and Safety Center for disposal.

A summary of solid waste generated from 1996 to 2015 given in Figure 11-13.[11-11] There was no waste disposal in 2003.

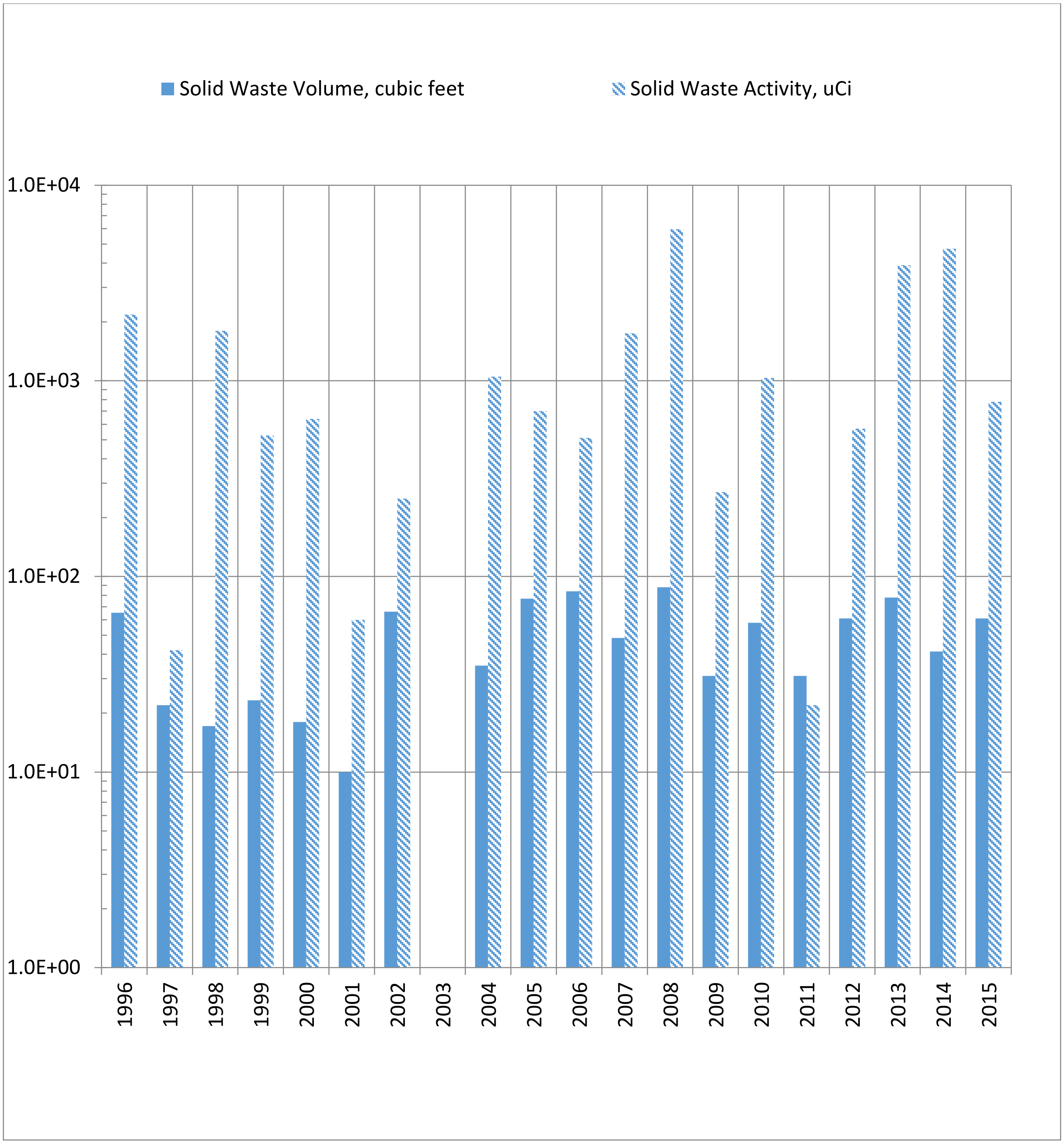


Figure 11-13 – Solid Waste – 1996 to 2015

11.1.2. Radiation Protection Program

A documented radiation protection (RP) program containing administrative policies and guidelines committed to by the Nuclear Reactor Program (NRP) are established in facility documents and the health physics (HP) procedures.[11-1,11-2,11-3,11-5]

Due to the unique nature of the reactor from the rest of North Carolina State University (NCSU) and regulatory jurisdiction differences, a specific RP Program is in place at the reactor. The PULSTAR reactor RP Program is based on ANSI/ANS-15.11.

In addition, the reactor RP Program meets requirements given in applicable federal and state regulations and the NCSU Radiation Safety Manual. The NCSU Radiation Safety Manual is developed and implemented by the NCSU Radiation Safety Officer and approved by the NCSU Radiation Safety Committee.

The RP Program and HP procedures include the following:

1. ALARA (As Low As Reasonably Achievable) policy statement, as well as specific practices.
2. Respiratory protection policy and practices.
3. Radiological controls for occupational workers.
4. Monitoring, assessment, and control of radioactive effluent and releases.
5. Monitoring, assessment, and control of radioactive sources.
6. Radiological surveys.
7. Radiation instrument calibration and use, including portable instruments, reactor radiation monitoring system, and other radiation monitors.
8. Radioactive shipment and waste disposal.
9. Radiological controls on reactor operation, maintenance activities, and experiments. These include radiation work permits, obtaining radioactive material authorizations from NCSU, and experiment reviews and approvals.
10. Training and radiological access control.
11. Requirements for escorting visitors.

The Reactor Health Physicist is responsible for development, implementation, and maintenance of the RP Program. The RHP works with the NRP Director and NRP Manager of Engineering and Operations (MEO) to implement the RP Program. The NCSU Radiation Safety Officer provides support and assistance to the RP Program. Oversight of the RP Program is provided by both the Reactor Safety and Audit Committee (RSAC) and Radiation Safety Committee (RSC). The RHP has the authority to halt work and report radiological safety concerns directly to higher level administrators at NCSU. Details on the organization are provided in Chapter 12.

Each experiment is reviewed and approved by the Manager of Engineering and Operations (MEO), Reactor Health Physicist (RHP), NCSU Radiation Safety Officer (RSO), Reactor Safety and Audit Committee (RSAC), and NCSU Radiation Safety Committee (RSC). Experiment reviews meet facility technical specification and 10 CFR Part 50.59 requirements. Individual experiments conducted under the experiment authorization are reviewed and approved by the MEO and RHP, and the RSAC and RSC as necessary, before being conducted. Radiation levels and radioactivity are estimated by the experimenter prior to each individual experiment conducted under the experiment authorization. Radiological controls for individual experiments follow the experiment authorization, the RP Program, and any additional controls as specified by the RHP. Reviews of experiments are also described in Section 12.

All radiological controls and limits on experiments follow the “As Low As Reasonably Achievable” (ALARA) principle. More details on experiment controls are given in Section 10. ALARA and the RP Program are described later in Section 11.

The RP Program, experimental controls, reactor systems, radiation monitoring system, and ventilation system are in place to minimize and measure airborne radioactivity levels. Radionuclides produced may become airborne depending on the physical form and release pathway. Release of radionuclides to the reactor building air space is minimized by various radiological, engineering, and experimental controls, such as decay, decontamination, design features, limitations on irradiations (mass, fluence) to limit the production of activity, encapsulation, and handling precautions. All experiments and maintenance activities are reviewed to determine if the potential for airborne radioactivity exists and to specify controls. Very few maintenance and experimental activities have the potential for producing an airborne radioactivity area as defined in 10 CFR Part 20.

Exposure of personnel and the public are limited to levels given in the RP Program and experiment authorizations. Radiation surveys of experiments and the reactor building are performed to verify radiation fields and radioactivity produced are within expected and established limits. Radiation fields range from background to a high radiation area (HRA), with most being below the radiation area limits as defined in 10 CFR Part 20. Some localized areas are within the radiation area limits and HRA limits as defined in 10 CFR Part 20. Very high radiation areas are not typically encountered. All HRA, and VHRA if present, are controlled as specified in the RP Program and as required by 10 CFR Part 20.

Radiation fields in occupied areas within the reactor facility range up to those defined in 10 CFR Part 20 for a high radiation area. Most occupied areas within the reactor facility are below those defined in 10 CFR Part 20 for a radiation area and are typically below 2 mrem/h. Occupational radiation dose typically ranges from background to approximately 10% of the annual limits. Few occupational radiation doses are above 10 percent of the annual limits, i.e. the level where official monitoring is required, and typically do not exceed 25 percent of the annual limits. Occupational radiation dose is closely monitored to avoid potentially exceeding the annual limits. Median annual occupational radiation dose is less than 0.2 rem for the operating history of the facility since 1972. ALARA principles are followed for all occupationally exposed individuals.

Radiation dose to members of the public occurs by direct exposure and exposure to radioactive effluent. Public doses are calculated and measured to be well within the annual limits given in 10 CFR Part 20, i.e. well below 100 mrem per year. Public dose has been within the annual constraint dose level given in 10 CFR Part 20 for airborne activity, i.e. less than 10 mrem per year. Liquid effluent typically meets sanitary sewer release limits and rarely exceeds unrestricted area release limits given in 10 CFR Part 20 Appendix B prior to being discharged. Accounting for dilution in the sanitary sewer system on the NCSU campus, 10 CFR Part 20 Appendix B limits have always been met since the facility became operational. All liquid effluent is collected and analyzed to meet 10 CFR Part 20 requirements prior to being batch released. Radioactive materials, other than those being shipped in accordance with applicable regulations or transferred to authorized personnel in accordance with the RP Program, are not released to unrestricted areas.

An environmental radiation monitoring program has been established and results are included in the annual report required by the facility license as described in Chapter 12. Based on annual reports for the facility's operating history, it is concluded that radiation dose to members of the public is well within regulatory limits and is typically non-detectable.

Release of radioactive materials to unrestricted areas is controlled by monitoring of materials leaving the facility, (e.g. trash, hand carried items, equipment, etc.). Monitoring practices and release criteria are established in the RP Program to avoid the potential inadvertent release of radioactive material.

Radiation dose to occupational workers and members of the public is assessed periodically throughout the calendar year. The RP Program requires personnel monitoring using official, accredited, radiation dosimetry meeting 10 CFR Part 20 requirements and by bioassay, as necessary, for occupational workers. Air sampling, radiation surveys, contamination surveys, effluent assessments, and environmental monitoring are performed at frequencies sufficient to detect if an annual radiation dose limit is being approached. Administrative radiation dose levels are established in the RP Program to reduce the potential of exceeding an annual limit.

11.1.3. ALARA Program

The ALARA (As Low As Reasonably Achievable) concept is inherent in radiation protection and a policy statement, as well as specific practices for use at the reactor facility, are formally stated in the Radiation Protection (RP) program. Every use of radioactive materials and reactor operations are conducted with ALARA as a guiding principle.

ALARA means making every reasonable effort to maintain exposures to radiation as far below the dose limits as is practical consistent with the purpose for which the authorized work activity is undertaken, taking into account the state of technology, the economics of improvements in relation to the state of technology, the economics of improvements in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest.

Implementation of ALARA requires that an authorized work activity or practice be first justified, second that protection is optimized, and finally that acceptable dose limits are established. Specifically, a practice shall:[11-1]

1. Provide benefits to exposed individuals that exceed radiation induced detriments.
2. Keep exposures ALARA for any source taking social factors into account but constrained by restrictions on risks to individuals to limit inequities.
3. Ensure that no individual is deliberately exposed to radiation risks judged to be unacceptable in any normal circumstances.

The application of the ALARA principle should not be misinterpreted as simply a requirement for dose reduction irrespective of the dose level; sound judgment is essential in its proper application. Specifically, unnecessary restrictions which forfeit beneficial outcomes or excessive monetary costs with little benefit are inappropriate.

Implementation of ALARA at the reactor facility shall use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are ALARA.

The RP Program, including the implementation of ALARA, are reviewed at least annually as required by 10 CFR Part 20 and the technical specifications (TS). The RHP and/or Radiation Safety Officer perform an annual review of the RP Program. The Reactor Safety and Audit Committee (RSAC) perform an annual audit as required by TS. The NCSU Radiation Safety Committee (RSC) also conducts audits and reviews of reactor operations, experiments, and radiation protection as specified in the NCSU Radiation Safety Manual and TS. All experiments are reviewed by the RHP prior to being performed as stated in TS and facility procedures.

Based on the operational history at the reactor facility and with ALARA practices employed, ALARA is considered as being successfully implemented by keeping:

1. Personnel doses below administrative dose levels and ALARA dose goals established in the RP Program and NCSU Radiation Safety Manual.
2. Public doses below the annual dose constraint of 1×10^{-2} rem as stated in 10 CFR Part 20.

3. External dose rates from reactor operations and experiments below 2 mrem per hour in commonly accessible areas.
4. Removable contamination below contamination limits given in the RP Program in commonly accessible areas.
5. Airborne activity inside the reactor building to levels that do not officially require air sampling, i.e. less than one percent of a Derived Air Concentration given in 10 CFR Part 20 Appendix B.
6. Radioactive liquid waste and long-lived radionuclides in the reactor coolant system to concentrations below those given in 10 CFR Part 20 for discharge to the sanitary sewer.
7. Solid radioactive waste volume and activity low by using short-lived radioactive materials, reusable items or items that are easily decontaminated, limiting the amount of radioactive material produced to that which is needed for successful completion of the experiment or operation.
8. Engineering controls that limit personnel and public dose.
9. Review of facility changes, e.g. procedures and design changes, license amendments, experiments, by the RHP and RSAC and RSC as stated in TS.
10. Adhering to approved facility procedures, radioactive material authorizations (RMA), and other radiological controls (e.g. radiation work permits (RWP)).
11. Access to the facility, radioactive sources, and experiments limited to authorized, trained personnel.

RMA are used at the reactor facility for conducting experiments as approved by RSAC and RSC. Limitations and conditions are established on activity, dose rates, handling, storage, monitoring, and waste disposal. The experimenter, reactor staff, and RHP review and implement the RMA as discussed in TS and facility procedures.

RWP are used at the reactor facility for establishing specific limitations and conditions for specific tasks associated with various work activities and experiments. Requirements for RWP are given in the RP Program and Health Physics Procedures. Personnel with unescorted access are trained on RWP and request RWP. The RHP is responsible for issuing RWP. Personnel performing work are to follow RWP requirements. If the work scope or conditions change, then the RWP is revised by the RHP.

All procedures and design changes and RMA are reviewed by the RHP and RSAC. RMA are also reviewed by the RSC. The NCSU Radiation Safety Officer and RHP both are required members on RSAC and RSC. Every occurrence of an experiment is reviewed by the RHP and reactor staff for compliance with RMA and facility procedures and reactor license requirements, including TS. Experimenters must be qualified to use the experimental equipment. In addition, for more complicated work approved experimental procedures are established. Appropriate radiological controls are discussed and included in documents affecting work activities and experiments at the reactor facility. All of these controls and documents include implementation of ALARA. Section 12 provides information on audits and reviews. Section 11 provides information on experiments.

Administrative dose and ALARA dose goals are established in the RP Program and NCSU Radiation Safety Manual. These doses are at a small fraction of the limits given in 10 CFR Part 20, typically 10 percent or less. These doses may be increased for select individuals performing specialized tasks with approval by the individual, their supervisor, and the RHP. If the ALARA goals are not met, then an investigation is performed to determine if the ALARA concepts were followed and/or if improvements are needed.

11.1.4. Radiation Monitoring and Surveying

Radiation monitoring is performed at the reactor facility by the radiation monitoring system (RMS), portable instruments, and air sampling and analyses.[11-1,11-14] Contamination surveys, personnel radiation monitoring, and environmental monitoring are also performed as described in Section 11.1.5 through 11.1.7.

Radiation Monitoring System (RMS)

RMS details are given in the following tables taken from Section 7.7. Setpoints of RMS are described later in this section. As noted in Section 7.7:

1. The radiation monitoring system (RMS) is composed of required and optional process radiation monitors. RMS radiation monitors are located throughout the reactor building and in or near reactor systems and components that are radiologically significant. All monitors are capable of measuring from background radiation levels up to anticipated accident radiation levels.
2. There are five RMS channels required for reactor operation. All of the required RMS radiation monitors are part of the radiation alarm panel in the control room. Each has two setpoints; warning and alarm. A warning generates an annunciation on the control console to notify the reactor operator of a potential problem. An alarm will automatically initiate evacuation and place the reactor building into confinement. The evacuation system warns personnel about abnormal radiation levels requiring evacuation of the reactor building.
3. Process RMS channels are not required for reactor operation, i.e. optional, but typically are in service. Process RMS channels are used for measuring radioactivity or radiation levels in reactor systems, reactor components, and/or experiments. Experiment area radiation monitors are used as specified in the radiation protection program, radioactive material authorizations (RMA), radiation work permits (RWP), and facility procedures. Access controls for entry into high radiation areas (HRA) required by 10 CFR Part 20 may be met using process RMS, e.g. by interlocks on doors, shutters, and exposure to irradiated samples. If so, these channels are displayed in the control room. Should an entry be made into a HRA that is interlocked by a RMS channel setpoint, control room annunciation occurs. The annunciation signal originates from the HRA entry door or gate being opened. HRA access controls are associated with the process RMS for the delay tanks, demineralizer, PN sample receiver, and some experiment areas.
4. The stack exhaust and CAM may be used as substitutes for the stack gas and stack particulate channels, respectively, provided that the warning setpoint provides control room annunciation and the alarm output automatically initiates evacuation and places the reactor building into confinement. Annunciation notifies the reactor operator of a potential problem and the evacuation system requires personnel to evacuate the reactor building due to abnormal radiation levels.
5. The stack exhaust monitor is a process RMS channel that is typically in service and functions as a redundant channel to the stack gas monitor.
6. The CAM samples unfiltered reactor building air for particulates. If the reactor building is

in normal ventilation, the stack particulate monitor and CAM are sampling the same air. If the reactor building is in confinement mode, the exhausted air is filtered. The CAM alarm setpoints for occupational levels are less than those for effluent levels due to filtration, if in confinement mode, and atmospheric dispersion.

7. Facility status and/or personnel radiation exposure are provided by RMS channels. Operator response to abnormal reading, annunciators, RMS channel warning or alarm, and RMS malfunction are specified in reactor operating procedures. Facility procedures also include notification of the Reactor Health Physicist and designated senior reactor operator, as needed.

Details on the RMS channels as shown below:

Table 11-19 – Required RMS Channels

RMS Channel	Radiation Detected	Range		Reading Display
Control Room	Gamma	1×10^{-2} mrem/h	1×10^{6} mrem/h	Control room, optional local display
Over the Pool	Gamma	1×10^{-2} mrem/h	1×10^{6} mrem/h	Control room, optional local display
West Wall	Gamma	1×10^{-2} mrem/h	1×10^{6} mrem/h	Control room, optional local display
Stack Gas	Beta or Gamma	6×10^{1} cpm	6×10^{6} cpm	Control room, optional local display
Stack Particulate	Beta or Gamma	6×10^{1} cpm	6×10^{6} cpm	Control room, optional local display

Table 11-20 – Setpoints for Required RMS Channels

<table>
<tr><th rowspan="2">RMS Channel</th><th colspan="2">Setpoint[a]</th><th rowspan="2">Notes</th></tr>
<tr><th>Warning</th><th>Alarm</th></tr>
<tr><td>Control Room</td><td>2 mrem/h</td><td>5 mrem/h</td><td rowspan="3">2 mrem/h is the public dose rate limit
5 mrem/h is the radiation area dose rate limit
100 mrem/h is the high radiation area dose rate limit
Refer to Section 11.1.4 for setpoint bases.</td></tr>
<tr><td>Over the Pool</td><td>5 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>West Wall</td><td>5 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>Stack Gas</td><td>3000 EC</td><td>3500 EC</td><td rowspan="2">EC is the airborne concentration taken from 10 CFR Part 20 Appendix B Table 2
Refer to Section 11.1.4 for setpoint bases.</td></tr>
<tr><td>Stack Particulate</td><td>8000 EC</td><td>8500 EC</td></tr>
</table>

[a] Setpoints listed are the typical setpoints for the channel. Setpoints may be set at more conservative values.

Table 11-21 – Process RMS Channels

Process RMS Channel	Radiation Detected	Range		Reading Display
Heat Exchanger (HXCH)	Gamma	1×10^{-2} mrem/h	1×10^{6} mrem/h	Control Room, optional local display
Delay Tanks (Delay)	Gamma	1×10^{-2} mrem/h	1×10^{6} mrem/h	Control Room, optional local display
Demineralizer (Demin)	Gamma	1×10^{-2} mrem/h	1×10^{6} mrem/h	Control Room, optional local display
Pneumatic (PN) Sample Return	Gamma	1×10^{-2} mrem/h	1×10^{6} mrem/h	Control Room and local display
Waste Tank Vault (WTV)	Gamma	1×10^{-2} mrem/h	1×10^{6} mrem/h	Control Room, optional local display
Experiment	Gamma or Neutron	1×10^{-2} mrem/h	1×10^{6} mrem/h	Control Room and local display
Recirculation Air Monitor (RCM)	Beta or Gamma	6×10^{1} cpm	6×10^{6} cpm	Control Room, optional local display
Stack Exhaust	Beta or Gamma	6×10^{1} cpm	6×10^{6} cpm	Control Room, optional local display
Constant Air Monitor (CAM)	Beta or Gamma	6×10^{1} cpm	6×10^{6} cpm	Control Room, optional local display

Detection capabilities and ranges are listed in Tables 11-18 and 11-20. Ion chambers or Geiger-Mueller (GM) detectors are used for area monitors. GM or scintillation detectors are used for ventilation system monitors.

TS channel calibrations are performed following TS surveillance requirements. Instruments are calibrated using facility procedures consistent with vendor manuals.

Table 11-22 – Setpoints for Process RMS Channels

<table>
<tr><th rowspan="2">Process RMS Channel</th><th colspan="2">Setpoint[a]</th><th rowspan="2">Notes</th></tr>
<tr><th>Warning</th><th>Alarm</th></tr>
<tr><td>HXCH</td><td>2 mrem/h</td><td>5 mrem/h</td><td rowspan="6">2 mrem/h is the public dose rate limit
5 mrem/h is the radiation area dose rate limit
100 mrem/h is the high radiation area dose rate limit
Refer to Section 11.1.4 for setpoint bases.</td></tr>
<tr><td>Delay</td><td>5 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>Demin</td><td>5 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>PN Sample Receiver</td><td>100 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>WTV</td><td>0.2 mrem/h</td><td>2 mrem/h</td></tr>
<tr><td>Experiment</td><td>100 mrem/h</td><td>100 mrem/h</td></tr>
<tr><td>RCM</td><td>3 DAC</td><td>10 DAC</td><td rowspan="3">EC is the airborne concentration is given in 10 CFR Part 20 Appendix B Table 2.
DAC is the airborne con centration given in 10 CFR Part 20 Appendix B Table 1.
Refer to Section 11.1.4 for setpoint bases.</td></tr>
<tr><td>Stack Exhaust</td><td>3000 EC</td><td>3300 EC</td></tr>
<tr><td>CAM</td><td>0.3 DAC</td><td>1 DAC</td></tr>
</table>

[a] Setpoints listed are the typical setpoints for the channel. Setpoints may be set at more conservative values.

Table 11-23 – Locations for RMS and Process RMS Channels

Channel	Location
Control Room	At the Control Room window overlooking the reactor bay
Over the Pool	Approximately at elevation [redacted] over the top of the reactor pool
West Wall	Approximately at elevation [redacted] on the west wall of the reactor building within 20 feet of the new fuel storage rack
Stack Gas	Inside and off-line, shielded chamber located in the ventilation equipment room above the control room
Stack Particulate	Inside and off-line, shielded chamber located in the ventilation equipment room above the control room
Stack Exhaust	Inside, or adjacent to, the stack exhaust and downstream from the confinement filters in the ventilation equipment room above the control room
CAM	Inside the reactor building approximately at elevation [redacted] within 20 feet of the top of the reactor pool
RCM	Inside, or adjacent to, the ventilation system air recirculation duct in the ventilation equipment room above the control room
Heat Exchanger	Near the heat exchanger in the mechanical equipment room.
Demineralizer	Near the primary coolant demineralizer entry point.
Delay Tanks	Near the primary coolant entry point to the first delay tank.
PN Sample Receiver	Near the PN sample return location.
Waste Tank Vault	In or above the waste tank vault.
Experiment	In close proximity to reactor experiments inside the reactor building, associated laboratories, or other areas within the site boundary.

Air Effluent Sampling System

The air effluent sampling system is comprised of (1) isokinetic sampling probes and stainless steel tubing, (2) a ventilation mode alignment switch, (3) an isokinetic splitter tube, (4) a stack particulate fixed filter unit and the stack particulate radiation detector, (5) the stack gas chamber and the stack gas radiation detector, and (6) the stack sample pump. See Figure 11-14.

Air samples are taken using isokinetic sample probes located in the normal and confinement ventilation system ducts located in the ventilation equipment room above the control room for the stack gas and stack particulate monitors.

The normal ventilation duct becomes the exhaust duct that exits the reactor building and connects to the exhaust stack. The confinement duct merges with the exhaust duct. The exhaust duct exits the reactor building to the outdoors, passes through the outer stack, and connects to the inner stack at the third floor elevation of the BEL (approximately [redacted]).

The normal and confinement ventilation ducts are rectangular shaped and are of different sizes with different volumetric and linear flow rates. The normal ventilation exhaust duct has a flow rate of 1870 cfm and the confinement duct has a flow rate 600 cfm. Each isokinetic sample probe was sized for the duct flow rate and dimensions at a sampling flow rate of 1 to 4 cfm.

The sample probe in the normal ventilation duct is located in the last straight section before becoming the exhaust duct. The sample probe in the confinement duct is located in a straight section of ductwork after the confinement filters. Both locations are in the last section of straight ductwork before entering the exhaust duct. These locations were selected based on 40 CFR Part 60 Appendix 1-A Method 1A for an alternative location and for the purposes of maintenance, inspection, and protection. 40 CFR Part 60 Appendix 1-A Method 1A, *Sample and Velocity Traverses for Stationary Sources With Small Stacks or Ducts*, states that for particulate measurements with steady or unsteady flow an alternative particulate measurement location at least two equivalent stack or duct diameters downstream and two and one-half diameters upstream from any flow disturbance may be used. 40 CFR Part 60 Appendix A Section 12.2 of Method 1 was used for calculating equivalent diameters for a rectangular cross-section.

A sampling switch automatically aligns the sample probe to be used based on the ventilation mode; normal or confinement. Stainless steel tubing was used to avoid a static charge buildup that would affect particulate sampling.

An isokinetic splitter tube rated for the sample flow rate from 1 to 4 cfm allows for the sampled air sample to be split between the stack particulate monitor and a by-pass line. The bypass line and stack particulate monitor line each have isolation valves and a flow control valve. A filter housing is used in each line that accommodates a particulate filter and a cartridge in series. The cartridge may be used to sample for radio-iodine and radio-bromine. Change out of the stack particulate filter during reactor operation may be made using the bypass line and stack particulate line isolation valves.

The stack particulate line and bypass line merge and then connect to the stack gas chamber. The gas chamber is shielded and houses the stack gas radiation detector. Air enters and exits the gas chamber on opposite sides to allow for air movement through the gas chamber. The exit line from the gas chamber is connected to the sample pump.

A regulated air sample pump is used to obtain the air sample. The sample pump has a flow rate control valve and provides flow rate status to the control room. If the flow rate is low or stops, an annunciator sounds and illuminates in the control room. Sample flow rate is set during calibration to that required for the stack particulate alarm setpoint. The sample flow alarm is checked on each filter change out, which occurs at periods not to exceed 10 days. Exhausted air from the sample pump is returned to the exhaust duct downstream of the sampling point.

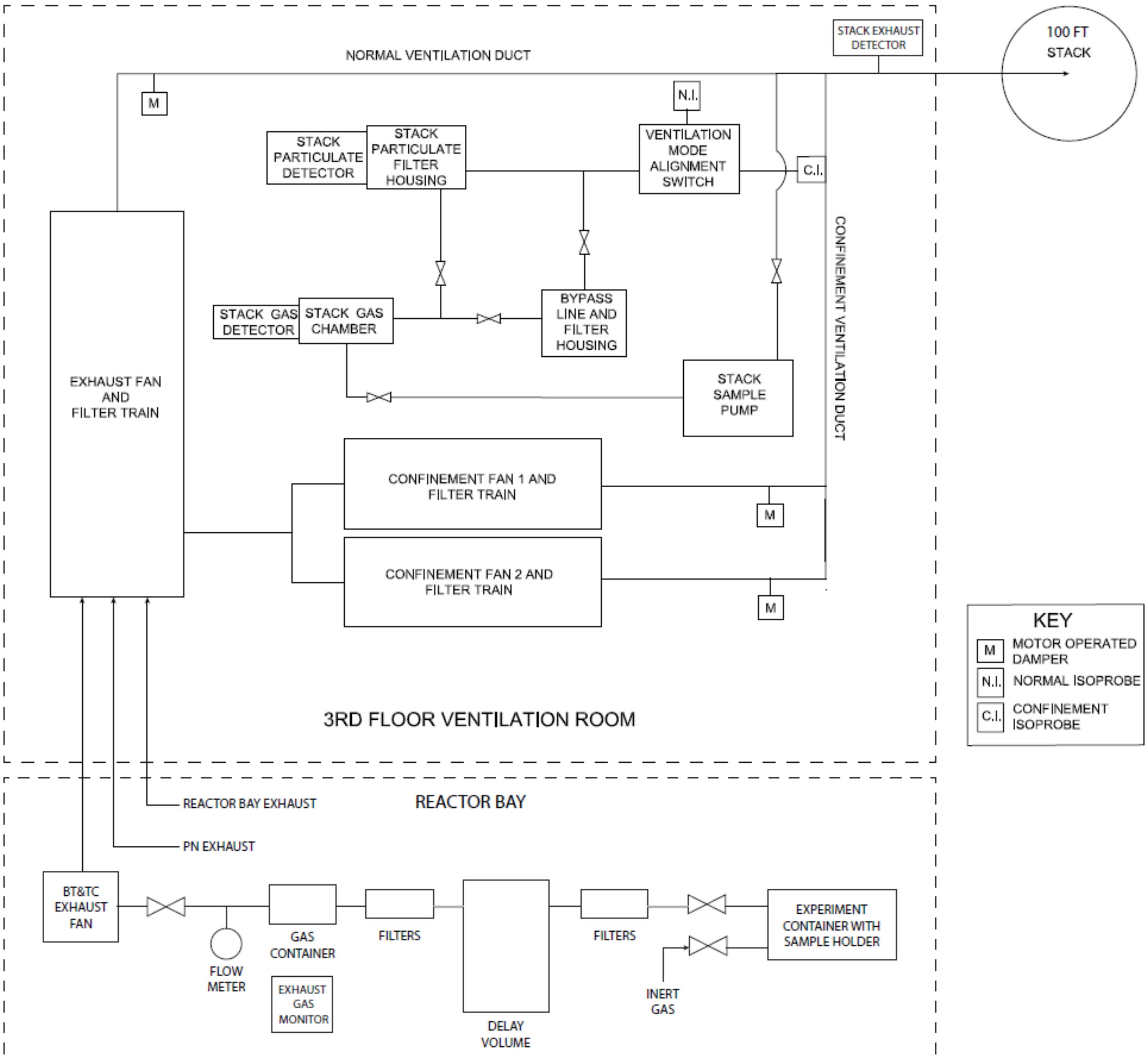


Figure 11-14 – Air Effluent Sampling System

Table 11-24 – Radiation Monitor Setpoints and Alarm Actions

RMS Channel	Alert Setpoint[a]	Alarm Setpoint[a]	Alert Action	Alarm Action	Units	Notes
Control Room	2	5	Control Room Annunciation	Evacuation Alarm	mrem/h	1,2
Over the Pool	5	100	Control Room Annunciation	Evacuation Alarm	mrem/h	2,3
West Wall	5	100	Control Room Annunciation	Evacuation Alarm	mrem/h	2,3,4
Stack Gas	3000 EC	3300 EC	Control Room Annunciation	Evacuation Alarm	cpm	5,7,10,11
Stack Particulate	8000 EC	8500 EC	Control Room Annunciation	Evacuation Alarm	cpm	6,7,10,11
Stack Exhaust	3000 EC	3300 EC	Control Room Annunciation	Evacuation Alarm	cpm	5,7,10,11,12

Process RMS Channel	Alert Setpoint[a]	Alarm Setpoint[a]	Alert Action	Alarm Action	Units	Notes
Constant Air Monitor (CAM)	0.3 DAC	1 DAC	Local audible and/or visible alarm. Control room annunciation as needed.	Local audible and/or visible alarm, evacuation alarm actuation as needed	cpm	9,11,13
Recirculation (RCM)	3 DAC	10 DAC	Control Room Annunciation		cpm	8,11
Demineralizer	5	100	Local audible and/or visible alarm	Lock HRA Gate to PPV, equipment actuation as needed	mrem/h	2,3
Delay Tanks	5	100	Local audible and/or visible alarm	Lock HRA Gate to PPV, equipment actuation as needed	mrem/h	2,3
Waste Tank Vault	0.1	0.2	Local visible alarm	Control room annunciation	mrem/h	
Experiment	2 to 5	5 to 100	Local audible and visible alarm	Local audible and visible alarm, equipment actuation as needed	mrem/h	1,2,3

Notes

1 Public dose rate limit is 2 mrem/h
2 Radiation area is defined at 5 to 100 mrem/h
3 High radiation area (HRA) is defined at > 100 mrem/h
4 Criticality monitor for new fuel storage as required by 10 CFR Part 70.24
5 Based on ^{41}Ar EC. Meets emergency action level (EAL) criteria and fueled experiment accident release limit.

[a] Setpoints listed are the typical setpoints for the channel. Setpoints may be set at more conservative values.

6 Based on ^{60}Co EC. Meets EAL criteria and fueled experiment accident release limit.
7 Effluent concentration (EC) and derived air concentration (DAC) as listed in 10 CFR Part 20 Appendix B
8 Based on 0.3 derived air concentration (DAC) fraction for ^{41}Ar adjusted for room volume
9 Based on 0.3 derived air concentration (DAC) fraction for ^{60}Co
10 X/Q for emergency is 0.00854 s/m^3
11 cpm is converted from detector efficiency
12 Stack exhaust monitor is used as a substitute for the stack gas monitor
13 Equipment actuation for HRA control, access control, airborne activity etc. may include interlocks and / or control room annunciation

Activity or concentration are converted to a count rate (cpm) based on the detector efficiency and assumed radionuclide(s) present. For routine operations, the stack gas, stack exhaust, and RCM are based on ^{41}Ar and the stack particulate and CAM are based on ^{60}Co. These radionuclides are considered limiting due to abundance and regulatory limits given for occupational and public dose in 10 CFR Part 20 Appendix B.

The CAM and RCM respond to airborne activity inside the reactor building. Air monitors inside the reactor building are based on occupational dose considerations. Airborne activity areas are defined at 0.3 DAC, or 12 DAC-h in a 40 hour week. Peak airborne activity concentrations are not to exceed 1 DAC per 10 CFR Part 20. The RCM setpoint includes adjustment for the reactor building volume to account for partial energy-spatial equilibrium from a finite cloud of ^{41}Ar, which is approximately a factor of 10.

Effluent monitors (stack gas, stack exhaust) setpoints depend on the EC for the radionuclide being released.

Experiment Monitors

Experiment monitors provide display of readings and local alarms and may be used to actuate equipment, e.g. door interlocks or shutter interlocks or control room annunciators or other alarm indication (lights, beacons, illuminated warning signage).

Stack Gas and Stack Exhaust Radiation Monitor Setpoints

The alarm setpoint for the stack gas and stack exhaust radiation monitors are based on the more limiting of an Emergency Action Level (EAL) of 50 EC for 24 hours (1200 EC-h)[11-35] giving approximately 15 mrem or an abnormal release for fueled experiments. The alert setpoint is based on 10 mrem in 24 hours or an abnormal release below the alarm level for fueled experiments.

Ar-41 is commonly released from routine reactor operations due to the activation of air in reactor systems and experiments and is used as the reference radionuclide for the stack gas radiation monitor setpoints. Ar-41 has an EC of 1×10^{-8} µCi/ml.

Using the limiting [X/Q] value of approximately 8.54×10^{-3} s/m^3 for occupied areas over 24 hours gives the stack gas monitor setpoints for Ar-41:

$$\text{EAL based alarm:} \quad 6631 \text{ EC} = \frac{50 \text{ EC}}{[(0.00854 \text{ s/m}^3)(0.883 \text{ m}^3/\text{s})]} \qquad \textit{Equation 11-50}$$

$$\text{EAL based alert:} \quad 4420 \text{ EC} = 6631 \text{ EC } \left[\frac{10 \text{ mrem}}{15 \text{ mrem}}\right] \qquad \textit{Equation 11-51}$$

For Ar-41, the setpoint concentrations are approximately 6.6×10^{-5} µCi/ml for the alarm and 4.4×10^{-5} µCi/ml for the alert.

For fueled experiments, fission gas releases may occur at a concentration in the reactor stack of 3.35×10^{-5}µCi/ml. The EC for the mixture of fission gases released is approximately 4×10^{-8} µCi/ml. This gives a concentration of 837 EC as the alarm setpoint, which has a 24 hour dose of approximately 2 mrem:

$$\text{Fueled experiment alarm: } 837 \text{ EC} = [\frac{3.35 \times 10^{-5}}{4 \times 10^{-8} \text{ per EC}}]$$ ***Equation 11-52***

$$\text{Fueled experiment 24 h dose: } 1.9 \text{ mrem} = 15 \text{ mrem } [\frac{837 \text{ EC}}{6631 \text{ EC}}]$$ ***Equation 11-53***

Adjustment of the fueled experiment alarm setpoint is made for fueled experiments to obtain an equivalent concentration of Ar-41:

$$\text{Ar} - 41 \text{ equivalent alarm: } 3300 \text{ EC} = 837 \text{ EC } [\frac{4 \times 10^{-8}}{1 \times 10^{-8}}]$$ ***Equation 11-54***

The alert setpoint for fission gases is based on a lower release concentration of 3×10^{-5}µCi/ml, which is equivalent to 750 EC. Adjustment is made to an equivalent Ar-41 concentration:

$$\text{Ar} - 41 \text{ equivalent alert: } 3000 \text{ EC} = 750 \text{ EC } [\frac{4 \times 10^{-8}}{1 \times 10^{-8}}]$$ ***Equation 11-55***

From the above discussion, the alarm setpoint is 3300 EC and the alert setpoint is 3000 EC based on Ar-41 concentration.

The stack gas and stack exhaust monitors sample air flowing in a small volume, e.g. sample chamber or air duct. Setpoints are based on activity concentration as follows:

$$setpoint\ in\ cpm = C\epsilon$$ ***Equation 11-56***

where, C is the setpoint concentration in µCi/ml

ε is the monitor efficiency in cpm per µCi/ml

Stack Particulate Radiation Monitor and Constant Air Monitor (CAM) Setpoints:

Stack Particulate Radiation Monitor

The alarm setpoint for the stack particulate radiation monitor is based on the more limiting of a EAL of 100 EC for 24 hours (2400 EC-h)[11-35]giving approximately 15 mrem to an adult or an abnormal release for fueled experiments. The alert setpoint is based on 10 mrem to an adult in 24 hours or an abnormal release below the alarm level for fueled experiments.

Co-60 is commonly released from routine reactor operations due to the activation of components in reactor systems and experiments and is used as the reference radionuclide for the stack particulate radiation monitor setpoints. Co-60 has an EC of 5×10^{-11} µCi/ml.

Using the limiting [X/Q] value of $8.54x10^{-3}$ s/m^3 for occupied areas over 24 hours gives the stack particulate monitor setpoints for Co-60:

$$\text{EAL based alarm:}\quad 13{,}261\ \text{EC} = \frac{100\ \text{EC}}{[(0.00854\,\text{s/m}^3)(0.883\,\text{m}^3/\text{s})]} \qquad \textit{Equation 11-57}$$

$$\text{EAL based alert:}\quad 8840\ \text{EC} = 13{,}261\ \text{EC}[\frac{10\ \text{mrem}}{15\ \text{mrem}}] \qquad \textit{Equation 11-58}$$

Co-60, the setpoint concentrations are approximately $6.6x10^{-7}$ µCi/ml for the alarm and $4.4x10^{-7}$ µCi/ml for the alert.

For fueled experiments, particulate releases are not expected but halogen releases may occur. The stack particulate monitor is equipped with both particulate and radioiodine air sample media. Activity deposited on the sample media would be detected.

For fueled experiments, the concentration of halogens released at a release fraction of 10 percent is $4.3x10^{-7}$ µCi/ml. The EC for the mixture of halogens released is approximately $1x10^{-9}$ µCi/ml. The release fraction of 10 percent allows for early detection of halogens. The alarm setpoint is based on 430 EC and a 24 hour dose of 0.5 mrem to an adult:

$$\text{Fueled experiment alarm:}\ 430\ \text{EC} = [\frac{4.3\times10^{-7}}{1\ \times10^{-9}\ \text{per EC}}] \qquad \textit{Equation 11-59}$$

$$\text{Fueled experiment 24 h dose:}\quad 0.5\ \text{mrem} = 15\ \text{mrem}\ [\frac{430\ \text{EC}}{13{,}261\ \text{EC}}] \qquad \textit{Equation 11-60}$$

Adjustment of the fueled experiment alarm setpoint is made for fueled experiments to obtain an equivalent Co-60 concentration:

$$\text{Co} - 60\ \text{equivalent alarm:}\ 8600\ \text{EC} =\ 430\ \text{EC}\ [\frac{1\times10^{-9}}{5\ \times10^{-11}}] \qquad \textit{Equation 11-61}$$

The alert setpoint for halogens is based on a concentration of $4x10^{-7}$µCi/ml. This gives an alert setpoint at 400 EC with a Co-60 equivalent alert setpoint of 8000 EC:

$$\text{Co} - 60\ \text{equivalent alert:}\ 8000\ \text{EC} =\ 400\ \text{EC}\ [\frac{1\times10^{-9}}{5\ \times10^{-11}}] \qquad \textit{Equation 11-62}$$

From the above discussion, the alarm setpoint is 8500 EC of Co-60 and the alert setpoint is 8000 EC of Co-60.

CAM

The alert and alarm setpoint for the CAM is based on 0.3 DAC (Derived Air Concentrations) and 1 DAC respectively. The CAM has a particulate air filter and may also be used to monitor for radioiodine activity. Co-60 is limiting for particulate activity and I-131 is limiting for radioiodine activity.

Setpoint Determination

The stack particulate and CAM monitors sample air using fixed filters. Setpoints for monitors with fixed filter sampling are based on the following:

1. Low sample flow rate from 1 cfm to 4 cfm. A regulated sample air pump is commonly used for air sampling. The sample flow rate is recorded either at the start and end of sampling or by a recorder. The low flow rate is insignificant compared to the reactor building ventilation rates thereby providing a representative sample.
2. Sampling time of 1 hour or more. 1 hour or longer is recommended by NUREG 1400[11-12]. Setpoints are based on a sampling time from 1 hour to 24 hours. 24 hours allows time for a representative sample to be obtained and monitored while providing early indication of abnormal airborne activity so that corrective actions may be taken.
3. Constant and uniform filter retention. This is achieved by a steady sample flow rate and low filter loading. Low filter loading minimizes changes in filter retention and detector efficiency. Typical sampling times of one week (7 to 10 days) for normal conditions in the reactor building do not cause significant filter loading.

Setpoints are calculated as follows for the stack particulate radiation monitor and CAM:

$$setpoint\ in\ count\ rate = \frac{CFK\epsilon R\left[1 - e^{-\lambda T}\right]}{\lambda}$$ ***Equation 11-63***

where,

C is the concentration in $\mu Ci/ml$, which is the EC for the stack particulate monitor and DAC for the CAM

F is the sample flow rate in ml/min

K is a conversion constant of 2.22×10^{6} $dpm/\mu Ci$

ϵ is the detector efficiency in c/d

$\left[1 - e^{-\lambda T}\right]$ corrects for decay while sampling

λ is the radioactive decay constant in min^{-1}

$(1/\lambda)$ is the mean-life of the sampled radionuclide in minutes

T is the sampling time in minutes

R is the filter retention

The setpoint count rate may be in the units of counts per minute (cpm) or converted to other count rates, e.g. counts per second (cps).

RCM Setpoint:

The RCM monitors air in the reactor building, primarily Ar-41, circulating in the reactor building air space. The RCM detector is placed located near the intake to the recirculation duct from the reactor building. This location is away from other radiation sources and improves sensitivity to changing airborne activity conditions due to the recirculation duct flow rate (approximately 5600 cfm). Air taken from the reactor building is mixed with a lesser amount of outside air before being recirculated back to the reactor building. ^{41}Ar in the reactor building air space is assumed to be uniformly dispersed in the reactor free air volume.

The reactor building presents finite boundaries for ^{41}Ar gamma photons. Corrections of the RCM response (R) to the finite cloud of ^{41}Ar inside the reactor building is made using the following equations:

$$R_{finite} = fR_{semi-infinite} \quad \text{Equation 11-64}$$

$$f = f'Gk = \mu_{en}RGk \quad \text{Equation 11-65}$$

$$f = (4.92 \times 10^{-5}\ cm^{-1})(944\ cm)(2)(1.1) = 0.1 \quad \text{Equation 11-66}$$

Alternately,

$$f = 2k\left(1 - e^{(-\mu_{en}r)}\right) = (2)(1.1)\left[1 - e^{\left(-4.92\times10^{-5}*944\right)}\right]\sim 0.1 \quad \text{Equation 11-67}$$

1. f' is the ratio of dose from a finite cloud to dose from a semi-infinite cloud given by the product of $\mu_{en}r$.
2. μ_{en} = energy absorption coefficient in air for photons, for photons above 50 keV this value is < 4.92×10^{-5} per cm and ~ 3.17×10^{-5} per cm for ^{41}Ar at 1.293 MeV.
3. r = effective radius of 944 cm based on the reactor building volume of 3.5×10^{9} ml.
4. G = geometry correction factor of 2 for a sphere (4π geometry for personnel at an elevated location) vs. hemisphere (2π geometry for semi-infinite cloud affecting personnel on a lower level surface).
5. k = ratio of mass energy absorption coefficients for tissue to air to convert to tissue dose having a value of approximately 1.1 for photon energies from 50 keV to several MeV.

f has a value of approximately 0.1 or less and is applied to the RCM response. RCM alert and alarm setpoints of 0.3 DAC and 1 DAC are changed using the factor of (1/f) to obtain 3 DAC and 10 DAC giving a correction for the finite cloud of ^{41}Ar.

Measurements of the reactor building experimental area were made giving a total volume of 3.5×10^{9} ml. The radius of an equivalent sphere, r is calculated as follows:

$$V = \left({}^{4}/_{3}\right)\pi r^{3} = 3.5 \times 10^{9}\ ml \quad \text{Equation 11-68}$$

$$r = 944\ cm$$

Free volume was measured to be 3.1×10^9 ml by accounting for existing equipment and experiments. A more limiting free volume of 2.4×10^9 ml is used to allow for additional reactor and experimental equipment in the future.

Setpoints for the RCM are based on the submersion dose rate from the finite cloud of ^{41}Ar in the reactor building. At 10 DAC and 3 DAC the submersion dose rate inside the reactor building is 2.5 mrem/h and 0.75 mrem/h, respectively. The RCM alarm and alert setpoint count rates are at the radiation dose rates of 2.5 mrem/h and 0.75 mrem/h, respectively. A conversion factor giving the count rate per unit dose rate for ^{41}Ar determined during calibration of the RCM is used for the setpoints.

Waste Tank Vault Monitor Setpoint

Liquid waste may be monitored using the RMS or by periodically surveying for external gamma radiation levels. Optional process RMS may include the waste tank vault.

Readings above 0.2 mrem/h are attributable to the high activity liquid waste that would require reprocessing, disposal, or removal due to the potential for exceeding the annual public dose limit:

$$0.2\,mrem/h \sim \frac{(100\,mrem/year)}{(8760\ h/1\ year)(1/20)}$$ ***Equation 11-69***

An occupancy factor of (1/20, or 5 percent) of a year is assumed for pedestrian traffic.

Radiological Surveys and Air Sampling

Radiation surveys are performed as specified in the RP Program.[11-1] This includes surveys for area radiation levels, surface contamination, and air contamination surveys performed routinely and as needed for experiments, maintenance, and RWP. Area radiation surveys typically include gamma and neutron radiation levels. Contamination surveys are performed as described in Section 11.1.6. Areas surveyed include the reactor building, associated laboratories and areas outside the reactor building, but within the site boundary. Surveys are performed upon removal of samples from experiments.

Air sampling is performed if the airborne activity may exceed 0.01 DAC or if airborne activity is suspected. Sampling and analyses for tritium and radio-iodine are performed as described in the facility RP Program.

Most areas in the reactor facility are typically below 2 mrem/h with non-detectable surface contamination (< 200 dpm/100 cm^2 beta-gamma), and non-detectable (< 0.01 DAC) particulate airborne activity from radionuclides associated with reactor operation or experiments. Controls as described in the RP Program are followed for areas with higher radiation or contamination levels, including the use of RWP as needed. Abnormal radiation and contamination levels are evaluated to determine the cause and action is taken to reduce levels and to reduce personnel dose, if possible. This may include adding shielding, decontamination, or implementing access controls.

Approved procedures and calibrated equipment are used in the performance of radiation surveys. Records of surveys are maintained as required by facility procedures.

Calibrations and Surveillance

RMS calibrations are performed using approved procedures for the radiation type, energy, and response needed.[11-1,11-14,11-18] Multiple ranges of the instrument that are useful are calibrated, which typically span several orders of magnitude. Calibration error is typically within 10 percent of a NIST traceable source. The RHP may approve calibration errors up to 20 percent with restrictions on use as stated in the RP Program and facility procedures. Calibration and surveillance procedures are based on vendor manuals and applicable ANSI and HPS references and US NRC guidance.

TS monitor channel calibrations and channel checks are performed as stated in TS surveillances.

Optional RMS may be used to monitor experiments and areas or components within the reactor building or reactor site boundary. These radiation monitors are installed and operated using approved procedures.

Portable radiation monitoring equipment and laboratory radiation measuring instruments are maintained and available for use by the RHP and reactor staff and experimenters. These include survey meters, personnel frisking equipment, electronic dosimeters, and laboratory radiation counting equipment.

The RHP is responsible for calibrations and setting and checking all radiation monitor setpoints. RMS are calibrated annually. TS RMS setpoints are checked weekly, not to exceed 10 days. Optional RMS setpoints are checked as specified in facility procedures.

Additional Monitors

In addition, other parameters useful for monitoring radiological conditions may be used as directed by the RHP (e.g. reactor pool resistivity), reactor pool level, ventilation system flow rate, and reactor building differential pressure).[11-1]

11.1.5. Radiation Exposure Control and Dosimetry

Radiation exposure controls are used as established in the RP Program to limit dose to occupational personnel.[11-1] These include the following:

1. Time restrictions, such as access limitation or restriction and stay times.
2. Distance, e.g. by using tape or rope or barricades.
3. Shielding.
4. Limitation of activity handled and stored.
5. Postings of areas as required by 10 CFR Part 20 and the RP Program.
6. Engineering controls, such as interlocks, annunciators and alarms.
7. Administrative controls, such as radioactive material authorizations (RMA).
8. Radiation work permits (RWP) and electronic dosimeters (ED) with alarms and dose limits.
9. Personnel dosimetry, radiation monitoring and surveys.

Radiation levels within the facility generally are below 2 mrem per hour. Higher levels are produced near some reactor equipment and experiments. These areas are controlled to limit access by occupational personnel and to prevent access by visitors. A description and analysis of shielding is given in this section.

High radiation areas (HRA) are controlled using one or more of the access control methods given in 10 CFR Part 20 and as required in the reactor facility RP Program. HRA are associated with the following:

1. Primary piping vault (PPV); the primary coolant purification system demineralizer and the ^{16}N delay tanks are located in the PPV. See Figure 11-16.
2. Beamtubes (BT); BT are associated with reactor experiments, e.g. collimated beams or open BT.
3. Irradiated samples, e.g. from irradiation facilities located in the reactor pool.
4. Reactor components that have been irradiated and are removed from the reactor pool.

Submersion dose from ^{41}Ar is controlled by reducing the amount of air present in reactor systems and beam tubes. Several systems are purged with nitrogen gas or the air space is reduced by use of shielding, e.g. water or concrete or lead.

The submersion dose from ^{16}N is negligible in both natural convection and forced cooling modes of cooling due to sufficient decay time and dilution within the reactor building air volume. Submersion dose from ^{16}N is controlled by limiting power in the natural convection mode of cooling to 0.1 MW.

Contamination controls are used to reduce the potential for internal and external dose associated with airborne activity and personal contamination. Details are given later in Section 11.

Shielding Description and Analysis

Shielding dimensions and materials by of the reactor pool are as follows (See Figure 11-15):

1. Elevation [redacted]: [redacted] of water and [redacted] of barytes concrete.
2. Elevation [redacted]: [redacted] of water and [redacted] of ordinary concrete.
3. Elevation [redacted]: [redacted] of water and [redacted] of ordinary concrete.
4. The pool water level is typically above elevation [redacted] and is full at elevation [redacted]. Safety System Setting is at elevation [redacted]. Maximum water level is at elevation [redacted]. Reactor core is at elevation [redacted]. Reactor fuel is stored at elevation [redacted] and at elevation [redacted].
5. The thermal column (TC) has been modified for a planned experiment. The graphite has been removed and is replaced with various shielding materials, including borated polyethylene and concrete. The TC nose piece has been removed and is replaced with water or a water and graphite fixture depending on experiment needs. The TC door has been modified with penetrations for experimental equipment. Additional shielding outside the TC may be installed.

[redacted]

Figure 11-15 – Reactor Elevation View

[redacted]

Figure 11-16 – Reactor Plan View – Elevation [redacted]

Beamtubes contain either experimental equipment with collimators and filters or shielding (water, concrete, lead). Beamtubes in use may have additional external shielding and access controls. Beamtube shield plugs are made of aluminum encased lead and concrete ranging in length up to 12 inches. The beamtube shield plugs match the stepped diameter of the beamtube. The shield plugs are installed to reduce dose rates emanating from unused beam tubes to acceptable levels.

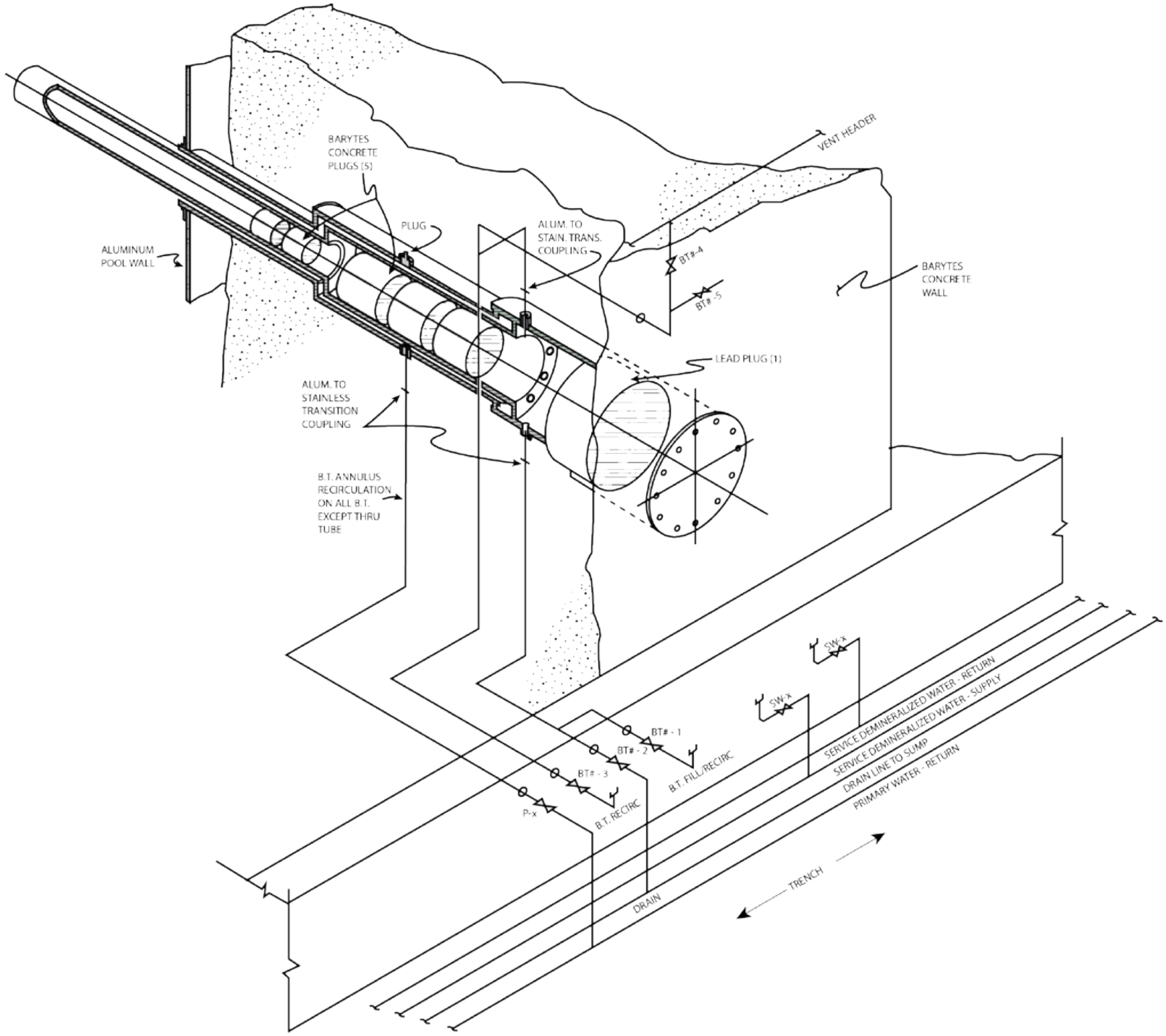


Figure 11-17 – Beamtube (BT) – Typical

[redacted]

Figure 11-18 – Beamtube Dimensions

External Radiation Dose Rate from Natural Convection Cooling

^{16}N is produced in significant amounts during natural convection cooling. Reactor power of 0.1 MW and 0.25 MW are analyzed in this section using data given in Section 11.1.1.1 for ^{16}N concentration leaving the reactor core and upward flow rate in natural convection cooling. In this analysis, the reactor pool is modeled as stacked segments in one foot increments to calculate the dose rate at the top of the reactor pool at elevation [redacted] (1 foot above the full pool level). The dimensions of the segment used were those of the reactor for the area, 15 inches by 15.875 inches, and a 12 inch depth. The segments above the one being calculated are used as shielding. Volume of each segment is 4.68×10^4 ml.

The initial concentration entering the first segment, C(0), was estimated from the concentration in the reactor core adjusted for the volume in the reactor to that for the segment. C(0) was calculated using C(rx) from Equation 11-7 and Table 11-3 for 0.1 MW and 0.25 MW.

$$C(0) = \frac{C(rx)\mu Ci/ml(10.7\ gal)(3785\, ml/gal)}{(4.68\times10^4\ ml)} \qquad \textit{Equation 11-70}$$

Entry concentration, C_{entry}, and average concentration, C_{ave}, in the segment are based on the decay occurring during transit. Since the segment dimensions and flow rate are fixed, the average concentration is a fixed fraction of C_{entry}. C_{entry} and C_{ave} for the segments were calculated for 0.1 MW and 0.25 MW. C_{ave} was 0.654 C_{entry} at 0.1 MW and 0.734 C_{entry} at 0.25 MW.

$$C_{ave} = C_{entry}\int e^{-\lambda\Delta t}dt = C_{entry}\left[\frac{1-e^{\lambda\Delta t}}{\lambda\Delta t}\right] \qquad \textit{Equation 11-71}$$

Where ΔT is the transit time for segment.

Results in mrem/h at elevation [redacted] for all segments at reactor pool levels of 20 feet, 17 feet, and 14 feet are shown in Table 11-25 and Table 11-26. If an individual were there for an entire work year of 2000 hours, the accumulated dose would be approximately 1 mrem, 7 mrem, and 50 mrem for pool levels of 20 feet, 17 feet, and 14 feet, respectively.

Table 11-25 – ^{16}N Pool Dose Rate – Natural Convection Cooling for Pool Height of 20 feet

Segment Height ft	mR/h per µCi/ml at elevation [redacted]	0.1 MW				0.25 MW			
		time sec	Entry C(0)exp(-λt)	Average µCi/ml	Segment mrem/h	time sec	Entry C(0)exp(-λt)	Average µCi/ml	Segment mrem/h
Air space of 1 foot above pool height of 20 feet									
20	2.68E+02	188.82	2.75E-07	4.51E-07	1.21E-04	134.31	1.19E-04	1.68E-04	4.52E-02
19	5.63E+01	179.38	6.89E-07	1.13E-06	6.36E-05	127.59	2.29E-04	3.24E-04	1.82E-02
18	1.54E+01	169.94	1.73E-06	2.83E-06	4.35E-05	120.88	4.41E-04	6.22E-04	9.57E-03
17	4.78E+00	160.50	4.32E-06	7.08E-06	3.38E-05	114.16	8.46E-04	1.19E-03	5.70E-03
16	1.60E+00	151.06	1.08E-05	1.77E-05	2.84E-05	107.45	1.63E-03	2.29E-03	3.68E-03
15	5.65E-01	141.62	2.71E-05	4.43E-05	2.51E-05	100.73	3.12E-03	4.40E-03	2.49E-03
14	2.07E-01	132.18	6.78E-05	1.11E-04	2.29E-05	94.02	6.00E-03	8.46E-03	1.75E-03
13	7.74E-02	122.74	1.70E-04	2.78E-04	2.15E-05	87.30	1.15E-02	1.62E-02	1.26E-03
12	2.95E-02	113.29	4.25E-04	6.96E-04	2.05E-05	80.59	2.21E-02	3.12E-02	9.20E-04
11	1.14E-02	103.85	1.06E-03	1.74E-03	1.99E-05	73.87	4.25E-02	5.99E-02	6.84E-04
10	4.48E-03	94.41	2.66E-03	4.36E-03	1.95E-05	67.15	8.16E-02	1.15E-01	5.16E-04
9	1.78E-03	84.97	6.66E-03	1.09E-02	1.94E-05	60.44	1.57E-01	2.21E-01	3.93E-04
8	7.09E-04	75.53	1.67E-02	2.73E-02	1.94E-05	53.72	3.01E-01	4.25E-01	3.01E-04
7	2.85E-04	66.09	4.18E-02	6.84E-02	1.95E-05	47.01	5.78E-01	8.16E-01	2.33E-04
6	1.15E-04	56.65	1.05E-01	1.71E-01	1.97E-05	40.29	1.11E+00	1.57E+00	1.81E-04
5	4.71E-05	47.21	2.62E-01	4.29E-01	2.02E-05	33.58	2.13E+00	3.01E+00	1.42E-04
4	1.93E-05	37.76	6.55E-01	1.07E+00	2.08E-05	26.86	4.10E+00	5.78E+00	1.12E-04
3	7.93E-06	28.32	1.64E+00	2.69E+00	2.13E-05	20.15	7.87E+00	1.11E+01	8.80E-05
2	3.28E-06	18.88	4.11E+00	6.72E+00	2.20E-05	13.43	1.51E+01	2.13E+01	6.99E-05
1	1.35E-06	9.44	1.03E+01	1.68E+01	2.28E-05	6.72	2.90E+01	4.10E+01	5.54E-05
0			25.73				55.78		
Total					**6.05E-04**				**9.15E-02**
				mrem/y	**1.2**				**183**

Table 11-26 – ^{16}N Pool Dose Rate – Natural Convection Cooling for Various Pool Heights

Pool Height of 17 feet				Pool Height of 14 feet			
Water Shield	**mR/h per µCi/ml at elevation [redacted]**	**0.1 MW Segment**	**0.25 MW Segment**	**Water**	**mR/h per µCi/ml at elevation [redacted]**	**0.1 MW Segment**	**0.25 MW Segment**
ft		**mrem/h**	**mrem/h**	**ft**		**mrem/h**	**mrem/h**
20	Air space of 4 feet above pool height of 17 feet			20	Air space of 7 feet above pool height of 14 feet		
19				19			
18				18			
17	2.97E+01	2.10E-04	3.55E-02	17			
16	1.16E+01	2.05E-04	2.66E-02	16			
15	4.45E+00	1.97E-04	1.96E-02	15			
14	1.72E+00	1.91E-04	1.45E-02	14	1.06E+01	1.18E-03	8.98E-02
13	6.70E-01	1.86E-04	1.09E-02	13	4.83E+00	1.34E-03	7.85E-02
12	2.64E-01	1.84E-04	8.25E-03	12	2.08E+00	1.44E-03	6.48E-02
11	1.05E-01	1.83E-04	6.31E-03	11	8.74E-01	1.52E-03	5.24E-02
10	4.23E-02	1.84E-04	4.87E-03	10	3.66E-01	1.59E-03	4.21E-02
9	1.70E-02	1.86E-04	3.77E-03	9	1.52E-01	1.66E-03	3.36E-02
8	6.88E-03	1.88E-04	2.92E-03	8	6.35E-02	1.74E-03	2.70E-02
7	2.82E-03	1.93E-04	2.30E-03	7	2.64E-02	1.81E-03	2.16E-02
6	1.15E-03	1.97E-04	1.81E-03	6	1.10E-02	1.88E-03	1.72E-02
5	4.75E-04	2.04E-04	1.43E-03	5	4.61E-03	1.97E-03	1.39E-02
4	1.96E-04	2.10E-04	1.13E-03	4	1.93E-03	2.07E-03	1.11E-02
3	8.11E-05	2.18E-04	9.01E-04	3	8.07E-04	2.17E-03	8.96E-03
2	3.36E-05	2.26E-04	7.17E-04	2	3.38E-04	2.27E-03	7.21E-03
1	1.41E-05	2.37E-04	5.76E-04	1	1.42E-04	2.39E-03	5.82E-03
0				0			
Total		**3.40E-03**	**1.42E-01**	**Total**		**2.50E-02**	**4.74E-01**
	mrem/y	**6.8**	**284**		**mrem/y**	**50**	**948**

Reactor Core Source Term

Equations for six defined gamma energy groups reported for short-lived fission products were used for the reactor source term.[11-25] These equations provide for a wide range of operating and decay times and neglect transmutation of fission products by neutron absorption. Equations are given for an energy release rate in MeV per second for a constant, specified number of fissions per second, Pf(t), for time t_0 prior to reactor shutdown and a shutdown time t_s :

$$\Gamma_j(t_o, t_s) = P_f \sum i\left[\alpha_{ij}/\beta_{ij}\right]\left[e^{-\beta_{ijts}}\right]\left[1 - e^{-\beta_{ijt(0)}}\right] \qquad \textit{Equation 11-72}$$

where,

$\Gamma_j(t_o, t_s)$ is MeV per second for the j[th] energy group

F is related to thermal power by 3.1×10^{10} fissions per watt

Summation is carried out for each energy group using the reported alpha (α) and beta (β) empirical fit parameters.

NOTE: As t_0 becomes large, equilibrium is achieved and conservatively estimates the energy release rate due to an overestimation of gamma radiation from long lived radionuclides.

If t_s is set at near 0, the energy release rate approximates that associated with the fuel for an operating nuclear reactor.

Conditions for 2 MW reactor operation are as follows:

1. Operating time of 9 years.
2. Decay time of 0 seconds for an operating reactor.
3. Fission product results are doubled to account for prompt gamma photons and activated reactor components (control rods, reactor grid structure) for the operating reactor.

An operating time of 9 years at full power of 2 MW was used based on the maximum fuel burnup of 20,000 MWd per MTU given in TS. 0.316 MTU is the initial core loading.

$$9\ years \sim \frac{(20{,}000\ MWD/MTU)(0.316\ MTU)}{(2\ MW)(365\ days\ per\ year)} \qquad \textit{Equation 11-73}$$

Results of the estimated reactor core source term are given in Table 11-27, where pps is photons per second.:

Table 11-27 – Reactor Core Source Term

MeV	Average MeV/photon	Estimated MeV/s	Estimated MeV/s Fraction	Estimated pps Fraction	Estimated pps at 2 MW
0-1	0.3	1.68×10^{17}	4.39×10^{-1}	8.31×10^{-1}	1.12×10^{18}
1-2	1.5	1.26×10^{17}	3.30×10^{-1}	1.25×10^{-1}	1.68×10^{17}
2-3	2.5	5.16×10^{16}	1.35×10^{-1}	3.06×10^{-2}	4.13×10^{16}
3-4	3.5	1.90×10^{16}	4.96×10^{-2}	8.05×10^{-3}	1.08×10^{16}
4-5	4.5	1.03×10^{16}	2.69×10^{-2}	3.40×10^{-3}	4.58×10^{15}
5-7.5	5.75	6.68×10^{15}	1.75×10^{-2}	1.73×10^{-3}	2.32×10^{15}
7.5-10	8.3	9.21×10^{14}	2.41×10^{-3}	8.24×10^{-5}	1.11×10^{14}
Total		3.83×10^{17}	1	1	1.35×10^{18}

Average photon energy per group was estimated by using reported gamma energy spectral equations for short lived fission products:

1. $N(E) = 26.8\ e^{(-2.3E)}$ for $0<E<1$ MeV
2. $N(E) = 8\ e^{(-1.1E)}$ for $1<E<8$ MeV
3. $N(E) = 6.7\ e^{(-1.05E)} + 30\ e^{(-3.8E)}$
4. $N(E) = 20\ e^{(-1.78E)}$ for $0.6<E<1.5$ MeV
5. $N(E) = 7.2\ e^{(-1.09E)}$ for $1.05<E<10.5$ MeV

E is energy in MeV and N(E) is the number of photons at energy E.

Using the above equations for the photon energy groups given by the empirical equations gives the following results as shown in Table 11-28.

Table 11-28 – Photon Energy Groups

MeV	<E> MeV
0 to 1	0.3
1 to 2	1.5
2 to 3	2.5
3 to 4	3.5
4 to 5	4.5
5 to 7.5	5.8
7.5 to 10.5	8.3

Historical radiation survey data at 1 MW is 2 mrem/h at 1 foot above the reactor pool surface (elevation [redacted]) with the reactor pool level at 20 feet (elevation [redacted]). At 2 MW, the historical dose rates at 1 MW are assumed to double giving 4 mrem/h at 1 foot above the reactor pool surface (elevation [redacted]) with the reactor pool level at 20 feet (elevation [redacted]).

Using the estimated pps fractions, the calculated average photon energies, and the historical survey data adjusted for 2 MW of 4 mrem/h at elevation [redacted] with a pool elevation of [redacted] (20 feet of water) and Microshield 5,[11-15] the pps emission rate at 2 MW was calculated. The Microshield 5 calculated values are approximately 4 percent different from the estimated values. See Table 11-29.

Table 11-29 – Comparison of Estimate to Calculated Reactor Source Term

MeV	Average MeV/photon	Estimated MeV/s	Estimated MeV/s Fraction	Estimated pps Fraction	Estimated pps at 2 MW	Calculated pps at 2 MW	mrem/h at 2 MW at elevation [redacted]	mrem/h per pps at elevation [redacted]
0-1	0.3	1.68E+17	4.39E-01	8.31E-01	1.12E+18	1.08E+18	1.11E-14	8.55E-33
1-2	1.5	1.26E+17	3.30E-01	1.25E-01	1.68E+17	1.62E+17	1.10E-06	8.50E-25
2-3	2.5	5.16E+16	1.35E-01	3.06E-02	4.13E+16	3.97E+16	7.39E-04	5.71E-22
3-4	3.5	1.90E+16	4.96E-02	8.05E-03	1.08E+16	1.04E+16	1.34E-02	1.03E-20
4-5	4.5	1.03E+16	2.69E-02	3.40E-03	4.58E+15	4.41E+15	8.01E-02	6.18E-20
5-7.5	5.75	6.68E+15	1.75E-02	1.73E-03	2.32E+15	2.24E+15	3.67E+00	2.83E-18
7.5-10	8.3	9.21E+14	2.41E-03	8.24E-05	1.11E+14	1.07E+14	2.38E-01	1.84E-19
Total		3.83E+17	1.00E+00	1.00E+00	1.35E+18	1.30E+18	4.00E+00	3.09E-18

Capture Gamma Photons

Capture gamma photons from in-pool experiments and components near the reactor core are minor in comparison to that from the reactor fuel and prompt fission photons. Estimates are in the range of 1×10^{12} photons per second (pps) as compared to an emission rate over 1×10^{18} pps from the reactor core region.

^{16}Nitrogen Photons

Contribution from ^{16}N is estimated to be approximately 5×10^{10} pps in forced convection cooling at 2 MW. This emission rate is minor in comparison to the estimated 1.3×10^{18} pps from fission products and prompt fission gamma photons.

$$5 \times 10^{10} \boldsymbol{photons} \sim (\mathbf{43}\, \boldsymbol{\mu Ci/ml})(\mathbf{10.7}\ \boldsymbol{gal})(\mathbf{3785}\, \boldsymbol{ml/gal})(\mathbf{0.73}\ \boldsymbol{photons/decay})(\mathbf{3.7} \times \mathbf{10^4}\ \boldsymbol{dps/\mu Ci})$$

Equation 11-74

Reactor shielding

Using the estimated reactor core source term and Microshield, dose rates were calculated for the following locations:

1. 0.2 mrem/h for an operating reactor at 2 MW on elevations [redacted].
2. 1 mrem/h for an operating reactor at 2 MW at elevation [redacted] on the reactor shield ledge.
3. 2×10^{-3} mrem/h for an operating reactor at 2 MW at elevation [redacted] behind the reactor shield wall.
4. 2 mrem/h for an operating reactor at 2 MW at elevation [redacted] at the edge of the reactor pool cavity on top of the reactor shield wall.
5. 4 mrem/h for an operating reactor at 2 MW at elevation [redacted] over the center of the

reactor on the reactor bridge.

These estimated dose rates are consistent with doubling those measured at 1 MW.

Dose rates at beamtube penetrations vary with the experiment installed and shielding present. Fully shielded and flooded beamtubes have dose rates up to 5 mrem/h at contact at 1 MW. Additional external shielding is installed as needed for experiments and beamtube penetrations in the reactor shield to keep dose rates ALARA as directed in the Radiation Protection Program.

Reactor Core Inventory

Radionuclide inventory of the reactor core and utilized fuel varies with operating and decay time, reactor core loading location, and other factors that affect burnup. Guidance on significant radionuclides for release are given in NUREG 1887. Radionuclide inventory activity is normalized to reactor power, Ci/MW or Bq/MW.

NUREG 1887 provides a reactor core inventory for commercial reactor fuel that may be adjusted for 2 MW operation of the PULSTAR reactor core. PULSTAR reactor fuel dimensions, other than length of fuel pins, materials (UO_2 fuel pellets and Zircaloy cladding), and ^{235}U enrichment are similar to commercial power reactor fuel. Core inventory from NUREG 1887 is used for low enriched fuel with a burnup of 30,000 MWd per MTU. PULSTAR fuel burnup is limited to 20,000 MWd/MTU.

Radionuclide inventories are used in Section 13 for accident scenarios involving release of activity from reactor fuel.

Neutron Radiation

The neutron fluence rate emitted during operation of the PULSTAR reactor has been measured to be approximately 1×10^{13} n·cm^{-2}s^{-1} at the reactor core periphery (~7.5 cm from the core face) for thermal and non-thermal energies at 1 MW. This value is doubled for 2 MW to 2×10^{13} n·cm^{-2}s^{-1}. These neutrons are associated with those emitted from fission. Upon leaving the reactor core, fission neutrons are thermalized by interactions occurring in the primary coolant water.

Using the reactor pool and reactor shield dimensions at the reactor core elevation of [redacted] and the reported neutron tenth value layers for water and heavy concrete of 22 g/cm^2 and 110 g/cm^2 for fission neutrons in water gives a significantly reduced neutron fluence rate escaping the reactor shield:[11-25]

$$\sim 0.2\ cm^{-2}s^{-1} = 2\times10^{13}\ cm^{-2}s^{-1}\left[{}^{7.5\ cm}\!\big/_{(99\ cm+198\ cm)}\right]^2 (0.1)^{\left(\frac{99}{22}+\frac{198*3.5}{110}\right)} \quad \textbf{\textit{Equation 11-75}}$$

Where 99 cm is the water shielding at a density of 1 g/cm^3 and 198 cm is the heavy concrete shielding at a density of 3.5 g/cm^3.

At a reactor power of 2 MW and using the dose conversion factor for thermal neutrons given in 10 CFR Part 20 of 9.8×10^8 cm^{-2}·rem^{-1} gives a dose equivalent rate of less than 1×10^{-3} mrem/h.

Many beam tubes are designed to provide a collimated beam of neutrons to an experiment located outside the reactor shield. Neutron fluence rates up to 1×10^{12} n·cm^{-2}s^{-1} at 1 MW are possible in the beam tube. This value is doubled for 2 MW to 2×10^{12} n·cm^{-2}s^{-1}.

Neutron radiation is non-detectable at the reactor pool top and shield wall. Neutron radiation dose-equivalent to personnel is measurable from beam tubes. Neutron shielding and other controls, such as access restrictions or barricades and beam shutters, are used to limit personnel dose from neutron radiation. These controls are established and reviewed for each experiment as required by facility procedures, the RHP, and RSAC and RSC, as necessary. Radiation surveys are made to confirm adequate shielding is in place.

Photo-neutrons from beryllium (Be) reflectors near the reactor core are insignificant in comparison to fission neutrons. Photon-neutrons are at higher energies and the fluence rate is also significantly reduced at the reactor pool liner and shield by interactions in water.

Primary Coolant System Shielding

Primary coolant system shielding is described below and is shown in Figures 11-11, 11-15, 11-17, 11-18, 11-19, and 11-21 with materials details listed in Table 11-30. It is noted that the primary coolant pump, heat exchanger, and upper level of the PPV are approximately at the same elevation [redacted] with the ^{16}N delay tanks and demineralizer in the lower level of the PPV. Refer to Figure 11-11.

The primary coolant system piping exiting the reactor pool is encased in a concrete pad underneath the reactor. The piping then enters a valve pit, in which isolation valves for the reactor are located, and proceeds through a pipe chase to the PPV.

The piping in the valve pit is located at elevation [redacted] and is shielded by [redacted] thick concrete shield plugs made which are stepped to avoid radiation streaming. The reactor experiment floor is directly above the valve pit.

In the PPV, the primary piping exiting from the reactor is shielded by concrete that is [redacted] thick (above) and [redacted] high. The primary pipe then enters the ^{16}N delay tank area which is shielded by a wall of concrete that is [redacted] high going from the floor to ceiling.

The PPV vault concrete walls and ceiling are [redacted] thick. The ceiling is covered with [redacted] of iron plating and earth. Above the delay tanks there is [redacted] of earth. Above the primary piping there is [redacted] of earth.

In the upper PPV hallway at elevation [redacted], the primary piping is located at elevation [redacted] and the roof is at elevation [redacted]. The roof is [redacted] and [redacted] of concrete.

[redacted]

Figure 11-19 – Primary Piping Vault – Plan View

[redacted]

Figure 11-20 – Primary Piping Vault – Shielding – Plan View

[redacted]

Figure 11-21 – Primary Piping Vault – Shielding – Section Views

Table 11-30 – Primary Coolant System Materials

[redacted]

Note: Earth/soil density taken from USDA Natural Resources Conservation Service (NRCS) data for North Carolina. Average density is 1.5 g/cm^3.

Activity in the primary coolant system is described in Section 11.1.1.2. Major sources of radiation requiring shielding are ^{16}N and the purification system demineralizer.

The purification system demineralizer is a major source of radiation during reactor operation due to retention of radionuclides with ionic charges. For typical, daily operating cycles the demineralizer radiation levels follow reactor power changes due to the buildup and decay of short-lived radionuclides. For prolonged operating times, e.g. several continuous days, the radiation level reaches an equilibrium value of 250 mrem/h at 1 MW. This is estimated to double at 2MW to 500 mrem/h due to the production rate of short-lived radionuclides being doubled. The demineralizer is located in the lower PPV to take advantage of shielding and access control. Longer lived radionuclides are a concern for solid waste disposal.

At 1 MW ^{16}N levels are decayed significantly in the primary coolant system after the delay tanks and dose rates are dominated by other short-lived radionuclides such as ^{41}Ar, ^{24}Na, ^{38}Cl, ^{28}Al, and ^{56}Mn. Sections of the primary coolant system after the delay tanks are located in the upper PPV and MER. For the upper PPV and MER, short lived products other than ^{16}N are used as a baseline for area radiation levels and are assumed to double at 2 MW due to the half-life and transit times involved.

At 1 MW, the radiation levels have been measured at 0.05 mrem/h from the primary piping in the upper PPV and 1 mrem/h at the primary pump and 3 mrem/h at the heat exchanger in the MER. At 2 MW, the estimated dose rates in the MER are estimated at 2 mrem/h at the primary pump and 6 mrem/h at the heat exchanger.

It is noted that the primary coolant system external radiation dose rates from ^{16}N that are estimated in this analysis will be revised by radiation survey measurements taken during start-up testing and power ascension to 2 MW.

^{16}N is a major external dose concern for sections of the primary coolant system from the reactor core to the delay tanks. At 1 MW ^{16}N levels in the lower PPV near the reactor outlet piping and delay tanks range up to 1500 mrem/h. At 2 MW, the radiation levels from ^{16}N are estimated from the decay time to reach various locations in the primary coolant system. The decay time, or transit time, is based on the volume of the primary coolant system pipe or component and volumetric flow rate. At 2 MW, the flow rate is twice that at 1 MW (1000 gpm vs 500 gpm).

At 1 MW:

As shown in Table 11-3, the concentration of ^{16}N is estimated from measured radiation levels at the valve pit using Microshield as 41.5 µCi/ml at the reactor core outlet at 1 MW in forced cooling:

$$41.5\ \mu Ci/ml = (13.9\ \mu Ci/ml)\left(e^{\lambda t}\right) \qquad \textit{Equation 11-76}$$

Where,

t is the transit time based on 22.5 feet of 10 inch diameter piping, 10 inch diameter piping at 500 gpm gives a linear flow rate of 2.0 feet per second

$$\lambda = \ln\frac{2}{7.13\ sec} = 0.0972\ s^{-1}$$

$13.9\ \mu Ci/ml$ in the primary piping gives the measured dose rate above the valve pit with no shielding at 1 MW

At 2 MW:

The neutron fluence doubles at 2 MW as compared to that for 1 MW. The irradiation time of water flowing through the reactor core at 2 MW is half that of 1 MW since the flow rate is 1000 gpm at 2 MW as compared to 500 gpm at 1MW. The net effect on ^{16}N activity produced in the reactor core is increased by approximately 3 percent:

$$42.8\ \mu Ci/ml = (41.5\ \mu Ci/ml)(\frac{2\ MW}{1\ MW})\frac{(1-e^{-\lambda t_2})}{(1-e^{-\lambda t_1})} \qquad \textit{Equation 11-77}$$

where,

t_1 is 1.284 seconds in the reactor core at 1 MW; $(10.7\ g/500\ gpm)(60\ \ s/min)$

t_2 is 0.642 seconds at 2 MW; $(10.7\ g/1000\ gpm)(60\ \ s/min)$

$13.9\ \mu Ci/ml$ in the primary piping gives the measured dose rate above the valve pit with no shielding at 1 MW

Reactor core water volume is 10.7 gallons. At 1000 gpm, the linear flow rate is 4 feet per second. The result is that the ^{16}N concentration at 2 MW is slightly higher and travels through the primary coolant system in half the time for that at 1 MW.

Decaying the ^{16}N source term for transit time at various locations in the primary coolant system indicates higher levels at 2 MW than at 1 MW. Although the ^{16}N activity decreases significantly after the delay tanks, ^{16}N radiation dose rates are still measurable downstream of the delay tanks. ^{16}N dose rates at various locations were calculated using Microshield and the average concentration within various primary coolant system components and piping based on primary coolant system dimensions and materials and external shielding present.

The average concentration was calculated as follows:

$$C_{ave} = C_{entry}\int e^{-\lambda\Delta t}dt = C_{entry}\left[\frac{1-e^{\lambda\Delta t}}{\lambda\Delta t}\right] \qquad \textit{Equation 11-78}$$

Where Δt is the transit time in a given part of the primary coolant system.

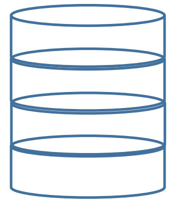
Microshield Model

1. Delay tanks were modeled as stainless steel tanks and segmented into 4 sections.
2. Primary coolant stainless steel pipe dimensions are 10 inch diameter and 0.365 inch wall.
3. Heat exchanger was modeled as a water filled tank with no external shielding.

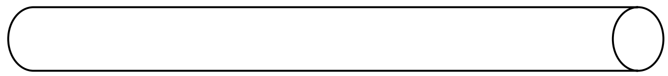
Dimensions used for the primary coolant system components and piping were taken from PULSTAR drawings.

Dimensions of the components were sectioned to avoid averaging of the source over one half-life; maximum time in any component section was 6.75 seconds. The delay tanks were modeled as 4 sections, RCS piping was modeled in 26 sections ranging in lengths from 1.5 feet to 22.5 feet, and the heat exchanger was modeled as a single tank.

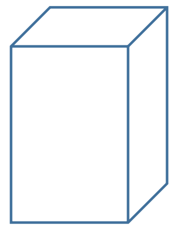
Delay Tanks A, B, C
450 gallons each
H = 7.5 feet
R = 1.6 feet
4 sections; each section height = 1.88 feet



Primary Pipe
10.02 in ID
10.75 in OD
0.365 in wall, SS 304



Heat Exchanger
2 ft x 2 ft x 5 ft
75 gallons



Dose rates at various locations of concern in the reactor (primary) coolant system (RCS) were estimated by using Microshield results at a uniform concentration of 1 µCi/ml of ^{16}N and then multiplying by the average concentration for each component or pipe or tank section. Distances to points of interest accounted for elevation changes and the shortest pathlength to the source. Shielding, self-shielding, and buildup were included in the dose rate calculations.

Dose rates from various sections of the primary coolant system were summed based on proximity to a given location to determine the total ^{16}N dose rate. ^{16}N dose rate calculations of were documented in Reference 11-36 and are summarized in Table 11-31.

Table 11-31 – Estimated Radiation Dose Rates from Primary Coolant System

Location	Elevation ft	Description	mrem/h from ^{16}N
Valve Pit	[redacted]	Above hot leg, unshielded	200
Valve Pit		Above hot leg, shielded	4
Lower PPV		unshielded area	13000
Lower PPV		shielded area	50
Upper PPV		PPV Gate/Ladder area	3
Upper PPV		PPV upper hall	0.1
MER		general area	5
Heat Exchanger		at 1 foot	1.5
Above PPV		Unrestricted area	0.15
Above PPV		Unrestricted area	0.1

Note: Grade level is elevation [redacted].

Summary of Primary Coolant System Radiation External Dose Analysis

To decrease the generation of radioactive materials associated with water impurities, a service water demineralizer is used to treat the city water before it is added to the pool.

With adequate demineralization, the principal activity in the primary water leaving the core is caused by ^{16}N. In the forced convection mode of cooling, the delay tanks in the primary system allows for decay of ^{16}N. Very low levels of activity normally exist downstream from the delay tanks in the primary coolant system.

The expected dose rate in public areas is expected to be 0.1 to 0.15 mrem/h at 2 MW above the PPV. The delay tanks are located in the PPV and are sufficiently shielded to meet 10 CFR Part 20 public dose limits assuming an occupancy time in unrestricted areas above the PPV is less than 800 hours at 2 MW. The area above the PPV is a grassy area that is rarely used by pedestrians with an observed occupancy by members of the public less than 800 hours (i.e. less than 33 days or 20 work weeks, 800 h ~ 100 mrem per y / 0.125 mrem per h).

Impurities found in the primary water are controlled by the purification system demineralizer where a portion of the primary flow is filtered and passed through a mixed bed ion exchange column. The demineralizer acting as a filter will accumulate radioactivity after continued operation. For this reason, a radiation monitor is placed at the demineralizer with a local readout at the entrance of the PPV, which provides an indication to personnel entering this room. Because of a potentially high radiation area, access to the PPV is controlled as required by 10 CFR Part 20 and the RP Program. Shielding in the PPV is in place and additional shielding may be used, as necessary. At 2 MW the estimated dose rate at the demineralizer is estimated to be 500 mrem/h. Unrestricted areas above the PPV, the demineralizer is estimated to give a dose rate of less than 0.1 mrem/h. ^{24}Na and ^{38}Cl and ^{56}Mn would be the radionuclides responsible for most of the dose rate from the demineralizer.

Corrosion control is accomplished by this demineralizer system, which maintains the resistivity of the water greater than 500 kΩ·cm. Pool water is periodically sampled to determine the resistivity. In addition, samples of primary coolant water are routinely processed to determine gross and isotopic radioactivity. This is used to determine the effectiveness of corrosion control and the existence of unusual radionuclides. Replacement of the primary demineralizer resin is scheduled when the pool resistivity meter indicates less than 1 MΩ·cm. At the time of replacement, demineralizer resins are estimated to have a dose rate of 20 mrem/h from longer lived radionuclides.

Shielding placed over the primary coolant piping in the valve pit was estimated to result in a dose rate of 4 mrem/h at 2 MW. Additional shielding may be added, e.g. iron, concrete, or lead, to reduce the dose rate in this area.

At 2 MW, the heat exchanger dose rate for all sources is estimated to be approximately 3 mrem/h at the primary pump and 8 mrem/h at the heat exchanger. Additional shielding and access restrictions are used as described in the RP Program to limit personnel dose.

The upper level of the PPV is estimated to have a dose rate of 0.1 to 3 mrem/h from ^{16}N at 2 MW. At 1 MW radiation levels in the upper PPV have been measured at 3 mrem/h and these are expected to double at 2 MW. At 2 MW the upper PPV is estimated to be range from 0.1 to 9 mrem/h depending on proximity to the demineralizer.

It is noted that the primary coolant system external radiation dose rates that are estimated in this analysis will be revised by radiation survey measurements taken during start-up testing and power ascension to 2 MW.

Waste Tank Vault

Liquid waste is held in three 904 gallon tanks located in the waste tank vault located outside the reactor building, but within the site boundary, off of Broughton Drive. The waste tank vault has a locked manhole and is at and below street elevation. Shielding of the waste tank vault beyond what is needed for structural integrity is not necessary due to the low activity present in liquid waste. The waste tank vault is made of concrete.

Occupational Exposure

Occupational exposure limits are established in the RP Program.[11-1] Occupationally exposed personnel include individuals that work with:

1. Non-exempt radioactive materials as defined in 10 CFR Part 30.
2. Work with equipment that generates or controls a radiation source.
3. Entry into areas that exceed 2 mrem per hour or 100 mrem per year.

All occupational workers are trained as required by the RP Program and NCSU Radiation Safety Manual and are monitored for external radiation dose.

Training includes topics required by 10 CFR Part 19, *Notices, Instructions, and Reports to Workers; Inspections and Investigations,*[11-26] the RP Program, radioactive material authorizations and the NCSU Radiation Safety Manual, radiation protection practices, radiation work permits, radiation protection practices, ALARA, and radiation dosimetry and radiation survey equipment. For unescorted access, personnel must successfully complete a security check or investigation as required by the facility security plan.

External radiation dosimetry meets 10 CFR Part 20 criteria for accreditation and monitoring the radiation type and energy that may be present. Accredited dosimeters at the reactor facility are capable of measuring deep dose-equivalent (DDE), lens dose-equivalent (LDE) , and shallow dose-equivalent (SDE) to the whole body and SDE to the extremities. Whole body dosimeters are sensitive to gamma, beta, and neutron radiation. Extremity dosimeters are sensitive to gamma and beta radiation. Extremity dosimeters are issued as needed or requested by the individual, RHP, or RSO.

Dosimeters used are accredited by National Voluntary Laboratory Accreditation Program (NVLAP). NVLAP accredited dosimeters are exchanged at least quarterly. Provisions are in place in the RP Program if a NVLAP accredited dosimeter is lost, damaged, or not used.

Submersion dose is an external radiation hazard and is monitored by the NVLAP accredited dosimeter for DDE, LDE, and SDE.

Declared pregnant women are instructed on the established policies of the reactor facility, 10 CFR Part 20, and the NCSU Radiation Safety Manual. Dose limits given in 10 CFR Part 20 are implemented upon voluntary declaration of pregnancy.[11-1,11-2,11-6]

A summary of DDE to occupational personnel from 2007 to 2014 is given in Table 11-32.[11-11]

Table 11-32 – DDE Occupational Personnel Exposures – 2007 to 2014

Year	Number of Personnel	Person-rem	Max rem	Average rem
2015	28	1.883	0.457	0.067
2014	26	2.200	0.556	0.085
2013	26	2.207	0.286	0.085
2012	29	3.420	1.113	0.118
2011	30	4.791	1.259	0.160
2010	33	2.689	0.644	0.081
2009	29	1.997	0.257	0.069
2008	23	2.432	0.869	0.106
2007	25	2.064	0.227	0.083
Average		2.631	0.630	0.095

Personnel dose varies due to work activities and reactor operating history. Average personnel doses indicate most personnel are receiving a net dose similar to external background dose of approximately 100 mrem per year. An annual dose of 0.5 rem from 2007 to 2015 was exceeded 7 times. No individuals have exceeded more than 30 percent of the annual limits at this facility since operations commenced in 1972.

Internal monitoring is not normally required due to various engineering and experiment controls that limit potential intakes to less than 10 percent of an annual limit on intake (ALI) given in 10 CFR Part 20. An active air sampling program is established to detect less than 10 percent of a derived air concentration (DAC) as given in 10 CFR Part 20. Bioassays are performed for respirator qualified personnel and as needed if internal contamination is suspected. If needed, the RHP and campus RSO perform internal dosimetry calculations using air sampling and bioassay data.

Results of dose assigned to the individual are provided annually as required by 10 CFR Part 19. Reports are made as required by the RP Program, 10 CFR Part 20, and TS.

Electronic dosimeters (ED) or self-reading dosimeters (SRD) may be used in addition to NVALP accredited dosimeters as authorized and issued by the RHP. ED are gamma and neutron sensitive. ED are calibrated annually using sources traceable to the National Institute of Standards and Technology (NIST) using an approved procedure or by an approved vendor. Gamma calibration is performed using ^{60}Co or ^{137}Cs. Neutron calibration is performed using ^{252}Cf or Pu(Be) or a characterized reactor beam. SRD are gamma sensitive and checked semi-annually with a NIST traceable source using an approved procedure. ^{60}Co or ^{137}Cs are normally used for SRD checks. ED and SRD respond to the radiation types and energies present. ED and SRD may be read by the user at any time.

Visitors and Students

Visitor access is limited as stated in the RP Program.[11-1] Visitors are not allowed to:

1. Do hands on work involving any radioactive material.
2. Work with equipment that generates or controls a radiation source.
3. Enter into areas that exceed 2 mrem per hour.

All visitors are escorted by authorized personnel. Visitors are issued a group NVLAP accredited dosimeter or ED or SRD. The visitor escort may wear the group dosimeter. DDE is limited to 100 mrem per year and 10 mrem per visit. Dosimeters used to monitor visitors have not exceeded DDE above 5 mrem in the history of the reactor facility. A visitor log and dose readings are maintained.

Students working in brief laboratory sessions are supervised by authorized personnel. Students are considered to be visitors that are allowed to perform hands-on work with exempt quantities of radioactive material as defined in 10 CF Part 30 or in areas that do not exceed 2 mrem per hour. Visitors and students are not allowed to work with contaminated items or in contaminated areas.

Visitor and student DDE is estimated in advance for laboratory sessions and other work. If this DDE estimate indicates that 100 mrem in a year may be exceeded, then the visitor or student is considered to be occupationally exposed and needs to complete occupational worker training.

11.1.6. Contamination Control

Contamination control is accomplished by limiting the amount of radioactive material produced in experiments and various work practices. Work practices include those given in radioactive material authorizations and radiation work permits, isolating clean and contaminated items, use of protective clothing, surveys, postings, and personal frisking.

The goal of the RP Program is to maintain accessible areas and commonly used items below detection limits. Limits, survey frequency, and the need for surveys for beta-gamma and alpha contamination are established in the RP Program. Detection limits of survey equipment meet the contamination limits established in the RP Program.[11-1] The contamination limits are based on guidance given in ANSI/HPS 13.2-1999 *Surface and Volume Radioactivity Standards for Clearance*.[11-29] Detection limits of less than 200 dpm per 100 cm^2 for beta-gamma emitting radionuclides are typically measured. Alpha emitting radionuclides are not routinely surveyed unless it is known that such radionuclides are present or if there is evidence of fuel failure. Detection limit less than 20 dpm per 100 cm^2 for alpha emitting radionuclides are typically measured. Contaminated items, equipment, and areas are surveyed and decontaminated if possible. Areas contaminated with short-lived radioactive materials may be permitted with adequate controls in place, e.g. protective clothing and limited access while decaying.

The personal contamination limit is established at a common standard of 5000 dpm per 100 cm^2 for beta-gamma emitting radionuclides based on survey instrument sensitivity and survey techniques. Every person exiting the reactor facility is required to monitor their hands, shoes, and other items that may be considered contaminated. Frisking is also performed upon exiting known contaminated areas or after handling potentially contaminated items. Contaminated individuals are trained to report such events to the reactor control room and / or RHP. Dose assessment is made and dose is assigned as described in the RP Program from personal contamination events. Historically, personal contamination events seldom occur and rarely require a dose assignment.

Regarding airborne activity, the goal of the RP Program is to maintain airborne activity below detectable limits for radionuclides that may result in an internal dose. Detection limits are below one percent of a derived air concentration given in 10 CFR Part 20 Appendix B. Airborne activity giving a submersion dose is controlled, monitored, and analyzed as an external dose in other parts of Section 11.

11.1.7. Environmental Monitoring

Environmental monitoring includes radiological surveys within the site boundary.[11-1,11-6] These include radiation and beta-gamma contamination surveys that are made at least quarterly. Also on a quarterly basis, NVLAP dosimeters located within the site boundary are used to assess external radiation and a ground water monitoring well is sampled and analyzed for reactor produced radionuclides.

Airborne effluent and discharges to the sanitary sewer are sampled and analyzed as required by 10 CFR Part 20 and the facility technical specifications. Radiation dose from airborne effluent and discharges to the sanitary sewer over the lifetime of the facility (1972 to date) have met regulatory requirements.

Additionally, environmental monitoring has been performed for locations outside the reactor site boundary since the facility was issued a license in 1972. Dosimeters are used to assess submersion dose from airborne activity. Samples and analyses of environmental media include air particulates, surface water, ground water, vegetation, and milk from areas near the reactor facility. The areas, monitoring frequency, and analyses are documented using facility procedures.

TS require that all environmental monitoring results be included in the annual report. Based on annual reports over the operating history of the facility since operations began in 1972 to 2016, it is concluded that radiation dose to members of the public is well within regulatory limits and typically is non-detectable.

Additional information is given in the environmental impact statement supporting the reactor license. The NCSU Radiation Safety Officer and RHP are responsible for implementing the environmental monitoring program.

11.2. Radioactive Waste Management

Reactor systems, operations and maintenance, and experiments are reviewed to determine potential sources of radioactive waste and necessary controls and monitoring. This review occurs as required by TS, RP Program, facility procedures, and the NCSU Radiation Safety Manual.[11-1]

11.2.1. Radioactive Waste Management Program

Materials used in experiments and operations are selected which are not likely to break, dissolve, or decompose or corrode.

Efforts are made to keep airborne activity well below regulatory limits, e.g. less than 1 percent of a derived air concentration, less than 10 percent of an annual limit on intake, and less than the concentration associated with the public constraint dose given in 10 CFR Part 20.

Coolant and corrosion activation products are kept minimal by maintaining and operating the reactor coolant purification system within TS limits. Typically, long-lived radionuclides in the reactor coolant meet the requirements for discharge to the sanitary sewer as stated in 10 CFR Part 20.

Experiments involving liquids and solids are contained. Liquid and solid radioactive waste is collected, monitored, and disposed as required by applicable regulations and the NCSU Radiation Safety Manual and NCSU Environmental Health and Safety Center.

11.2.2. Radioactive Waste Controls

Various controls are used to limit the production and to control the volume and activity of radioactive waste. Efforts are made to keep radioactive waste minimal in activity and volume. Examples of these efforts include the following:[11-1]

1. Production of short-lived radionuclides and holding for radioactive decay.
2. Minimizing production of activity to that needed for experiments.
3. Contamination controls and decontamination.
4. Segregation of contaminated from non-contaminated materials.
5. Segregation of short-lived radioactivity from long lived activity.
6. Limiting materials used in the reactor facility and in experiments and operations.
7. Re-use or recycling of materials.

Limitations and conditions are given in radioactive material authorizations on experiments and facility procedures regarding production of radioactive materials. Radiological surveys are used to identify potential sources of radioactive waste.

Airborne activity controls include air monitoring, reduction of air in experiments and experimental systems, limitation of materials, and encapsulation. Air monitoring at the reactor facility is performed routinely in the general areas and in the ventilation exhaust system and as needed for specific tasks (e.g. using a RWP for maintenance or experiments). Monitoring of the ventilation exhaust is required for reactor operation as described in TS. Filtration of exhaust by the confinement system is activated if a monitor setpoint is reached and for fueled experiments. If high airborne activity persists in the reactor building or ventilation system, several personnel have the authority to order reactor shut down (reactor operators, designated senior reactor operator, RHP, MEO, and Director NRP). The ventilation system monitoring and filters are maintained as required by TS.

^{41}Ar is the major radionuclide produced and released from the reactor facility. Argon is a component of air. By reducing the amount of air exposed to a neutron fluence, ^{41}Ar production decreases. Air may be displaced by use of nitrogen gas, shielding (lead, concrete, water), or an air vacuum.

Materials used in operations and experiments are selected to minimize airborne activity production, e.g. by off-gassing, evaporating, vaporizing, or over pressurizing. Encapsulation is used to contain airborne activity. TS for experiments have a pressure rating on encapsulation to avoid inadvertent releases.

Other controls may be used to limit the release of airborne activity. These include use of glovebags or gloveboxes with filtration, keeping items wet or submerged in water, or allowing activity to decay.

Both liquid and solid activity and volume are kept minimal by limiting the amount of material used, encapsulation of experiments, and radioactive decay of short lived radioactivity. Solid activity and volume is also kept minimal by selecting materials that are not likely to break, dissolve, or decompose or corrode. Additionally, many solid objects used have long useful lifetimes.

11.2.3. Release of Radioactive Waste

Airborne Radioactive Effluent

Airborne radioactivity is monitored as previously described in this section and released to the environment by an exhaust stack. Filtration of the exhaust by the confinement ventilation system HEPA and charcoal filters occurs automatically upon exceeding a monitor setpoint or manually by operator action. RMS data is frequently reviewed and summarized monthly using approved procedures to determine airborne effluent concentrations. ^{41}Ar is the primary radionuclide released. Data for ^{41}Ar release from 1996 to 2015 is given in Figure 11-10. Particulate activity is monitored in the exhaust air and reactor building. Submersion dose is measured by the area radiation monitors and personnel dosimeters. Radiation dose to occupational personnel and the public are well within limits given in 10 CFR Part 20. Dose from airborne activity is typically below 10 percent of the applicable limits.

Air sampling for other radionuclides may be performed using in-line equipment from the reactor exhaust stack and portable equipment used inside the reactor building, as necessary. Any abnormal air sample analysis requires an investigation as described in the RP Program.

Liquid Radioactive Waste

Radioactive liquids from the reactor building and laboratories located in the Burlington Engineering Laboratory (BEL) are collected by the liquid waste system. The liquid waste system consists of a central sump located in the MER, holding tanks and filters and three waste tanks located in a vault outside the reactor building but within the site boundary. See Figure 11-22 through Figure 11-25. Liquid waste is treated by filtration routinely and optionally with a demineralizer and/or evaporator.

Waste water from the liquid waste sump and air conditioning condensate is processed for disposal to the sanitary sewer. Holding tanks receive liquid waste water from the liquid waste sump and condensate from the air conditioning system. The water in the holding tanks is treated to meet particle size requirements using particulate filters, and optionally with a demineralizer and/or evaporator, prior to being pumped to the waste tanks. Representative samples are taken after treatment is complete during pumping to the external waste tanks. There are three external waste tanks each with a capacity of 904 gallons. Only one waste tank is normally filled at any time. The waste tanks are used to hold liquid waste until all analyses for discharge are complete.

Liquid levels in the waste tanks are tracked on filling and discharge. Levels are recorded using facility procedures. Optionally, water level sensors are located in each waste tank. Waste water level indication and waste tank fill and discharge valve controls are provided in Burlington Lab Room B103. The waste tanks are interconnected by piping to avoid an overflow, i.e. if one tank is filled the liquid waste will be routed to another tank by this piping. Valves on this piping are left open to prevent overflow spills. Isolation of liquids discharged from the reactor building, when necessary, can be accomplished by valves located between the sump and the three waste tanks or by disconnecting power to the sump. The sump in the MER is turned off and operated only when needed. Also, isolation valves connecting the waste tank drains to the sanitary sewer may be opened or closed to control release of liquid waste. The waste tank drain valves are operated using air pressure and are normally closed. The sanitary sewer drain valve is operated manually.

[redacted]

Figure 11-22 – Liquid Waste System Overview

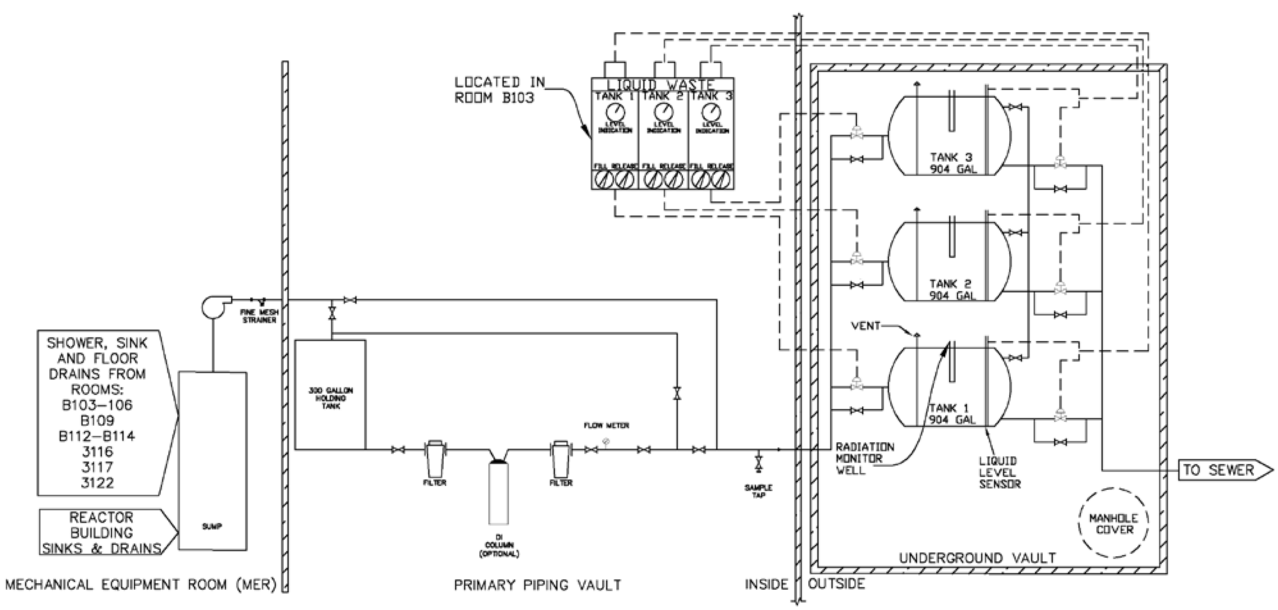


Figure 11-23 – Liquid Waste System

Liquid waste treatment, sampling, analysis, and discharge to the sanitary sewer or other disposal method are controlled by the RHP using approved procedures. If discharge requirements to the sanitary sewer are met as given in 10 CFR Part 20 and applicable local ordinances and NCSU policies, then liquid is discharged by opening the waste tank drain valves. The controls for releasing the contents of the liquid waste tanks are located in the BEL basement laboratories, locked, and under the administrative control of the RHP to prevent inadvertent release.

If discharge requirements to the sanitary sewer given in 10 CFR Part 20 and other applicable local ordinances and NCSU policies are not met, then the waste tanks are pumped to containers or drums and either filtered a second time by the liquid waste system or transferred to NCSU Environmental Health and Safety Center for disposal in accordance with applicable regulations.

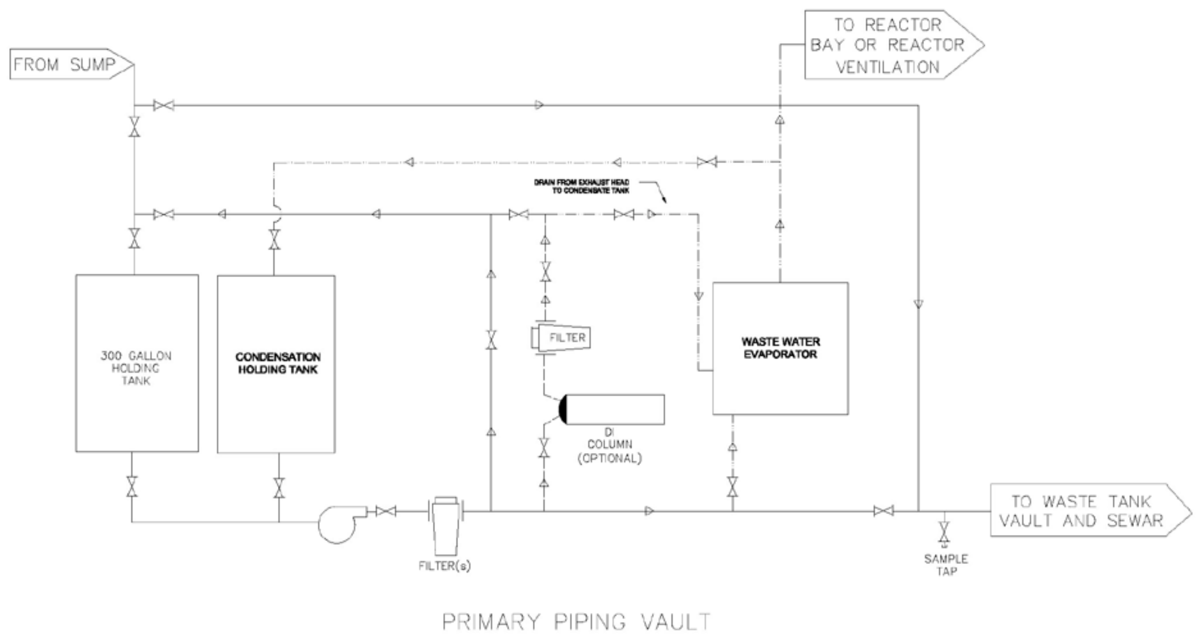


Figure 11-24 – Sump Waste Water Processing Diagram

NOTE: The waste water demineralizer and waste water evaporator are optional.

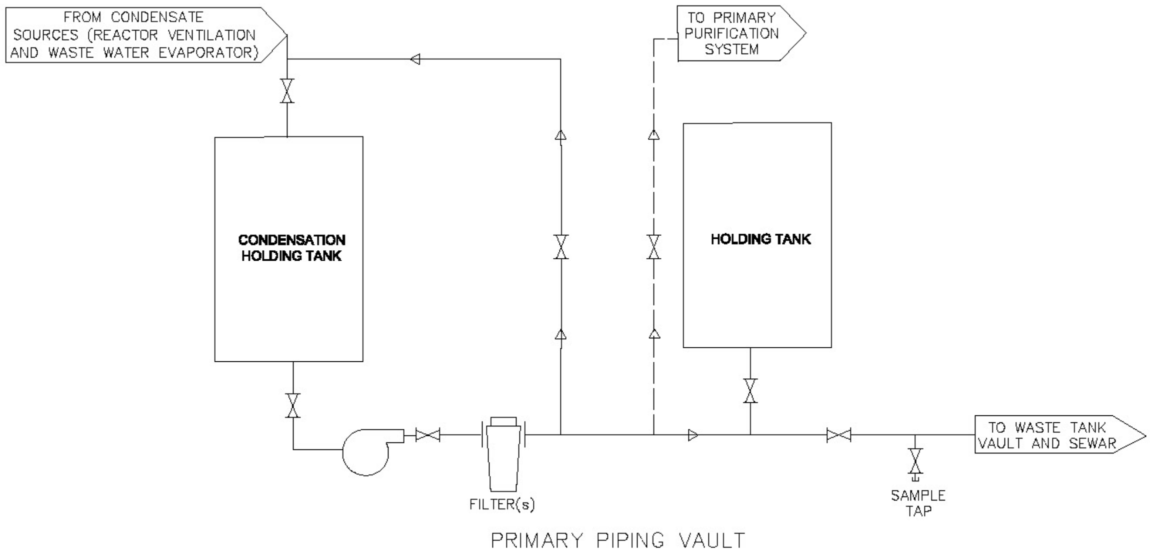


Figure 11-25 – Condensate Processing Diagram

NOTE: Recycling to the primary purification system is optional.

Solid Radioactive Waste

All radioactive solids from the reactor building and associated laboratories are collected, monitored, and stored for transfer to the NCSU Environmental Health and Safety Center for disposal. Disposal is in accordance with facility procedures, NCSU Radiation Protection Manual, 10 CFR Part 20, 10 CFR Part 61 *Licensing Requirements for Land Disposal of Radioactive Waste,*[11-30] and 10A NCAC 15 Section 1600 State of North Carolina Regulations for Protection Against Radiation.[11-31]

Dedicated radioactive trash containers are used to collect low activity solid waste. Spent resins are placed in US Department of Transportation (DOT) approved 55 gallon steel drums. Occasionally, low activity activated or contaminated equipment is collected and disposed using DOT approved containers. All solid radioactive waste to date meets the definition of Class A low level waste given in 10 CFR Part 61. Preparation and waste transfer records includes the waste container used, measurements to identify radionuclide activity and concentration upon being placed in an approved waste container, and external radiation and contamination surveys of the waste container.

Release of Liquids and Solid Items

Radioactive items are released to authorized personnel holding a valid radioactive materials license. Arrangements are made through the RHP and NCSU RSO. Radioactive transfers and shipments are made as specified in applicable regulations and license conditions, facility procedures, and the NCSU Radiation Safety Manual.

Release of items from the reactor facility for unrestricted use is made as described in the RP Program and approved procedures. Radiation surveys are made using field instruments, laboratory counting instruments, and gamma isotopic detection systems as necessary. Survey instruments used have a sensitivity ranging down to background radiation levels. Readings exceeding the minimum detectable activity or concentration above background levels are considered radioactive and withheld from release to unrestricted areas. Guidance on minimum detectable radiation and radioactivity given in US Nuclear Regulatory Commission NUREG-1507 *Minimum Detectable Concentrations with Typical*

Radiation Survey Instruments for Various Contaminants and Field Conditions[11-32] is used in facility procedures.

Liquid and solid waste disposal records are maintained as specified in facility procedures that meet 10 CFR Part 20 requirements and for 5 years as required by the technical specifications.

Radioactive Material Shipments

Radioactive material shipments are received and made following facility procedures, the NCSU Radiation Protection Manual, and applicable regulations. Records of shipments are maintained for 5 years as required by the technical specifications.

The following documents are used, as needed, for radioactive shipments:

1. Quality Assurance Program for Packaging and Transportation of Radioactive Material,[11-27] required by 10 CFR Part 71 and as approved by the US NRC, approval number 71-0331.
2. Security Plan for Shipment of Hazardous Material at the NCSU PULSTAR Nuclear Reactor,[11-28] required by 49 CFR Part 172.

Radioactive packages are received and shipped from the designated locations at the reactor facility only by authorized and trained personnel with approval by the RHP and NCSU Radiation Safety Officer. Shipments occur during normal business hours or at other prearranged times. Radioactive packages are promptly received or shipped for security and dose considerations.

11.3. References

11-1 North Carolina State University PULSTAR Reactor, *Radiation Protection Program.*

11-2 10 CFR Part 20-*Standards for Protection Against Radiation.*

11-3 ANSI/ ANS 15.11-2009-*Radiation Protection at Research Reactor Facilities.*

11-4 North Carolina State University PULSTAR Reactor, *R120 Facility License*, <date TBD>.

11-5 North Carolina State University, *Broad Scope Radioactive Materials License: 092-0090-3.*

11-6 North Carolina State University, *Radiation Safety Manual.*

11-7 10 CFR Part 30-*Rules of General Applicability to Domestic Licensing of Byproduct Material.*

11-8 10 CFR Part 50-*Domestic Licensing of Production and Utilization Facilities.*

11-9 10 CFR Part 74-*Material Control and Accounting of Special Nuclear Material*.

11-10 A. Keith Furr, *Chemical Rubber Company-CRC Handbook of Laboratory Safety*, 5th Ed., 2000.

11-11 North Carolina State University PULSTAR Reactor, *Annual Operating Reports.*

11-12 NUREG 1400, *Air Sampling in the Workplace,* September 1993, NRC Accession Number: ML15076A019, https://www.nrc.gov/docs/ML1305/ML13051A671.pdf.

11-13 North Carolina State University PULSTAR Reactor, *Safety Analysis Report-Appendix C-3: Additional Information and Analysis in Support of North Carolina State University PULSTAR Core and its Technical Specifications*, 1989.

11-14 North Carolina State University PULSTAR Reactor, *R120 Facility License-Appendix A:*

Technical Specifications, <date TBD>.

11-15 Microshield Computer Code, Grove Engineering; 1995.

11-16 Health Physics Journal Volume 11 Issue 6.

11-17 NRC Reg. Guide 2.2, *Development of Technical Specifications for Experiments in Research Reactors.*

11-18 PULSTAR Surveillance procedures.

11-19 ASME N510-1989-*Testing of Nuclear Air Treatment Systems.*

11-20 ANSI/ANS 15.7-*Research Reactor Site Evaluation.*

11-21 NRC 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 50 Appendix I,* https://www.nrc.gov/docs/ML0037/ML003740384.pdf.

11-22 NRC Regulatory Guide 1.111, *Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light Water Cooled Reactors*, https://www.nrc.gov/docs/ML0037/ML003740354.pdf.

11-23 NUREG 1887, RASCAL (Radiological Assessment System for Consequence Analysis) 3.0.5-*Description of Models and Methods, 2007,* https://www.nrc.gov/docs/ML0724/ML072480633.pdf.

11-24 EPA-520/1-88-020 (Federal Guidance Report 11), *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, https://www.nrc.gov/docs/ML1015/ML101590171.pdf.

11-25 R Faw and JK Shultis, *Radiological Assessment; Sources and Doses*, ANS publications, 1992.

11-26 10 CFR Part 19-*Notices, Instructions, and Reports to Workers; Inspections and Investigations.*

11-27 North Carolina State University PULSTAR Reactor, *Quality Assurance Program for Packaging and Transportation of Radioactive Material* (docket no. 71-0331), https://www.nrc.gov/docs/ML1523/ML15233A073.pdf.

11-28 North Carolina State University PULSTAR Reactor, *Security Plan for Shipment of Hazardous Material at the NCSU PULSTAR Nuclear Reactor.*

11-29 ANSI/HPS 13.2-1999-*Surface and Volume Radioactivity Standards for Clearance*.

11-30 10 CFR Part 61-*Licensing Requirements for Land Disposal of Radioactive Waste*.

11-31 10A NCAC 15 Section 1600-*North Carolina Regulations for Protection Against Radiation*

11-32 NUREG-1507, *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions*, https://www.nrc.gov/docs/ML0036/ML003676046.pdf.

11-33 North Carolina State University PULSTAR Reactor, *Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor*, June 2019.

11-34 US Department of Energy Report DOE/EH--0070 DE88 014691 *External Dose-Rate Conversion Factors for Calculation of Dose*, July, 1988.

11-35 North Carolina State University PULSTAR Reactor, *Emergency Plan, March 29, 2017.*

11-36 International Commission on Radiation Protection Publication 30, *Limits for Intakes of Radionuclides by Workers*, Part 1, Section 8, 1978.

11-37 North Carolina State University PULSTAR Reactor, ^{16}N Radiation Dose Calculations and RDU Airport Atmospheric Stability Classification, August 2019.

11-38 NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, Reviewed April 2014.

12. CONDUCT OF OPERATIONS

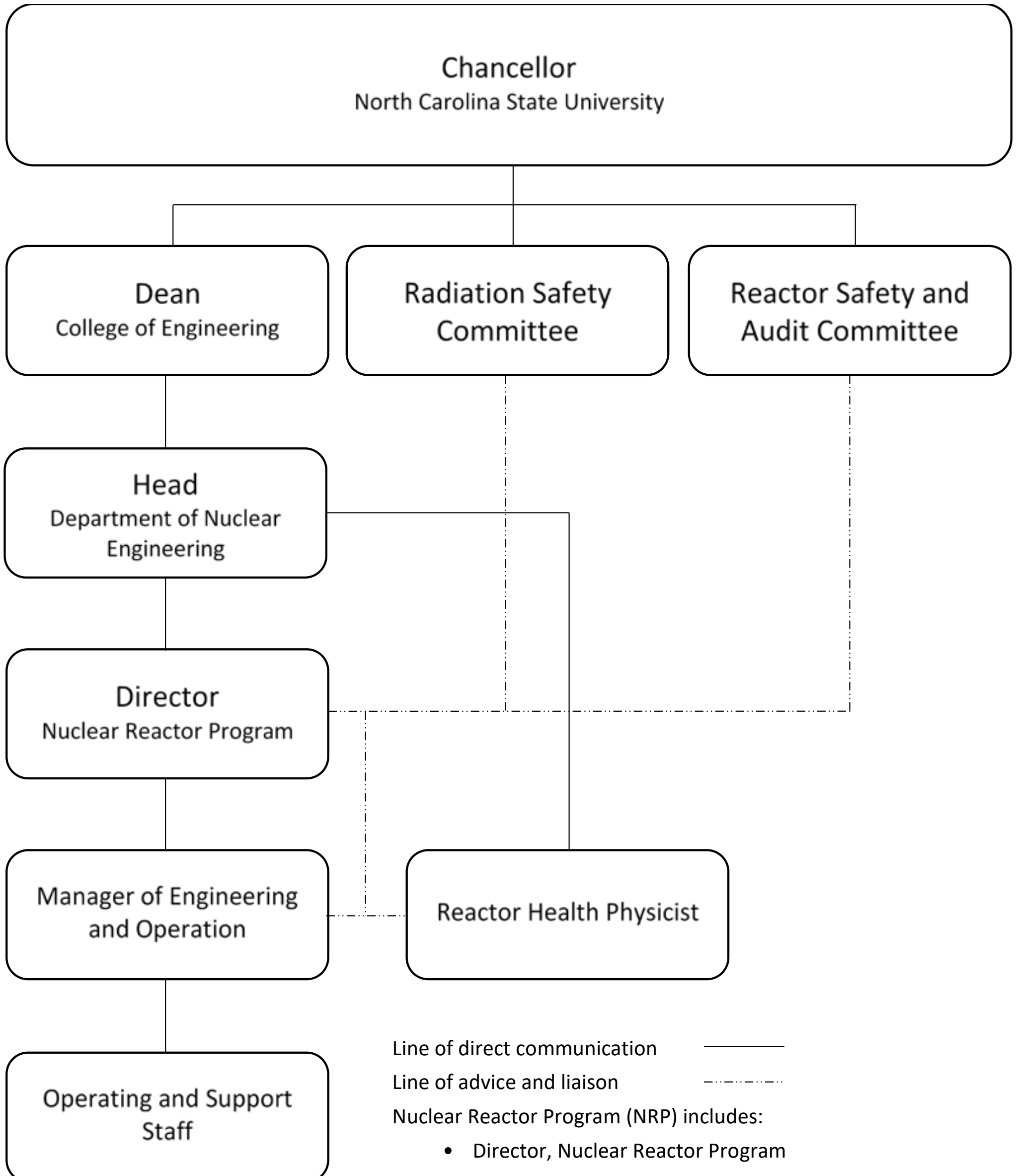
12.1. Organization

The administration of North Carolina State University (NCSU) and the Nuclear Reactor Program (NRP) recognize their responsibility to operate the PULSTAR Reactor safely and efficiently, and to provide experimental facilities for various teaching, research, and service programs in support of the university's mission. The administrative organizational structure of the PULSTAR Facility follows guidance given in ANSI/ANS Regulatory Guide 15.1, *Development of Technical Specifications for Research Reactors*.[12-1] Refer to Figure 12-1. The University is administered by the Chancellor. The College of Engineering is part of the University and is administered by the Dean who reports to the Chancellor. The Department of Nuclear Engineering is under the College of Engineering and is administered by the Department Head who reports to the Dean. The Nuclear Reactor Program is part of the Department of Nuclear Engineering and is administered by the Director who reports to the Department Head. The Director of the Nuclear Reactor Program (NRP) is primarily responsible for the safe and efficient operation of the PULSTAR Reactor Facility and for assuring that the facility resources are effectively utilized in meeting the mission of the University. The Manager of Engineering and Operations (MEO) reports to the Director of the Nuclear Reactor Program. The reactor facility operations staff report to the MEO. The Reactor Health Physicist (RHP) reports to the Nuclear Engineering Department Head and implements the radiation protection program at the reactor facility as a liaison and advisor to the NRP Director and MEO.

Operational activities at the reactor are reviewed and approved by review committees at NCSU. The Radiation Safety Committee (RSC) has the primary responsibility to ensure that the use of radioactive materials and radiation producing devices at NCSU, including the nuclear reactor, are in compliance with state and federal licenses and all applicable regulations. The RSC reviews and approves all license changes and experiments involving the use and potential release of radioactive material conducted at NCSU. The RSC also provides oversight of the NCSU Radiation Protection Program and is informed of the actions of the Reactor Safety and Audit Committee (RSAC). RSC may require additional actions by the RSAC and the NRP. RSAC has the primary responsibility to ensure that the reactor is operated and used in compliance with the facility license, technical specifications, and all applicable regulations. RSAC reviews and approves license changes, experiments, procedures, and design changes made to the reactor facility. RSAC also performs an annual audit of reactor operations and performance of the NRP. The annual audit report, including any recommendations, is provided to the RSC. RSAC may require additional actions by the NRP. In addition to RSC and RSAC approvals, occupational health and safety aspects of reactor operations are reviewed and approved by NCSU Environmental Safety.

12.1.1. Structure

The PULSTAR Reactor is an integral part of the Department of Nuclear Engineering within the College of Engineering at NCSU. The line and functional descriptions reflect the administrative controls for operation of the PULSTAR Reactor facility. The line organization is shown in Figure 12-1. The Director and the Reactor Health Physicist have formal reporting lines as well as documented secondary lines of communication if nuclear or radiation safety concerns cannot be resolved with the normal administrative reporting lines. Communication between the Reactor Health Physicist, Manager of Engineering and Operations, and the Operating and Support staff is continuous even though reporting lines may be separate.



Line of direct communication

Line of advice and liaison

Nuclear Reactor Program (NRP) includes:

- Director, Nuclear Reactor Program
- Manager, Engineering and Operations
- Operating and Support Staff

Reactor Health Physicist (RHP) reports to the Head, Department of Nuclear Engineering and serves both the NRP and Department of Nuclear Engineering.

Communication on reactor operations, experiments, radiation safety, and regulatory compliance occurs between the NRP, RHP, Reactor Safety and Audit Committee, Radiation Safety Committee, and campus Radiation Safety Division as described in the Technical Specifications and facility procedures.

Figure 12-1 – Nuclear Reactor Program Organizational Structure

12.1.2. Responsibility

12.1.2.1. Level 1 – Administration

This level includes the Chancellor, the College of Engineering Dean and the Nuclear Engineering Department Head.

12.1.2.2. Level 2 – Nuclear Reactor Program Director

The NRP Director is responsible for the safe and efficient operation of the PULSTAR reactor facility, general conduct of reactor performance and NRP operations, adherence to the conditions in the reactor license and technical specifications, long range development of the NRP, and NRP personnel matters. The NRP Director evaluates new service and research applications, develops new facilities and support for needed capital investments, and controls the NRP budget. The NRP Director works through the Manager of Engineering and Operations (MEO) to monitor daily operations and with the Reactor Health Physicist (RHP) to monitor radiation safety practices and regulatory compliance. The NRP Director is a faculty member and reports to the Nuclear Engineering Department Head.

12.1.2.3. Level 3 – Manager of Engineering and Operations

The Manager of Engineering and Operations (MEO) performs duties, as assigned by the NRP Director, associated with the safe and efficient operation of the facility. The MEO is responsible for adherence to the conditions of the reactor license and technical specifications, coordination of operations and experiments, maintenance at the facility, review and approval of experiments, and making changes to procedures. The MEO reports to the NRP Director.

12.1.2.4. Level 4 – Operating and Support Staff

This level includes licensed senior reactor operators (SRO), licensed reactor operators (RO), reactor operator assistants (ROA), and other personnel assigned to perform maintenance and technical support of the facility. SROs and ROs are responsible for assuring that operations are conducted in a safe manner and in compliance with conditions of the reactor license and technical specifications, applicable regulations and NCSU policies, and provisions as approved by the RSC and RSAC. Level 4 personnel report to the MEO.

12.1.2.5. Reactor Health Physicist

The Reactor Health Physicist (RHP) is responsible for implementing the radiation protection program and monitoring regulatory compliance at the reactor facility. The RHP reports directly to the Nuclear Engineering Department Head and is independent of the campus Radiation Safety Division. The RHP serves both the NRP and Nuclear Engineering Department.

12.1.2.6. Radiation Safety Committee

The RSC is a review and audit committee composed of faculty members and professional staff at NCSU that are appointed by the NCSU Administration. RSC reports directly to the NCSU Administration. RSC serves as the primary review and approval body at NCSU in all matters related to the use of radioactive material and radiation-producing devices and provides oversight of the NCSU Radiation Protection Program. The RSC is informed of the actions of the Reactor Safety and Audit Committee (RSAC) and may require additional actions by RSAC and the Nuclear Reactor Program (NRP).

The Radiation Safety Committee (RSC) has the primary responsibility to ensure that the use of radioactive materials and radiation producing devices at NCSU, including the nuclear reactor, are in compliance with state and federal licenses and all applicable regulations and NCSU policies. RSC responsibilities regarding reactor operations are consistent with the areas of expertise provided by its members. The membership of RSC includes personnel knowledgeable in the areas of reactor operations and radiation safety. RSC membership may not collectively represent a broad spectrum of expertise in reactor technology. RSC is informed of RSAC actions and the RSAC annual audit to provide oversight of reactor operations. Experiments involving the potential release of radioactive material and changes to the reactor facility license, except those containing safeguards information, are reviewed by the RSC. RSC may require additional actions by RSAC and the NRP. RSC membership and specific items that RSC reviews and approves are described in Section 12.2.

12.1.2.7. Reactor Safety and Audit Committee

The Reactor Safety and Audit Committee (RSAC) is a review and audit committee composed of faculty members and professional staff at NCSU that are appointed by the NCSU Administration. RSAC reports directly to the NCSU Administration and informs the RSC of its actions. RSAC provides specialized oversight of the PULSTAR reactor and audits reactor operations and performance of the NRP. RSAC may require additional actions by the NRP. RSAC membership and responsibilities are described in Section 12.2.

RSAC provides specialized and detailed oversight of the PULSTAR reactor at NCSU and has the primary responsibility to ensure that the reactor is operated and used in compliance with the facility license, technical specifications, and all applicable regulations. RSAC membership collectively represents a broad spectrum of expertise in reactor technology and radiation safety. RSAC performs an annual audit of the operations and performance of the NRP. RSAC reviews and approves all items affecting reactor operations, including safeguards information. RSAC informs the RSC of its actions and reports the results of the annual audit to RSC. RSAC membership and specific items that RSAC reviews and approves are described in Section 12.2.

12.1.3. Staffing

Staffing needs are based on regulatory requirements and the safe and efficient operation of the PULSTAR reactor. To meet this objective, a trained and qualified staff is maintained at the reactor facility. Training and qualifications for NRP staff are described below.

Minimum staffing when the reactor is not secured is listed below:

a. A licensed operator reactor operator (RO) or senior reactor operator (SRO) shall be present in the Control Room.

b. A reactor operator assistant (ROA), capable of being at the reactor facility within five (5) minutes upon request of the RO on duty.

c. A designed senior reactor operator (DSRO) is readily available on-call, meaning that the individual

   i. has been specifically designated and the designation known to the reactor operator on duty,

   ii. provides immediate contact information to the reactor operator on duty, e.g. telephone, pager, radio,

iii. is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 minutes or within a 15 mile radius).

d. A Reactor Health Physicist (RHP) or designated alternate is present or on call, under the same limitations as described for the DSRO. The designated alternate may be the Campus Radiation Safety Officer or designated alternate. Designated alternates shall meet the RHP qualifications listed in Section 12.1.4.

The following events require the presence of a SRO at the reactor facility:

1. Initial startup and approach to power.
2. All fuel or control rod relocations within the reactor core or pool.
3. Relocation of any in-core experiment with a reactivity worth greater than one dollar (730 pcm).
4. Recovery from unplanned or unscheduled shutdown or significant power reduction with documented approval for resumption of operations.

12.1.4. Selection and Training of Personnel

The research reactor personnel shall have a combination of academic training, job related experience, and skills commensurate with their level of responsibility that provides reasonable assurance that decisions and actions taken during all normal and abnormal conditions will be such that the reactor is operated in a safe manner.

The minimum qualifications for the NRP Director are a Master of Science in engineering or physical science and at least six years of nuclear experience related to fission reactor technology. The degree may fulfill up to four years of the required six years of nuclear experience on a one-for-one time basis. Within three months of appointment, the NRP Director receives briefings sufficient to provide an understanding of the general operational and emergency aspects of the facility.

The minimum qualifications for the MEO are a Bachelor of Science in engineering or physical science and at least six years of nuclear experience related to fission reactor technology. The degree may fulfill up to four years of the required six years of nuclear experience on a one-for-one time basis. The MEO receives appropriate facility specific training within three months of appointment and must be certified as a senior reactor operator within one year of appointment.

All senior reactor operators must have three years of nuclear experience and a high school diploma or have successfully completed a general education development (GED) test. A maximum of two years equivalent full-time academic training may be substituted for two years of the required three years of nuclear experience as applicable to research reactors for SRO. Other Level 4 personnel shall have a high school diploma or shall have successfully completed a GED test.

The RHP shall have a high school diploma or shall have successfully completed a GED test and have three years of relevant experience in applied radiation safety. A maximum of two years equivalent full-time academic training may be substituted for two years of the required three years of experience in radiation safety as applicable to research reactors.

The training program for the NCSU PULSTAR operations staff is designed to meet the needs of each person appointed, depending on background, previous experience, and training. The phases of training include basic reactor theory, reactor facilities and design, and operation of the PULSTAR reactor under the supervision of a licensed reactor operator. Training meets the regulatory requirements given in 10 CFR Part 19, 10 CFR Part 20, and 10 CFR Part 55 for occupationally exposed radiation workers and reactor operators.

All operators undergo a selection, training and certification program prior to unsupervised operation of the PULSTAR reactor.[12-2,12-3] All licensed operators participate in a requalification program[12-2] conducted over a two year period. The requalification program is then followed by successive two year programs. A requalification program is on file with the NRC and described in more detail in Section 12.10.

12.1.5. Radiation Safety

The Reactor Health Physicist (RHP) is responsible for implementing the Radiation Protection Program[12-4] at the reactor facility. The RHP reports directly to the Nuclear Engineering Department Head and is independent of NCSU Radiation Safety as shown in Figure 12-1.

The designated alternate to the RHP is the NCSU Radiation Safety Officer or another designee that meets the RHP qualifications and is familiar with the reactor facility and NCSU Radiation Protection Programs.[12-5]

The RHP or designated alternate have the authority to interdict or terminate activities conducted at the reactor facility if any safety concern arises. Both the RHP and NCSU Radiation Safety Officer may raise concerns directly to the NRP Director, NE Department Head, RSC, RSAC, or other levels of the NCSU Administration shown in Figure 12-1 regarding activities at the reactor facility as needed. NRP personnel are instructed to follow any such stop work orders made by the RHP. In addition, NRP personnel are informed on how to raise safety concerns with their supervisor (MEO or NRP Director), the RHP, and NCSU Environmental Safety personnel. NCSU Environmental Safety includes the NCSU Radiation Safety Officer.

Laboratories associated with the reactor are regulated by the State of North Carolina while activities within the reactor facility are regulated by the NRC. Experiments, equipment, radioactive sources and samples, and personnel frequently enter and exit the two regulated areas. A documented radiation protection program is required by both NRC and State of North Carolina regulations. North Carolina is an agreement state with radiation safety regulations similar to those of the NRC. Coordination of radiation safety requirements at the reactor and NCSU is therefore beneficial. Because the reactor is unique from the other laboratories at NCSU, the Radiation Protection Program at the PULSTAR reactor includes specific documents and procedures. The NCSU Radiation Protection Program may be included by reference in the PULSTAR Reactor Radiation Protection Program. Requirements given in the PULSTAR Reactor Radiation Safety Program meet or exceed those given in the NCSU Radiation Protection Program.

12.2. Review and Audit Activities

A conscientious safety program is part of NRP operations. This includes self-assessments and periodic supervisory reviews, e.g. the radiation protection program annual review required by 10 CFR Part 20 and an annual occupational safety plan self-assessment.

Independent safety committees have been established to oversee and audit the manner in which reactor operations are carried out. Reviews and audits of reactor operations are conducted by the Reactor Safety and Audit Committee (RSAC) and may be conducted by the Radiation Safety Committee (RSC). RSC and RSAC are appointed by the NCSU Administration and are composed of persons from such fields as reactor analysis, design, operations, instrumentation, and other engineering and scientific fields, and are active in and concerned with safety analysis. RSC and RSAC interact with the NRP and RHP regarding safety reviews and audits and both report to the NCSU Administration.

12.2.1. Composition and Qualifications

RSC shall consist of members from the general faculty who are actively engaged in teaching or research involving radioactive materials or radiation devices. RSC may also include non-faculty members who are knowledgeable in nuclear science or radiation safety. RSC membership includes the NCSU Radiation Safety Officer, RSAC Chair, RHP, and a member of the NRP.

RSAC consists of at least five individuals who have expertise in one or more of the component areas of nuclear reactor safety. These include nuclear engineering, nuclear physics, health physics, electrical engineering, chemical engineering, material engineering, mechanical engineering, radiochemistry, and nuclear regulatory affairs. At least three of the RSAC members are appointed from the faculty. The faculty members shall include the NRP Director, one member from an appropriate discipline within the College of Engineering, and one member from the faculty. The remaining required members of RSAC include the Reactor Health Physicist (RHP) and a member from the campus Radiation Safety Division of the Environmental Health and Safety Center. One additional member from an outside nuclear related establishment may be appointed to RSAC. At the discretion of RSAC, specialist(s) from other universities and outside establishments may be invited to assist in its appraisals.

The NRP Director, RHP, and a member from the campus Radiation Safety Division of the Environmental Health and Safety Center are permanent members of RSAC.

12.2.2. Charter and Rules

RSC and RSAC committee member appointments are made by the NCSU Administration for terms of three (3) years.

RSC meets as required by the broad scope radioactive materials license issued to NCSU by the State of North Carolina. RSAC meets at least four (4) times per year, with intervals between meetings not to exceed six months. RSC and RSAC may also meet upon call of the respective committee Chair.

A quorum of RSC or RSAC shall consist of a majority of the full committee membership and must include the committee Chair or a designated alternate for the committee Chair. For both RSC and RSAC, members from the line organization shown in Figure 12-1 may not constitute a majority of the quorum.

RSC minutes are distributed to the RSAC Chair and NRP Director. A summary of RSAC meeting minutes, reports, and audit recommendations approved by RSAC are distributed to the College of Engineering Dean, the Nuclear Engineering Department Head, the NRP Director, the RSC Chair, Director of Environmental Health and Safety, RSAC Chair, and the MEO prior to or at the time of the next scheduled RSAC meeting.

12.2.3. Review Function

Reviews of activities proposed or conducted at the reactor facility are conducted by both RSC and RSAC. Reviews performed by RSC and RSAC vary based on the expertise of committee membership. The membership of RSC includes personnel knowledgeable in the areas of radiation safety. RSAC membership collectively represents a broad spectrum of expertise in reactor technology, while membership of RSC may not. Specific items that are reviewed and approved by the committees are described below. RSC is informed of the actions of the RSAC and may require additional actions by RSAC and the NRP.

a. The following items are reviewed and approved by the RSC:

    i. All new experiments or classes of experiments that could result in the release of radioactivity.

    ii. Proposed changes to the facility license and/or technical specifications, excluding safeguards information.

b. The following items are reviewed and approved by the RSAC:

    i. Determinations that proposed changes in equipment, systems, tests, experiments, or procedures which have safety significance meet facility license and technical specification requirements.

    ii. All new procedures and major revisions having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

    iii. All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

    iv. Proposed changes to the facility license or technical specifications, including safeguards information.

    v. Proposed changes to the emergency procedures.

c. The following are reviewed by both the RSC and RSAC:

    i. Violations of the facility license or technical specifications.

    ii. Violations of internal procedures or instructions having safety significance.

    iii. Operating abnormalities having safety significance.

    iv. Reportable events as defined in the technical specifications.

    v. New or untried experiments as defined in technical specifications. All proposed experiments are reviewed by the Manager of Engineering and Operations and the Reactor Health Physicist (or their designated alternates). Either of these individuals may deem that the proposed experiment is not adequately covered by the documentation and/or analysis associated with an existing approved experiment and therefore constitutes an untried experiment that will require the approval process.

    vi. Substantive changes to previously approved experiments.

    vii. Proposed changes to the Emergency Plan.

12.2.4. Audit Function

The audit function consists of selective, but comprehensive, examination of operating records, logs, and other documents. Discussions with facility personnel and observation of operations are also used as appropriate. In no case shall an individual immediately responsible for the area perform an audit in that area. The audit includes the following:

a. Facility operations for conformance to the facility license and technical specifications, annually, but at intervals not to exceed fifteen (15) months.

b. The retraining and requalification program for the operating staff, biennially, but at intervals not to exceed thirty (30) months.

c. The results of actions taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, annually, but at intervals not to exceed (15) months.

d. The Emergency Plan and emergency procedures, biennially, but at intervals not to exceed thirty (30) months.

e. Radiation Protection annually, but at intervals not to exceed fifteen (15) months.

Deficiencies uncovered that affect reactor safety are immediately reported to the Nuclear Engineering Department Head, Director of the Nuclear Reactor Program, and the RSC. The annual audit report made by the RSAC, including any recommendations, is provided to the RSC.

Additionally, laboratory self-assessment for compliance with Occupational Safety and Health regulations and a radiation protection program review are performed annually. Informal audits may also be performed by the MEO and RHP, independent of the RSC or RSAC, to review operations.

12.3. Procedures

Preliminary procedures were furnished by the contractors during the startup of the facility. The staff then developed and augmented each procedure required for PULSTAR operations. A PULSTAR Operations Manual details reactor operating procedures while health physics procedures, special procedures, and surveillance procedures have also been developed to cover the operation of the facility.

Written procedures shall be prepared, reviewed and approved prior to initiating any of the following:

a. Startup, operation and shutdown of the reactor.

b. Fuel loading, unloading, and movement within the reactor.

c. Maintenance of major components of systems that could have an effect on reactor safety.

d. Surveillance checks, calibrations and inspections required by the facility license or technical specifications or those that may have an effect on the reactor safety.

e. Personnel radiation protection, consistent with applicable regulations and that include commitment and/or programs to maintain exposures and releases as low as reasonably achievable (ALARA).

f. Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.

g. Implementation of the Emergency Plan and Security Plan.

Administrative controls have been established to insure that all operations, tests, and emergencies are handled in accordance with written procedures. New procedures are written as necessary for reactor operations, experiments, and health physics. Procedures are periodically reviewed and revised as necessary to address normal, abnormal, and emergency operating conditions. Review and approval as described in Section 12.2.3 is required prior to a new procedure or procedure change being implemented.

Minor changes to the original procedures which do not change the original intent may be made by the Manager of Engineering and Operations, but the modifications shall be approved by the Director of the Nuclear Reactor Program within fourteen (14) days.

Temporary procedure deviations may be made by the designated senior reactor operator or by the Manager of Engineering and Operations in order to deal with special or unusual circumstances or conditions. Such deviations are documented and reported to the Director of the Nuclear Reactor Program. RSAC is also informed of any minor changes or temporary procedure deviations.

The NCSU Health and Safety Manual is followed for laboratory work involving non-radioactive and/or physical hazards for compliance with occupational safety requirements. These are implemented using a laboratory safety plan that is reviewed and approved by NCSU Environmental Safety. If laboratory work, experimental activities, or operational activities at the reactor involve hazardous materials or processes that may affect reactor operations, written procedures for that activity are reviewed and approved by NCSU Environmental Safety, RSC and RSAC.

12.4. Required Actions

12.4.1. Actions to be Taken in Case of a Safety Limit Violation

a. The reactor shall be shutdown and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission.

b. The Safety Limit violation shall be promptly reported to the Director of the Nuclear Reactor Program or his designated alternate.

c. The Safety Limit violation is reported to the Nuclear Regulatory Commission in accordance with the technical specifications.

d. A Safety Limit violation report shall be prepared that describes the following:

   i. Circumstances leading to the violation including, when known, the cause and contributing factors.

   ii. Effect of violation upon reactor facility components, systems, or structures and on the health and safety of facility personnel and the public.

   iii. Corrective action(s) to be taken to prevent recurrence.

The report is reviewed by the RSC and RSAC and any follow-up report is submitted to the Nuclear Regulatory Commission when authorization is sought to resume operation.

12.4.2. Actions to be Taken for Reportable Events (other than Safety Limit violations)

In case of a reportable Event as defined in Section 12.4.3, the following actions shall be taken:

a. Reactor conditions are returned to normal or the reactor shall be shutdown. If it is necessary to shutdown the reactor to correct the occurrence, operation shall not be resumed unless authorized by the Director of the Nuclear Reactor Program, or his designated alternate.

b. The occurrence is reported to the Director of the Nuclear Reactor Program, and to the Nuclear Regulatory Commission in accordance with the technical specifications.

c. The occurrence is reviewed by the RSC and RSAC at their next scheduled meeting.

12.4.3. Reportable Events

A reportable event is any of the following

a. Violation of a Safety Limit.

   Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity (10 CFR Part 50.36).

b. Release of radioactivity from the site above allowed limits. Allowed limits for radioactive releases are given in 10 CFR Part 20.

c. Operation with actual Safety System Settings (SSS) for required systems less conservative than the Limiting Safety System Settings (LSSS) specified in the technical specifications.

   Limiting safety system settings (LSSS) for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting is chosen such that an automatic protective action will correct the abnormal situation before a safety limit is exceeded (10 CFR Part 50.36).

d. Operation in violation of Limiting Conditions for Operation (LCO) established in the technical specifications.

   Limiting conditions for operation (LCO) are the lowest functional capability or performance levels of equipment required for safe operation of the facility (10 CFR Part 50.36).

e. A reactor safety system component malfunction which renders or could render the reactor safety system incapable of performing its intended safety function unless the malfunction or condition is discovered during maintenance tests or periods of reactor shutdown.

   Where components or systems are provided in addition to those required by the technical specifications, the failure of the extra component or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.

f. An unanticipated or uncontrolled change in reactivity greater than one dollar (730 pcm). Reactor trips resulting from a known cause are excluded.

g. Abnormal or significant degradation in reactor fuel, or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks), which could result in exceeding radiological limits for personnel or environment, or both, as prescribed in the facility Emergency Plan.

h. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence of an unsafe condition with regard to reactor operations.

12.4.4. Actions to be Taken for Emergency Action Levels (EAL)

Actions are required if an Emergency Action Level (EAL) is reached at the reactor facility. EALs are defined in the Emergency Plan and described in Section 12.7.[12-6] Notifications are made to local, state, and federal agencies as specified in the Emergency Plan and emergency procedures if an EAL is exceeded or imminent.

12.5. Reports

Reports made to the US Nuclear Regulatory Commission are listed below. These reports are made to the NRC as required by regulations. Details on reports made are given below:

12.5.1. Reportable Event

For reportable events, a report is made not later than the following work day by telephone to the Nuclear Regulatory Commission Operations Center followed by a written report within fourteen (14) days that describes the circumstances of the event to the NRC Document Control Desk.

12.5.2. Permanent Changes in Facility Organization

Permanent changes in the facility organization involving either Level 1 or 2 personnel require a written report within thirty (30) days to the NRC Document Control Desk.

12.5.3. Changes Associated with the Safety Analysis Report

Significant changes in the transient or accident analysis as described in the Safety Analysis Report require a written report within thirty (30) days to the NRC Document Control Desk.

12.5.4. Annual Operating Report

An annual operating report for the previous calendar year is required to be submitted no later than March 31st of the present year to the NRC Document Control Desk. The annual report shall contain at a minimum, the following information:

a. A brief narrative summary:
   i. Operating experience including a summary of experiments performed.
   ii. Changes in performance characteristics related to reactor safety that occurred during the reporting period.
   iii. Results of surveillance, tests, and inspections.
b. Tabulation of the energy output (in megawatt days) of the reactor, hours reactor was critical, and the cumulative total energy output since initial criticality.
c. The number of emergency shutdowns and unscheduled scrams, including reasons and corrective actions.
d. Discussion of the corrective and preventative maintenance performed during the period, including the effect, if any, on the safety of operation of the reactor.
e. A brief description, including a summary of the analyses and conclusions of changes in the facility or in procedures and of tests and experiments carried out pursuant to 10 CFR Part 50.59.
f. A summary of the nature and amount of radioactive effluent released or discharged to the environs beyond the effective control of the reactor facility as measured at or prior to the point of release or discharge, including:
   i. Liquid Waste, summarized by calendar month, quarter, and year, indicating number of batch releases, total radioactivity released, total tritium activity released, total liquid volume released, diluent volume required.
   ii. Identification of fission and activation products:

Whenever the undiluted concentration of radioactivity in the liquid waste tank at the time of release exceeds 2 $\times 10^{-5}$ µCi/ml, as determined by gross beta/gamma count of the dried residue of a one liter sample, a subsequent analysis is also to be performed prior to release for principal gamma emitting radionuclides. An estimate of the quantities present is reported for each of the identified nuclides.

The 2 $\times 10^{-5}$ µCi/ml limit is based on a gross beta value of 1 $\times 10^{-7}$ µCi/ml, the limit for unlisted nuclides with a half-life greater than 2 hours given in 10 CFR Part 20 Appendix B Table 3, and a minimum daily campus dilution volume of 671,000 gallons. The assumed release is three full waste water tanks per day; [2 $\times 10^{-5}$ µCi/ml ~(1$\times 10^{-7}$ µCi/ml) (671,000 gallons per day / 2712 gallons per day]. Data from fiscal years 2013 to 2016 indicates daily water use at NCSU of 1$\times 10^{6}$ gallons per day.[12-7]

Continued use of 671,000 gallons per day is sufficient to meet the requirements of 10 CFR Part 20 for discharges to the sanitary sewer.

iii. Disposition of liquid effluent not releasable to the sanitary sewer system:

Any waste tank containing liquid effluent failing to meet the requirements of 10 CFR Part 20 Appendix B, is reported to the NRC and includes the method of disposal, total radioactivity in the tank prior to disposal, total volume of liquid in the tank, concentrations of identified principle gamma-emitting radionuclides from the dried residue of a one liter sample, and the tritium concentration.

iv. Airborne radioactivity concentrations and total activity discharged for gases and particulates, with half-lives greater than eight days. The average radioactive concentration and total radioactivity discharged by calendar month and year are estimated for each identified nuclide based on representative isotopic analysis.

v. Solid waste disposal including volume, total activity, and dates of shipment and disposition (if shipped off-site).

g. A summary of radiation exposures received by facility personnel and visitors, including pertinent details of significant exposures.

h. A summary of the radiation and contamination surveys performed within the facility and significant results.

i. A description of environmental surveys performed outside the facility.

12.5.5. Other Reports

Reports are also made as required by 10 CFR Part 20, 30, 50, 55, 70, and 71, as applicable, including:

a. Surface contamination or external radiation levels from radioactive packages received in excess of regulatory limits as specified in 10 CFR Part 20.1906.

b. Theft or loss of licensed materials as specified in 10 CFR Part 20.2201.

c. Notification of incidents as specified in 10 CFR Part 20.2202.

d. Radiation exposures, radiation levels, and concentrations of radioactive material exceeding the constraints or limits as specified in 10 CFR Part 20.2203.

e. Planned special exposures (PSE) as specified in 10 CFR Part 20.2204.

f. Individuals exceeding dose limits as specified in 10 CFR Part 20.2205.

g. Lost or misdirected radioactive shipments as specified in 10 CFR Part 20 Appendix G.

h. Domestic licensing of by-product material as specified in 10 CFR Part 30.50.

i. Activation of the Emergency Plan is reported as specified in 10 CFR Part 50.54 and 10 CFR Part 50 Appendix E. Details about the Emergency Plan are given in Section 12.7 below.

j. Change in reactor operator status as specified in 10 CFR Part 50.74.

k. Commission of a felony by a reactor operator as specified in 10 CFR Part 55.53.

l. Domestic licensing of special nuclear material as specified in 10 CFR Part 70.50, 70.52, or 10 CFR Part 70 Appendix A.

m. Packaging and transportation of radioactive materials as specified in 10 CFR Part 71.95.

Security related events are reported as specified in the Security Plan, Emergency Plan, and reactor license conditions. Refer to Section 12.8.

12.6. Records

Operating records required administratively and by the technical specifications are needed to ensure adequate surveillance of reactor operations and to provide sufficient repair history and recorded symptoms to detect malfunctions and perform remedial maintenance. The use of standard log book entries, periodic recording of plant parameters, and following detailed checklists are most important.

Activities such as shutdowns, malfunctions, maintenance performed, research activities, and sample irradiations are recorded. Data from these logs are used to summarize and render long term evaluations of the facility operation. Included in the records on file will be routine operating logs, preventive maintenance and malfunction reports, and equipment history.

12.6.1. Records to be retained for a period of at least five (5) years

a. Normal plant operation and maintenance.

b. Principal maintenance activities.

c. Reportable events.

d. Equipment and component surveillance activities as detailed in the technical specifications.

e. Experiments performed with the reactor.

f. Changes to operating procedures.

g. Facility radiation and contamination surveys other than those used in support of personnel radiation monitoring.

h. Audit summaries.

i. RSC and RSAC meeting minutes.

12.6.2. Records to be retained for the life of the facility

a. Gaseous and liquid radioactive waste released to the environs.

b. Results of off-site environmental monitoring surveys.

c. Radiation exposures for monitored personnel and associated radiation and contamination surveys used in support of personnel radiation monitoring.

d. Fuel inventories and transfers.

e. Drawings of the reactor facility.

12.6.3. Records to be retained for at least one (1) license period of six (6) years

Records of retraining and requalification of certified operating personnel are maintained at all times the individual is employed, or until the certification is renewed. These records are retained for at least one license period of six years.

12.7. Emergency Planning

An Emergency Plan for the PULSTAR reactor has been developed and is on file with the Nuclear Regulatory Commission.[12-6] The PULSTAR Emergency Plan is consistent with NRC Regulatory Guide 2.6 *Emergency Planning for Research and Test Reactors*, ANSI/ANS 15.16-2008 *Emergency Planning for Research Reactors,*[12-8] and NRC NUREG 0849 *Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors.*[12-9]

From the approved Emergency Plan, emergency procedures were developed and are implemented to maintain a state of readiness in case of an emergency. Emergency procedures include response and activation of the emergency organization, notifications, release of information, emergency classification, training, drills, and equipment inventory.

The Emergency Plan is reviewed and approved by both the RSC and RSAC and the NRC. Emergency procedures are reviewed and approved by RSAC. RSAC audits the Emergency Plan and emergency procedures biennially.

Site specific emergency action levels (EAL) are identified in the PULSTAR Emergency Plan. Trained and qualified personnel activate the PULSTAR Emergency Plan if an EAL has occurred or is imminent.

Emergency response includes on-campus and off-campus organizations including the NRP, NCSU Police, NCSU Environmental Health and Safety, NCSU News Services, City of Raleigh Fire Department, Rex Hospital, Wake County Emergency Management, and the State of North Carolina Emergency Management. Letters of agreement from off-campus support organizations are maintained and updated as required in the Emergency Plan.

Training is conducted at least biennially and drills are conducted annually. Emergency equipment is located within the reactor facility, Burlington Engineering Laboratory, and at the NCSU Environmental Health and Safety Building. Records are maintained as required by the Emergency Plan, including changes to the Emergency Plan and emergency procedures, training, drills, and equipment inventory.

12.8. Security Planning

A Security Plan for the PULSTAR reactor has been developed and is on file with the Nuclear Regulatory Commission.[12-10] From the approved Security Plan, security procedures have been developed and are implemented to maintain security. As required by NRC regulations, the Security Plan and security procedures are withheld from public disclosure.

The Security Plan addresses requirements given in applicable NRC regulations including those for physical security, background checks and investigations of personnel with unescorted access or access to safeguards information, and control of safeguards information. Security requirements for radioactive materials and special nuclear materials are implemented by use of approved procedures.

Security events are reported to the NRC as required by the Emergency Plan, Security Plan, applicable regulations, and facility license conditions.

Records are maintained as required by the Security Plan, including changes to the Security Plan and security procedures, background checks and investigations of personnel with unescorted access or access to safeguards information and documentation of required training and testing.

Security regarding transportation of radioactive materials and hazardous materials follows requirements given in applicable regulations, e.g. 10 CFR Part 20, 10 CFR Part 71, and 49 CFR Part 172. These requirements are implemented by use of approved documents, including Health Physics procedures, Security Plan for Shipment of Hazardous Material at the NCSU PULSTAR reactor, and Quality Assurance Program for Packaging and Transportation of Radioactive Material.

12.9. Quality Assurance

The PULSTAR reactor was manufactured by the American Machine and Foundry Company (AMF) and its design, fabrication, and installation were based on the proven PULSTAR reactor located at the Buffalo Materials Research Center (BMRC) at the State University of New York at Buffalo (SUNYAB). AMF was the designer, fabricator, and installation and checkout supervisor of the NCSU PULSTAR reactor. AMF furnished all of the 4% enriched reactor fuel. Both the 4% and 6% enriched reactor fuel were produced by Canadian Westinghouse Company Limited in Port Hope, Ontario, Canada. Fuel specifications and quality control documentation is on file with the Nuclear Reactor Program (NRP).[12-11]

At the time of construction of the PULSTAR reactor, NCSU defined and implemented a Quality Assurance Program (QAP)[12-12] with a quality assurance coordinator (QAC) who directed/coordinated all quality assurance measures for the project, both on-site and off-site, as they related to the nuclear safety and facility operational aspects. The three levels of QA were; control by contractor, surveillance by design engineers, and audit by the QAC. The Quality Assurance Program is discussed in detail in Section 3.1.1.

Quality Assurance (QA) at the PULSTAR reactor is implemented by including specific requirements in facility procedures, design changes to the facility, and experiments to assure that safety related activities are performed in a manner that maintains quality. Specific requirements may include the use of applicable codes and standards for the activity. Associated documents are reviewed and approved by RSAC and RSC as described previously. Records of these activities are audited by RSAC.

To support 2 MW operations, equipment in the primary coolant system and secondary coolant system were updated and relocated within the facility. NCSU Facilities Design and Construction Services provided oversight while qualified contractors performed the modifications. The designers and engineers were Enercon Corporation of Kennesaw, Georgia and Edmondson Engineers of Durham, North Carolina.

All changes and modifications to the facility have been made in accordance with applicable regulations, license conditions, and accepted engineering and building practices and codes. Changes to the facility are documented and reviewed and approved by the RSAC and RSC, as necessary. These include changes to procedures, facility design, experiments and experimental facilities, and license documents. License amendments have been made and approved by the NRC prior to being implemented, as required.

A Quality Assurance Program for Packaging and Transportation of Radioactive Materials as required by 10 CFR Part 71 has been documented and approved by the NRC.

12.10. Operator Training and Requalification

An operator training and requalification program for the PULSTAR reactor has been developed and is on file with the Nuclear Regulatory Commission (NRC).[12-2] This program meets the requirements of 10 CFR Part 55 as applied to research reactors and follows ANSI/ANS 15.4 *Standard for Selection and Training of Personnel for Research Reactors.*[12-3] Deviations from 10 CFR Part 55 are justified by the mode of operation and unique design of the PULSTAR reactor.

All operators undergo a selection, training and certification program prior to unsupervised operation of the PULSTAR reactor. All licensed operators participate in a requalification program conducted over a period of two years. The requalification program is then followed by successive two year programs. The operator training and requalification program consists of lectures, quizzes, examinations, and document review.

Lectures include theory and principles of operation, general and specific operating characteristics, reactor instrumentation and control, reactor protection systems, engineered safety features, operating procedures (normal, abnormal, and emergency), radiation control and safety, technical specifications, administrative controls, and applicable regulations.

Written quizzes and exams are conducted. The written exam is similar in content to the NRC licensing exam. Reactor operator (RO) and senior reactor operator (SRO) exams are conducted. The SRO exam contains questions covering additional subjects.

On-the-job training is accomplished by routine operation of the PULSTAR reactor. Each licensed operator maintains proficiency by operating the reactor a minimum number of hours each calendar quarter. SROs serve a minimum number hours each calendar quarter as the designated SRO (DSRO). In addition, each RO and SRO performs a minimum number of reactivity manipulations in any combination of reactor startup, shutdown or power changes annually. Direct supervision of reactivity manipulations as DSRO is considered equivalent to actual performance. Annually, each licensed operator controls the reactor during power changes and performs walk-throughs of the reactor in a number of simulated conditions (e.g. reactor operation or facility equipment failures). Finally, oral and demonstrational exams are conducted biennially for each licensed operator for a number of tasks to evaluate the ability of the licensed operator to satisfactorily operate the reactor.

Designated documents are reviewed semi-annually by each licensed operator. Changes to the facility procedures, design changes, reactor license and technical specifications are routed to operators for review in a timely manner.

Provisions for remedial training and reexamination are included in the requalification program. Failure will result in the operator being relieved from licensed activities. The requalification program Administrator is exempt from written, oral, and demonstrational examinations. Records of the training and requalification program are maintained as specified in Section 12.6.3.

A Reactor Operator Assistant (ROA) is an individual that has been trained to assistant the reactor operator in performing non-NRC licensed activities. Examples of such activities might include: sample loading and unloading, pneumatic sample operation, checking and changing liquid nitrogen dewars, starting and stopping makeup water to the primary coolant system, performing building checks, verifying proper operation of equipment such as, cooling pumps, cooling tower, and air compressor. The ROA is also trained in abnormal conditions and will assistant in implementing the evacuation of the Reactor Building. The training of the ROA is documented using facility procedures.

12.11. Startup Plan

12.11.1. 1 MW Operations

This section remains in this safety analysis report for historical reference. The initial startup included acceptance and integrated systems testing, initial core loading and criticality, approach to power, and the confirmation of reactor parameters. These tests were carried out by the regular staff of the Department of Nuclear Engineering under the direction of the Nuclear Operations Administrator (now referred to as the Director). Supporting and advising the Nuclear Operations Administrator was a PULSTAR qualified reactor contractor (AMF) field engineer, a cold licensed (pre-critical) senior reactor operator, and a cold licensed reactor operator from the Department. The operators had either completed qualification training on a similar reactor prior to initial startup or had previous research reactor experience. After initial operations and a sufficient period of training, a staff of operators was qualified by the Department. The methods and procedures for operating and maintaining the PULSTAR are based on manuals provided by the equipment vendors and specific operating procedures developed by the PULSTAR staff.

12.11.1.1. Initial Tests and Operations

An initial testing and operation program was performed on the reactor and auxiliary equipment to ensure proper operation and to verify the operational characteristics of the core. For historical purposes, this section will continue to summarize the initial battery of tests required in support of the initial operating license of the PULSTAR reactor facility.

The following tests were performed prior to loading fuel in the core. Each test is followed by a written statement of the purpose of the test. The procedures followed in these tests are contained in the facility Startup Manual.

a. Reactor Building Ventilation Test

   The ventilation system capacity was tested under normal and confinement modes.

b. Radiation Monitoring System

   The radiation monitoring system was tested using radioactive sources and the alert and alarm setpoints were set.

c. Primary and Secondary Coolant

   The Primary coolant system flow, flapper valve, and tank level alarm setpoint were adjusted along with a verification of successful pump operation. The secondary coolant system pump and cooling tower fan operability were verified. In addition, the primary demineralizer flow rate was adjusted and resistivity control verified.

d. Console and Nuclear Instrumentation

   All console alarm settings were made and acceptable performance verified. The neutron detectors were source checked. A signal generator was used for checkout of the pulse channel response time. The various interlocks for different modes of operation were also checked.

e. Control and Pulse Rod Drive Packages

   The alignment, drop time, rod travel speeds, pulse ejection time and proper operation were tested.

f. Evacuation Procedure

A practice evacuation was made to verify the adequacy of the system and to identify any potential problems and/or improvements with the system.

After the tests above were successfully completed, the initial criticality test program proceeded. The initial critical test program covered the initial loading of fuel through the zero power tests. The tests are listed below along with a brief statement of the test purpose. Detailed procedures for these tests are contained in the PULSTAR Startup Manual.

a. Initial Approach to Criticality

The reactor was loaded with fuel in a deliberate, safe and controlled manner. After criticality was attained and initial power determination was made, nuclear instrumentation was checked for proper operation.

b. Excess Reactivity Measurement

Upon completion of initial criticality, the 5×5 Standard Core loading was completed by loading additional fuel, and control rod calibrations were performed to determine excess reactivity. The results of these tests are detailed in the PULSTAR Startup Manual. Shutdown Margin with one stuck rod criteria was verified for this initial core loading.

c. Flux Measurements

Flux measurements were made for the 5×5 Standard Core loading by irradiating and analyzing copper flux wires. This allowed the determinations of the core peaking factors.

d. Experimental Facilities Worth

The reactivity effect of the various experimental facilities was measured. The reactivity of each beamtube was determined by the change in critical rod height associated with draining and filling the beamtube.

e. Pulse Rod Calibration

The pulse rod was calibrated for both the 5×5 Standard Core and the 5×5 Pulsing Core with the pneumatic air supply disconnected.

f. Power Calibration

The flux measurements referenced above were used to generate an initial estimate for adjusting the position of the power detectors.

g. Approach to 1 MW Steady-state Operation

i. Natural Convection Cooling

Using the 5×5 Standard Core loading, the reactor power was increased in 50 kW increments up to 250 kW. Control rod positions were recorded to determine the power defect associated with the power level.

ii. Forced Cooling

After the core was operating at 100 kW with natural convection cooling, the power was reduced to zero and the primary pump started to establish forced cooling. The power level was then increased in 100 kW increments. A power check was made at each power level using the flow × ΔT technique. Nuclear instruments were then adjusted to agree with this power level check. Doppler and power coefficients were verified to be negative and their values estimated on this power increase.

h. 1 MW Shielding and Building Survey

After achieving the 1 MW power level, a shield survey was made to verify the adequacy of the Biological Shield. Radiation readings throughout the Reactor Bay were measured and the mechanical equipment room (MER) was also surveyed.

i. Xenon Poison Reactivity

During the initial power operation, control rod positions were recorded to determine the xenon poison reactivity.

j. Automatic Control

After 1 MW operation was completed, the reactor power was reduced to 500 kW for a checkout of the automatic control system.

k. Square Wave Test

It was initially envisioned that square wave startup testing would be completed during the initial checkout. However, this testing was not completed and the PULSTAR reactor has no plans to perform a square wave startup.

l. Pulse Tests

A pulse test program was executed to reach the design pulse. The test consisted of incremental increases in the reactivity step input to the core. Calibration of the pulse measuring channel had already occurred as a part of the steady-state tests. After each reactor pulse, the peak power and total energy release was recorded and used to develop pulsing curves.

m. Approach to Design Pulse

During the approach to the design pulse, auxiliary equipment was used to determine the pulse shape, peak power, peak energy release and total energy release. The data was taken from a Visicorder. The reactor period was then determined from the pulse traces.

n. Measurement of 1/βeff

The data measured during the approach to the design pulse was plotted to determine the $1/\beta_{eff}$ for the reactor core loading.

o. Design Pulse

After reaching the reactivity insertion for the design pulse, this amount was repeated ten times to check repeatability of the pulse.

p. Fuel Pin Inspection

The hottest fuel pin identified in the 5×5 Standard Core loading was examined following the ten pulses to verify no physical changes had resulted from the design pulsing.

q. Post-critical Tests

Following completion of the above test program, the operation data was reviewed to identify any improvements that could be made to increase the safety of the reactor. In addition, normal operating background radiation levels versus their respective alarm setpoints were verified.

r. Operating Restrictions

Throughout the initial test program, the operating restrictions as identified in the PULSTAR technical specifications were adhered to.

12.11.2. 2 MW Operations

A test program for reactor operation at 2 MW has been developed and shall be maintained. This program consists of reviewed and approved documents by RSAC, e.g. design changes and procedures, similar to those performed initially and for the approach to full power. Documentation and records, e.g. procedure forms, logs, etc., are maintained.

The majority of the systems and components will not be affected by the increase from 1 MW to 2 MW. All systems and components will be evaluated and those that are affected in the increase from 1 MW to 2 MW will be incorporated into the startup plan. The following modifications and tests will be performed:

a. Primary and Secondary Coolant

The primary coolant system flow rate will be increased from 500 gpm to 1000 gpm and corresponding alarms and scram setpoints will be adjusted to be consistent with the limiting conditions for operations. The secondary system flow rate will also be increased from 700 gpm to 1000 gpm.

b. Console and Nuclear Instrumentation

Nuclear instrumentation will be modified as necessary to display proper ranging for 2 MW. Alarm and scram setpoints will be adjusted to be consistent with the limiting conditions for operations. The various interlocks for different modes of operation will also be checked.

c. Approach to 2 MW Steady-state Operation

i. Natural Convection Cooling

No new testing will be performed for natural convection cooling mode of operation. All current limits and restrictions will remain.

ii. Forced Cooling

Using an approved core loading pattern, the reactor power will be increased in 100 kW increments up to 1000 kW. Control rod positions will be recorded to determine the power defect associated with the power level and flow rate. A calorimetric will be performed at 1 MW and nuclear instrumentation will be adjusted accordingly.

Following the calorimetric, the power level will be increased in 100 kW increments. A power check will be made at each power level using the flow × ΔT technique. A calorimetric will again be performed at 1.8 MW and nuclear instrumentation will be adjusted as necessary.

Nuclear instruments will be adjusted to agree with these power level checks. Reactivity coefficients will be verified to be negative and their values estimated during this evolution.

d. 2 MW Shielding and Building Survey

After achieving the 1.8 MW power level, a shield survey will be made to verify the adequacy of the biological shield. Radiation readings throughout the reactor bay, mechanical equipment room (MER), primary piping vault and exterior of the building will be surveyed.

e. Xenon Poison Reactivity

The reactor will be operated continuously to determine the xenon poison reactivity behavior at 2 MW.

12.12.Environmental Reports

An updated environmental impact statement has been made as required by 10 CFR Part 51 for relicensing of the reactor[12-13] This statement provides a description of the facility, environmental effect of the facility site preparation and construction, environmental effects of facility operation, alternatives to construction and operation of the facility, and cost and benefits of facility alternatives. The environmental impact statement concludes that there is no significant environmental impact associated with the relicensing and increase in maximum licensed power to 2 MW of the reactor.

Routine environmental monitoring of the facility is summarized in the annual operating reports. These reports have been submitted from since the initial startup of the reactor facility in 1972 to the present time. These reports concluded that environmental limits have not been exceeded and that the PULSTAR reactor has been operated safely.

12.13.References

12-1 ANS/ANSI-15.1-2007-*The Development of Technical Specifications for Research Reactors.*

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12-3 ANS/ANSI-15.4-2007-*Selection and Training of Personnel for Research Reactors*.

12-4 North Carolina State University PULSTAR Reactor, *Radiation Protection Program.*

12-5 North Carolina State University, *Radiation Safety Manual.*

12-6 North Carolina State University PULSTAR Reactor, *Emergency Plan*, February 24, 2017.

12-7 North Carolina State University, *Strategic Energy and Water Annual Report*, 2015, https://sustainability.ncsu.edu/sustaindev/wp-content/uploads/2015/10/2015-NC-State-Energy-and-Water-Report.pdf.

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12-9 NUREG-0849, *Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors*, October 1983, https://www.nrc.gov/docs/ML0621/ML062190191.pdf.

12-10 North Carolina State University PULSTAR Reactor, *Security Plan*, February 24, 2017

12-11 American Machine and Foundry Co., *Specifications for PULSTAR Fuel Assemblies APR-1 Revision 5*, April 1970.

12-12 North Carolina State University-Nuclear Science and Engineering Research Center, *Quality Assurance Instructions Manual*, July 1, 1969

12-13 North Carolina State University PULSTAR Reactor, *Environmental Report*, March 2, 2017

13. ACCIDENT ANALYSES

13.1. Accident Initiating Events and Scenarios

In this section, the results of select accident analyses are presented that show that the health and safety of the public are protected in the event of a credible accident at the PULSTAR Reactor facility. The facility design features, safety limits, limiting safety system settings, and limiting conditions for operation have been selected to ensure that no credible accident, up to and including the Maximum Hypothetical Accident (MHA), could lead to unacceptable radiological consequences to people or the environment.

The accidents analyzed were selected upon review of guidance provided in NUREG 1537[13-1] which gives general categories of postulated accidents to be considered. Additional guidance in ANS/ANSI-15.21 Appendix C[13-2] provides a detailed list of initiating events for postulated accidents which was evaluated to determine credible scenarios to be analyzed in each accident category. All credible accident scenarios were considered and the cases with the most limiting and bounding conditions in each category were analyzed and are presented. Analyses of these accidents and discussions of their consequences are given in Section 13.2.

1. Insertion of excess reactivity
2. Loss of coolant
3. Heat exchanger pressure boundary breach
4. Loss of coolant flow
5. Mishandling or malfunction of fuel
6. Experiment malfunction
7. Loss of normal electric power
8. Release of radioactive waste to unrestricted areas
9. External events
10. Mishandling or malfunction of equipment

13.1.1. Maximum Hypothetical Accident

The maximum hypothetical accident (MHA) associated with the PULSTAR reactor facility is one that results in radiation dose to personnel or members of the public. Accident scenarios that could result in radiation doses are:

1. Complete loss of coolant (LOCA) leading to the uncovering of the reactor core.
2. A fuel cladding failure accident (CFA) resulting in fission product release.
3. A fueled experiment failure resulting in the release of activation products, fission products, and the target fissionable material.

Of these, the released activity for the CFA exceeds that for the fueled experiment failure, therefore the fuel cladding failure accident results in the greatest potential for significant radiation doses to personnel and to the public and is considered the MHA for the PULSTAR Reactor. The analysis for this accident is detailed in Section 13.2.6 and it is concluded that that postulated 24 hour radiation dose is within 10 CFR Part 20 dose limits.

13.1.2. Insertion of Excess Reactivity

Insertion of reactivity events can be either step insertions where a large amount of reactivity is rapidly inserted into the core, or ramp insertions where reactivity is continuously inserted into the core. The following reactivity insertion initiating events are considered:

1. Fuel loading accident
2. Continuous withdrawal of control rods
3. Control rod failure
4. Cold primary coolant slug
5. Experiment malfunction and mishandling
6. Fuel storage

Detailed analyses of these reactivity insertion events are given in Section 13.2.2.

13.1.3. Loss of Coolant

A loss of coolant accident (LOCA) resulting in a complete loss of pool water would uncover the core. Primary concerns with a LOCA are core decay heat and radiation levels inside and outside the reactor facility.

For this accident scenario, it is assumed that the reactor operates with a 25 fuel assembly core at a power level of 2.0 MW for an infinite time, and that fission product concentrations attain equilibrium within the fuel pins. Postulated failures leading to complete and sudden reactor pool drain would be a large scale breach, e.g. in a reactor pool penetration or primary coolant system pipe or component. As a result, the failure is postulated to drain the reactor pool suddenly into the reactor building, exposing an unshielded reactor core. Maximum clad temperatures remain well below the safety limit of 2200 °F.[13-3][a] All of the drained primary coolant would remain in the lower levels of the reactor building.

A detailed analysis of the loss of coolant accident is given in Section 13.2.3.

13.1.4. Heat Exchanger Pressure Boundary Breach

The heat exchanger is composed of plates which separate the primary coolant water from the secondary coolant water. Heat from the primary coolant is transferred through the plates to the secondary coolant, which then removes the heat through evaporative cooling at the cooling tower. The cooling tower is located adjacent to the reactor building.

Depending on the failure mechanism, the primary water can either leak to the floor in the mechanical equipment room or into the secondary cooling system. Only in the unlikely event of a hole developing in a plate would water leak from the primary system into the secondary system.

A detailed analysis of the heat exchanger pressure boundary breach accident is given in Section 13.2.4.

[a] Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

13.1.5. Loss of Coolant Flow

Loss of coolant flow could occur due to failure of a key component in the reactor primary system, e.g. a pump seizure, an accidental closing of a primary valve, operator error, loss of electrical power, or from a blocked inlet to a fuel assembly.

A detailed analysis of the loss of flow accident is given in Section 13.2.5.

13.1.6. Mishandling or Malfunction of Fuel

Initiating events in this class are dropping or otherwise causing damage to a fuel assembly, or operation with damaged fuel, such as a water-logged fuel pin.

13.1.6.1. Cladding Failure Accident

Fuel damage associated with a cladding failure accident (CFA) is postulated to occur from mishandling or mechanical shock. Failure of fuel pin cladding is assumed to occur by impacting an assembly during fuel movement. The design basis accident for the release of fission products is the failure of all 25 pins in one fuel assembly with maximum burnup of 20,000 MWd/MTU. The design bases accident would be detected shortly after occurring by the radiation monitoring system with activation of the evacuation alarm. A less sudden release of fission products would be detected by routine water chemistry and isotopic analysis of the primary coolant and air samples and by higher readings on the radiation monitoring system.

Fuel cladding failure results in the greatest potential for significant radiation doses to personnel and to the public, therefore it is considered the MHA for the PULSTAR Reactor. The analysis for this accident is detailed in Section 13.2.6 and it is concluded that the postulated 24 hour radiation dose is within 10 CFR Part 20 dose limits.

13.1.6.2. Water-logging

Such failure could occur in the event of a defect or leak in the fuel cladding, which would permit in-leakage of water during low power operations or at shutdown. Operation with water-logged fuel is a concern since the sudden addition of energy to the fuel may cause the water to turn quickly to steam and damage the fuel cladding.

A detailed analysis of the water-logging accident is presented in Section 13.2.6.2.

13.1.7. Experiment Malfunction

Experiment malfunction or failure may result in personnel exposure to radioactive material causing an external dose and/or an internal dose. External dose is associated with the loss of shielding, exposure to a high activity sample or exposure to an open beam tube. Internal dose is associated with the release of radioactive material. Administrative experiment limitations and engineering controls are in place to prevent these scenarios. Experiments are conducted after being reviewed by the reactor staff and review committees as required by the technical specifications and as described in Sections 11 and 12.

Accidental release of fission products associated with a fueled experiment failure is postulated to occur from a non-violent, encapsulation failure, e.g. a rupture or large leak. If encapsulation fails, all fission gas activity and halogen activity is assumed to be immediately released into the reactor building. The design basis for this accident assumes the fission gas and halogen activity is saturated. A release of fission gases and halogens from the design basis accident would be detected shortly after occurring by the radiation monitoring system with activation of the evacuation alarm and confinement ventilation system. A less sudden release of fission product activity would be detected by routine air sample analysis of the reactor building and airborne effluent and by higher readings on the radiation monitoring system.

A detailed analysis of the experiment malfunction accident is presented in Section 13.2.7 and in the *Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor*[13-24].

13.1.8. Loss of Normal Electrical Power

The design of the PULSTAR reactor is such that the reactor can be shut down and safely maintained in a shutdown condition under a complete loss of electrical power. There are no electrical power supplies that are critical for maintaining the facility in a safe shutdown condition, even for extended periods of time.

A discussion of the loss of normal electrical power is presented in Section 13.2.8.

13.1.9. Release of Radioactive Water to Unrestricted Areas

Release of radioactive water to unrestricted areas may occur from the following scenarios:

1. Primary coolant is assumed to enter the secondary coolant by a failure of the heat exchanger. The primary coolant overflows the cooling tower basin spilling into nearby storm drains. This scenario has a maximum leak volume of 6300 gallons.
2. Overflow of the waste tanks may occur if waste water is pumped with the waste tanks completely full. The waste water would spill from the waste tank vent onto the ground and paved areas outside the reactor building. Waste water spilled on the paved areas will enter a nearby storm drain.
3. A spill may also result from failure of the waste tank equipment, e.g. valves or piping. In this scenario, waste water spills into the waste tank vault and then enters the storm drain in the waste tank vault. The maximum volume for a single spill would be 300 gallons.
4. A breach in the primary coolant system or a leak from the reactor pool that causes water to enter the ground is also postulated, e.g. from a leak in the primary piping section located in the reactor foundation pad.

The reactor pool level is continuously monitored and has an alarm setpoint at -36 inches, or approximately 2200 gallons below the maximum pool level. Upon receipt of the alarm, reactor staff are notified and respond. The reactor primary coolant and secondary coolant systems have several isolation valves to prevent water loss. Waste water tank levels are monitored by manual methods. Isolation valves are used to control the filling and draining of waste water tanks. Environmental monitoring is performed as described in Section 11 for the Rock Branch creek and the on-site ground water monitoring well. Notifications are made as required by facility procedures that meet applicable regulations and licenses.

A detailed analysis of the release of radioactive water to unrestricted areas is given in Section 13.2.9.

13.1.10. External Events

External events considered that may affect the reactor facility include natural phenomena, activities in the surrounding area, and security related incidents. Natural phenomena are associated with meteorology, hydrology, geology, or seismology. Accidents in the surrounding area are associated with vehicle and air traffic, and activities in NCSU laboratories and business and manufacturing in the surrounding area.

There are no postulated external events that would adversely affect the reactor as described in Section 2. Emergency and Security Plans are in place to prevent, mitigate, and respond to external events.

A discussion of external events is presented in Section 13.2.10.

13.1.11. Mishandling or Malfunction of Equipment

No credible accident initiating events were identified for this class of accidents that are not already discussed for other accident scenarios, e.g. pump failure resulting in a loss of flow. Situations involving an operator error at the reactor controls, a malfunction or loss of safety related instruments or controls, and an electrical fault in the control rod system were anticipated during the reactor design stage. As a result, many safety features such as control system interlocks and automatic reactor shutdown circuits were designed into the overall control system.

A discussion of mishandling or malfunction of equipment is presented in Section 13.2.11.

13.2. Accident Analysis and Determination of Consequences

13.2.1. Maximum Hypothetical Accident

The maximum hypothetical accident (MHA) at the NCSU PULSTAR reactor is due to the radiation dose resulting from a fuel cladding failure accident. The analysis is discussed in detail in Section 13.2.6.1.

13.2.2. Insertion of Excess Reactivity

13.2.2.1. Fuel Loading Accident

The fuel loading accident is an excursion type accident associated with the NCSU PULSTAR reactor. It is calculated that a twenty five assembly core of PULSTAR fuel is capable of absorbing a step input of more than $2.2 (1612 pcm) without failure of the fuel pin cladding.[13-4] Three power/flow scenarios were evaluated.[13-4] It was determined that the scenario for a step reactivity insertion of 1612 pcm from hot zero power at limiting conditions without forced flow cooling was most limiting.

The core fuel loading pattern is strictly controlled by facility procedures. An MCNP core model is generated prior to the loading of each core to confirm that all parameters are within technical specification limits.[13-5,13-6] A reactivity insertion of 1600 pcm into a critical core is equivalent to a k_{eff} of 1.01626, as shown in Equation 13-1, and is the limit given by TS 3.1.e which states that the worth of a fuel assembly while being loaded into the reactor grid plate shall not result in a k_{eff} of greater than 1.01626.

$$k_{eff} = \frac{1}{1-\rho} \qquad \textit{Equation 13-1}$$

$$k_{eff} = \frac{1}{1 - 0.01600\ \Delta k/k} = 1.01626$$

This specification applies to the loading of fuel assemblies inserting reactivity above the minimum required to bring the reactor critical. For the MCNP model of Reflected Core No.11, the last assemblies to be loaded in row F of the grid plate have calculated worths of less than 1228 pcm. A fuel loading accident involving one of these assemblies would result in a step insertion of less than 1600 pcm which would be bounded by this analysis. Fuel assemblies loaded prior to the minimum critical loading could not bring the reactor critical, and therefore could not cause a step insertion accident.

For the accident analyses cases, hot zero power refers to a primary coolant temperature at the Limiting Safety System Setting (LSSS) of 117 °F, which is more limiting than the hot zero coolant temperature at nominal conditions of 100 °F. For this analysis, zero power is defined to be 100 watts. All other initial conditions are at the LSSS setpoint or technical specification limit and are given in Table 13-1.

The following parameters were investigated:

1. Maximum instantaneous power level (MW)
2. Maximum hot pin fuel centerline temperatures (°F)
3. Maximum hot pin cladding temperatures (°F)

For this scenario it is assumed that mishandling of the fuel occurs under the following conditions:

1. The core has been loaded in the optimum configuration that places the core and associated parameters at the technical specification limits.
2. The reactor is critical and operating under the conditions listed in Table 13-1.
3. A fuel assembly is rapidly inserted into the core.
4. The hot pin in the core is assumed to have a peaking factor at the technical specification limit of 3.0.

The step insertion happens at time equal to 0.0 seconds. The flow/flapper scram signal actuates at the LSSS setpoint of 0.25 MW. The rods begin to drop after a 0.05 second scram circuitry delay. Control rod drop times are at the technical specification limit of 1.0 seconds. The complete event sequences are summarized in Table 13-2.

The analysis shows that the power excursion related to the 1600 pcm step insertion is turned over by negative temperature coefficient (Doppler) of the fuel and void formations in the core. The reactor scram fully inserts the control rods within 1.0 second and ensures a stable subcritical reactor.

For this scenario, reactor power, shown in Figure 13-1, peaks at 3186.3 MW resulting in a maximum cladding temperature of 1989.5 °F, which is well below the safety limit of 2200 °F for zircaloy cladding for PULSTAR type fuel given in NUREG 1537 Appendix 14.1.[13-3][a] The fuel temperature peaks at 3117.9 °F , which is well below the limit of 4352°F given for UO_2 fuel in NUREG 1537 Appendix 14.1.[13-3] Fuel and cladding temperatures for the hot pin are shown in Figure 13-2 and Figure 13-3.

The above analysis indicates that no loss of cladding integrity or fuel damage would result from a step reactivity insertion event.

[a] Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

Table 13-1 – Initialization – Step Reactivity Insertion Accident

Parameters	Hot Zero Power without Flow
Reactor Power (MW)	0.0001
Peaking Factor	3.0
Primary Flow (gpm)	0
Core Inlet Temperature (°F)	117
Height of Water Above the Top of the Core (inches)	204
Secondary Coolant System	off
Step reactivity Insertion (pcm)	1612 ($2.2)
Scram Setpoint (MW)	0.25
Scram Delay (sec)	0.05
Control Rod Drop Time (sec)	1.0

Table 13-2 – Sequence of Events – Step Reactivity Insertion Accident

Parameters	Hot Zero Power without Flow	
	Time (s)	Parameter
Reactivity Insertion	0.0	
Reactor Scram Signal (MW)	0.16	0.25
Scram Rods Start to Drop	0.21	
Maximum Power Level (MW)	0.21	3186.3
Maximum Fuel Temperature (°F)	3.5	3117.9
Maximum Cladding Temperature (°F)	3.0	1989.5

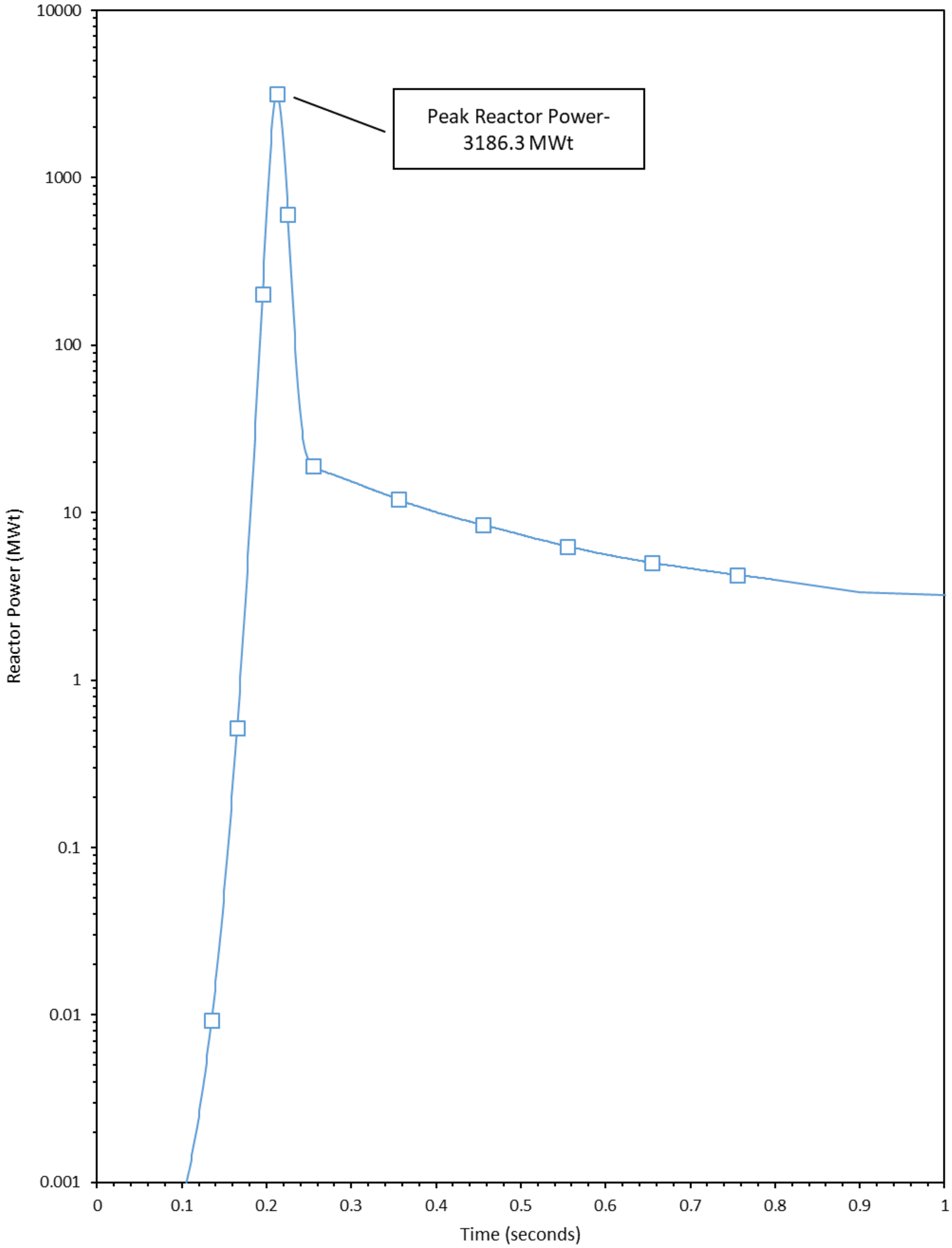

Figure 13-1 – Reactor power level resulting from a step insertion of 1600 pcm for an initial condition of hot zero power without forced flow cooling.

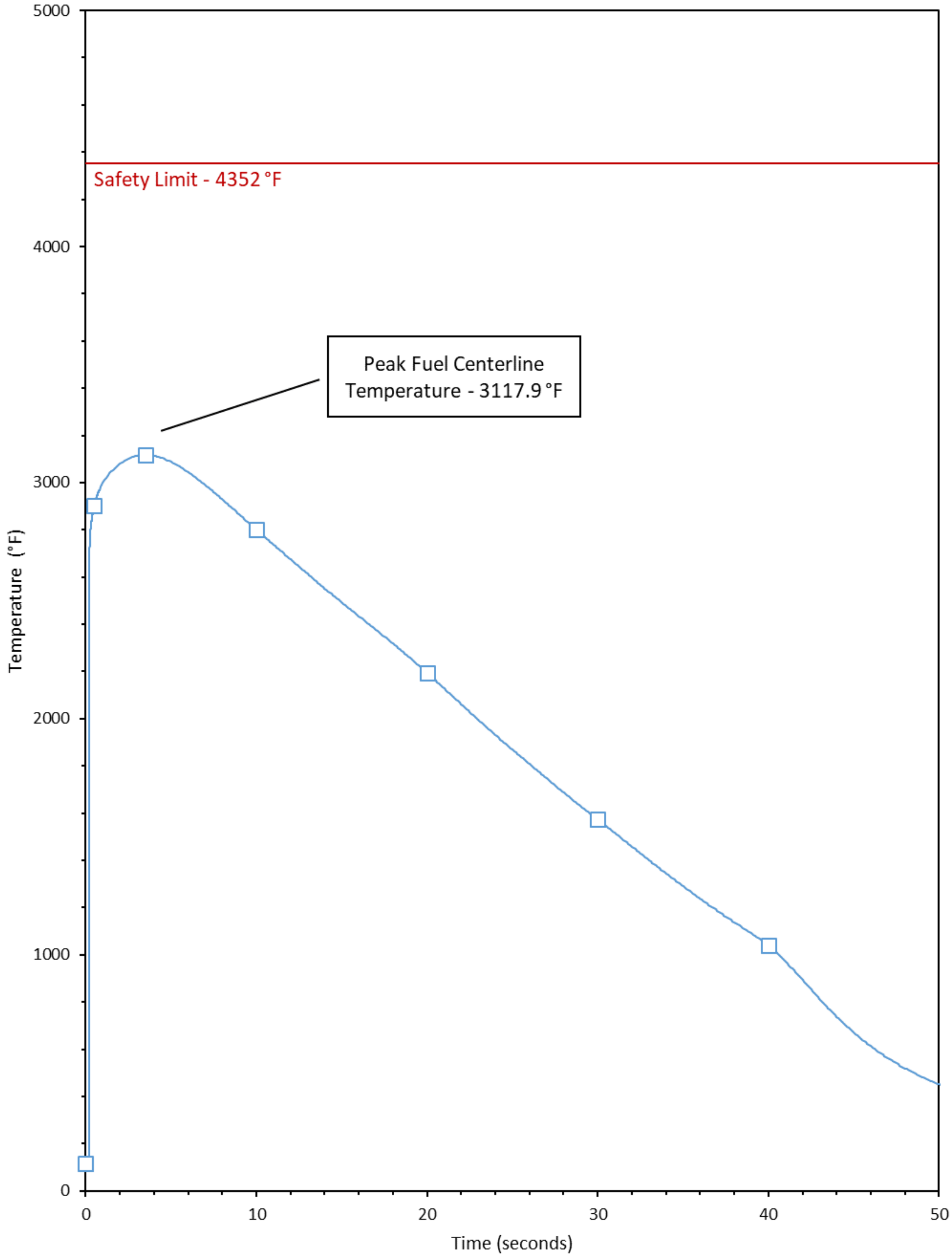

Figure 13-2 – Fuel centerline temperature in the hot channel resulting from a step insertion of 1600 pcm for an initial condition of hot zero power without forced flow cooling.

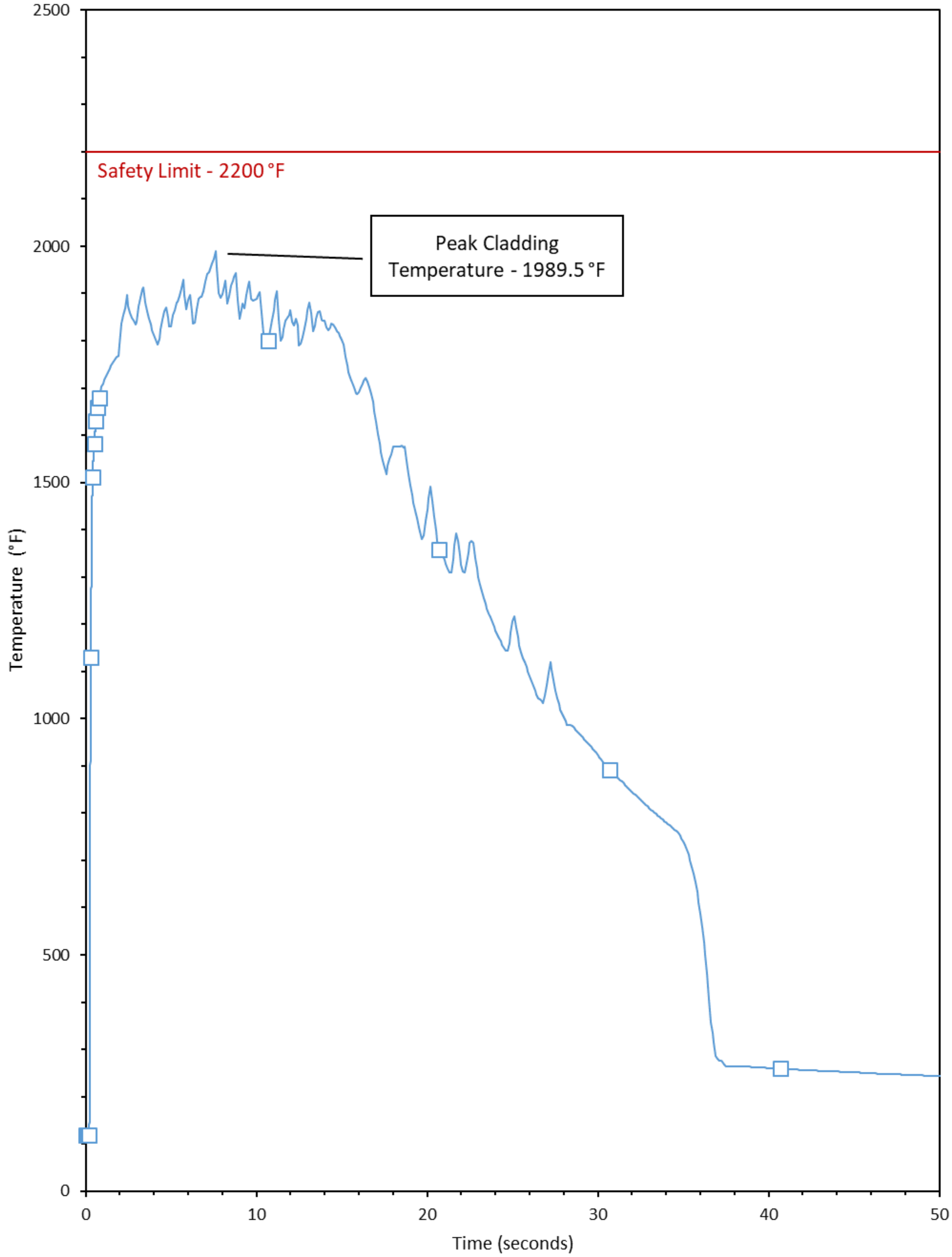

Figure 13-3 – Fuel cladding temperature in the hot channel resulting from a step insertion of 1600 pcm for an initial condition of hot zero power without forced flow cooling.

13.2.2.2. Continuous Withdrawal of a Control Rod

The following accident analysis was made to determine the result of a ramp reactivity insertion due to the continuous withdrawal of all control rods from a critical core at 100 watts until shutdown occurs by the high level neutron flux scrams.

The following parameters were investigated:

1. Maximum resulting power level (MW)
2. Maximum hot pin fuel centerline temperatures (°F)
3. Maximum hot pin cladding temperatures (°F)

The hot zero power at limiting conditions with forced flow scenario was determined to be the most limiting and was evaluated.[13-4]

For the accident analyses cases, hot zero power refers to a primary coolant temperature at the Limiting Safety System Setting (LSSS) of 117 °F, which is more limiting than the hot zero coolant temperature at nominal conditions of 100 °F. For this analysis, zero power is defined to be 100 watts. All other initial conditions are at the LSSS setpoint or technical specification limit and are listed in Table 13-3.

For this accident the following is assumed:

1. The reactor operator has failed to return the linear channel range selector to the most sensitive or mid-scale position. This would mean that the over-power level scrams would not be set at the most sensitive or on-scale position prior to withdrawal of control rods, therefore an over-power level scram would occur at the LSSS scram setpoint of 2.0 MW.
2. It is assumed that the control rods are withdrawn in gang and the gang rate at which reactivity is inserted into the core is at the technical specification limit of 200 pcm/second.
3. The power level of the core is assumed to be 100 watts at the initiation of the event.
4. The core has been loaded in the optimum configuration that places the core and associated parameters at the technical specification limits.
5. The hot pin in the core is assumed to have a peaking factor at the technical specification limit of 3.0.
6. The reactor is critical and operating under the conditions listed in Table 13-3.

The ramp insertion initiates at time equal to 0.0 seconds. The scram signal actuates at the LSSS setpoint of 2.0 MW at time equal to 3.93 seconds and the rods begin to drop after a 0.05 second scram circuitry delay at time 3.99 seconds. Control rod drop times are at the technical specification limit of 1.0 seconds. The controls rods are fully inserted by 4.99 seconds and ensure a stable subcritical reactor. The event sequence is summarized in Table 13-4.

The analysis shows that the excursion is terminated by insertion of control rods due to the actuation of the over-power scrams. As shown in Figure 13-4 the negative temperature coefficient of the fuel is not sufficient to prevent the power increase and the excursion results in a maximum power level of 4.96 MW.

Figure 13-5 shows the transient behavior of the hot pin fuel temperature and hot pin cladding temperature. The power excursion causes an initial increase in fuel temperature before the reactor scrams. Maximum hot pin fuel and cladding temperatures remain well below the limits of 4352°F and

2200°F, respectively, listed in NUREG 1537 Appendix 14.1 for PULSTAR fuel.[13-3][a] Forced convection cooling results in a rapid decrease in fuel temperatures. Maximum fuel and cladding temperatures for the hot pin are listed in Table 13-4.

The above analysis indicates that no loss of cladding integrity or fuel damage would result from a ramp reactivity insertion event.

Table 13-3 – Initialization – Ramp Reactivity Insertion Accident

Parameters	**Ramp Insertion**
	Hot Zero Power with Flow
Reactor Power (MW)	0.0001
Peaking Factor	3.0
Primary Flow (gpm)	900
Core Inlet Temperature (°F)	117
Height of Water Above the Top of the Core (inches)	204
Secondary Coolant System	off
Delayed Neutron Fraction (pcm)	733
Prompt Neutron Generation Time (sec)	36×10^{-6}
Ramp Reactivity Insertion (pcm/sec)	200
Insertion Time Length (sec)	until scram
Scram Setpoint (MW)	2.0
Scram Delay (sec)	0.05
Control Rod Drop Time (sec)	1.0

[a] Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

Table 13-4 – Sequence of Events – Ramp Reactivity Insertion Accident

Parameters	Ramp Insertion Hot Zero Power with Flow	
	Time (s)	Parameter
Reactivity Insertion	0.0	
Reactor Scram Signal (MW)	3.93	2.0 MW
Scram Rods Start to Drop	3.99	
Maximum Power Level (MW)	3.98	4.96 MW
Maximum Fuel Temperature (°F)	4.84	139.5 °F
Maximum Cladding Temperature (°F)	4.35	120.7 °F

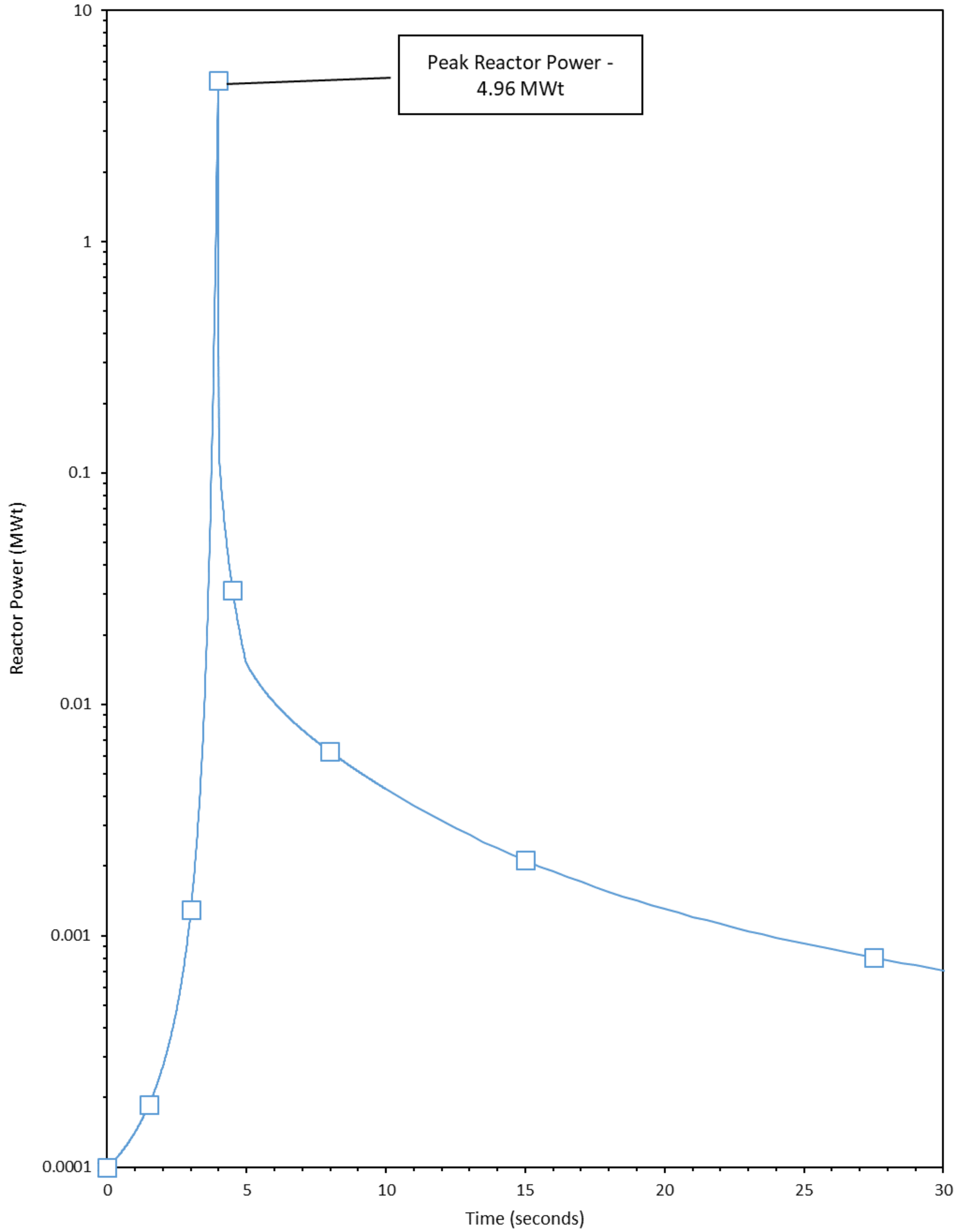

Figure 13-4 – Reactor power level resulting from a ramp insertion for an initial condition of hot zero power with forced flow cooling.

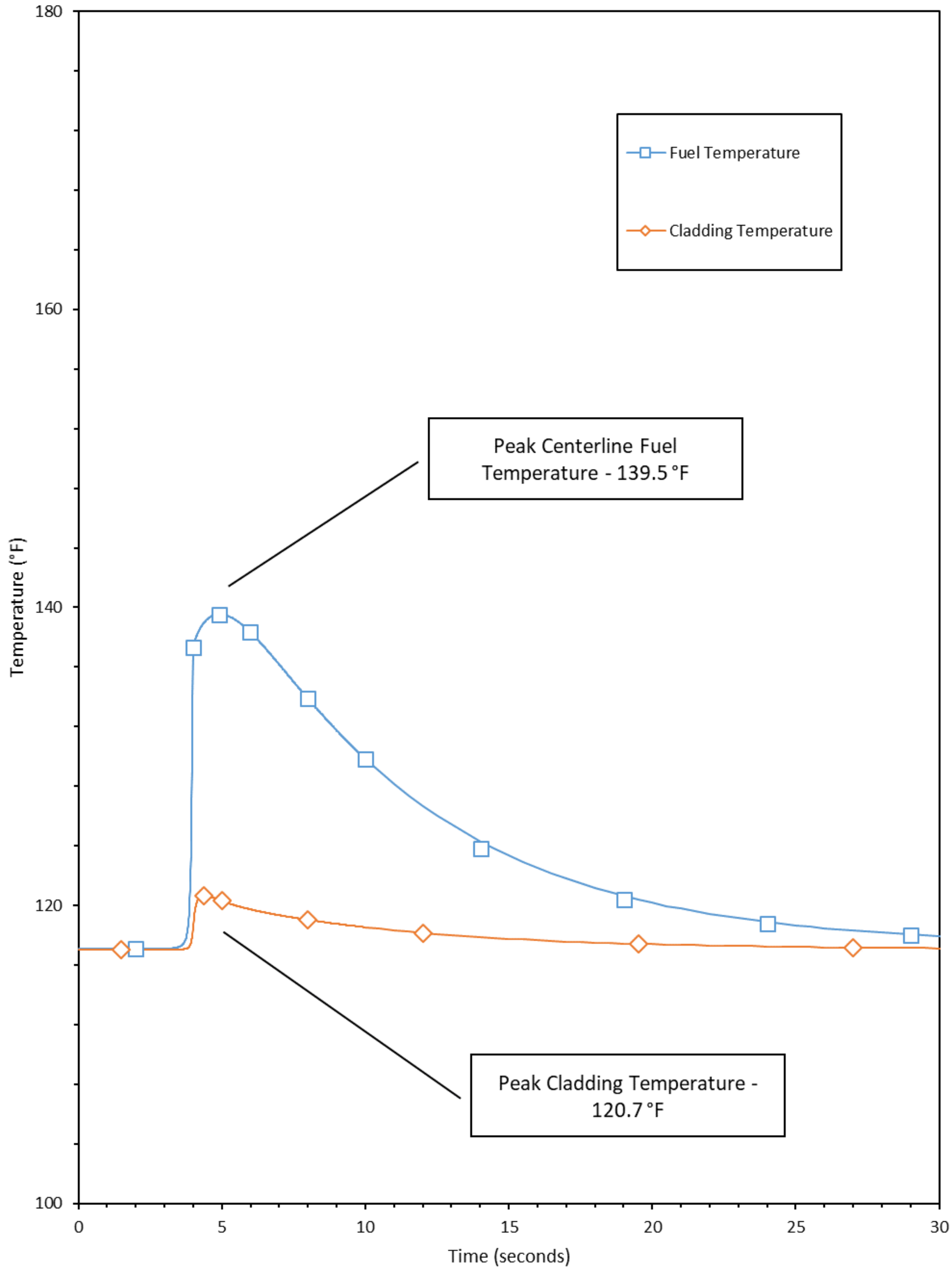

Figure 13-5 – Fuel centerline and cladding temperatures in the hot channel resulting from a ramp insertion for an initial condition of hot zero power with forced flow cooling.

13.2.2.3. Control Rod Failure

In the event a single control rod does not fall under the force of gravity to its fully seated position following a reactor scram, the control rod drives, which automatically insert on scram demand, will attempt to push the stuck rod into the core. Nevertheless, the one stuck rod criterion will ensure that the failure of a single control rod to drop or fully insert after a scram will not result in a hazard to reactor operations since the shutdown margin requirement will be satisfied. Refer to SAR Section 4.5.

The primary coolant provides sufficient cooling of the Ag-In-Cd control rods to prevent melting.

13.2.2.4. Cold Primary Coolant Slug

In the natural convection cooling mode of operation, the Limiting Safety System Setting is 250 kW. From previous test data,[13-7] the average temperature difference across the core at 250 kW is approximately 37°F under natural convection cooling. For this scenario, it is assumed that the reactor is operating in natural convection mode at 250 kW when the flapper is closed and the primary pump is started. Closing the flapper will start forced convection mode such that the 37°F temperature rise across the core will no longer exist. The analysis conservatively assumes that the temperature rise across the core will drop to zero. This decrease in temperature will result in a positive reactivity affect as calculated using Equation 13-2 and summarized in Table 13-5 below.

Table 13-5 – Cold Slug Parameter Values

Parameter	Value
α_T	-3.7 pcm/°F
ΔT	-37°F
ρ_{in}	137 pcm
ρ_{2MW}	-560 pcm
β_{eff}	733 pcm
λ_{eff}	0.1 seconds^{-1}
P_{final}	489 kW

$$\rho_{in} = \alpha_T \times \Delta T_{moderator} \qquad \textit{Equation 13-2}$$

Using the Stable Period Equation 13-3, the change in the core coolant temperature of 37°F would result in a positive reactor period of 44 seconds.

$$\tau = \frac{\beta_{eff} - \rho}{\lambda \cdot \rho} = 44\ seconds$$ ***Equation 13-3***

The reactor would rise on the 44 second period until the Doppler coefficient and moderator temperature reduce the 137 pcm net positive reactivity to zero. Applying a power coefficient of -560 pcm the reactor would level out at approximately 489 kW.

$$P_{final} = 2\ MW \times \frac{\rho}{\rho_{2MW}} = 489\ kW$$ ***Equation 13-4***

The above analysis indicates that no hazard exists from a cold water slug accident.

13.2.2.5. Experiment Malfunction or Mishandling

The maximum reactivity worth of any single experiment is 1600 pcm. For this event, it is assumed that the experiment fails and rapidly inserts 1600 pcm into the reactor core. This excursion accident scenario is bounded by the fuel loading (step insertion) accident analyzed in Section 13.2.2.1.

13.2.2.6. Fuel Storage

Improper storage configuration of PULSTAR fuel could cause an excursion. To eliminate this potential, subcritical wet storage is provided in the PULSTAR pool with [redacted] linear storage racks, one with a capacity of [redacted] assemblies and the other with a capacity of [redacted] assemblies. The k_{eff} for the [redacted] assembly rack is less than 0.28 and 0.35 when loaded with the 4% and 6% enriched fuel, respectively.[13-8][13.9] There are also [redacted] assembly capacity round fuel storage pits at the bottom of the pool liner. The k_{eff} for the storage pit was measured to be 0.77 when loaded with 4% enriched fuel.[13-8] Six percent fuel is not permitted in the storage pits. Therefore, no critical array will exist, and all rack and storage assemblies will be significantly subcritical.

The storage pits and storage racks are designed so that fuel can only be inserted into allowed locations. The fuel installed in the storage pits are positioned well below the top the of the pit to prevent accidental critically from a fuel assembly dropped on top of the pit.

Subcritical dry storage of unused fuel is provided in the [redacted] storage rack. The k_{eff} for the [redacted] storage rack has been calculated to be less than 0.9 for 6% enriched fuel.[13-10] The radiation monitoring system activates the confinement system and the evacuation alarm at gamma radiation levels at or below 0.1 rem per hour. Refer to 9.2.2 for more details.

The above discussion indicates that no hazard exists from an improper fuel storage configuration accident.

13.2.3. Loss of Coolant

In many non-power reactor designs, the loss of coolant accident (LOCA) is of no consequence to fuel integrity because decay heat in the fuel is so small as to be incapable of causing fuel failure. This is the case for the PULSTAR reactor, therefore there is no need for an engineered safety feature such as an emergency core cooling system. At the PULSTAR, reactor possible initiators of LOCAs are the malfunction of a component in the primary coolant loop or malfunction of an experimental facility, such as a beamtube.

A loss of coolant accident resulting in the complete or partial uncovering of the reactor core is possible since the pool liner has several large penetrations and reactor coolant pipes are located at elevations that would permit a near compete drain down in the event of a failure. The potential breaks are large enough that a loss of coolant could occur in a short period of time while a significant amount of heat is still being generated in the fuel pins from fission product decay heat.

This accident was analyzed using the RELAP5/MOD3.3 code of the US Nuclear Regulatory Commission (NRC).[13-11] The report Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor[13-4] describes the RELAP5 model of the PULSTAR systems, initializations, and results of the accident analyses in greater detail.

Key features for the RELAP5 PULSTAR model are summarized below and transient analysis is carried out until stable core cooling is achieved:

1. RELAP5 nodalization of the PULSTAR system is given in the report referenced above.
2. Three break locations are modeled and individual transients are initiated by opening the breaks from full power limiting safety system settings (LSSS) as given in Table 13-6.
3. Breaks discharge coolant to a downstream pit or the reactor bay floor, depending on the break location.
4. Reactor scram occurs following transient initiation with total time delay of 2.05 sec. See Table 13-8.
5. Primary pump stops due to the low pool level scram signal.
6. Flapper valve is modeled to open by differential pressure across the valve, that is, differential pressure between the reactor plenum and the adjacent pool. For the Beamtube No.6 break scenario, the flapper is modeled both open and stuck closed.
7. Metal water reactions is turned off for the pool inlet and reactor outlet break scenarios since the environment is almost completely air filled, and turned on for the Beamtube No.6 break scenario since the environment is almost completely steam filled.
8. The core consists of 4 fuel channels submerged in the pool. An additional hot fuel pin is imbedded in the hot fuel channel.
9. Convection heat transfer is modeled at the heat structure surfaces of the fuel rods and fuel boxes.
10. 2-D radial and axial heat conduction is modeled in the fuel and fuel box using the reflood model.
11. Radiation heat transfer is modeled from the fuel rods to the fuel box and from the fuel box to the pool wall.
12. Core power distribution provided by the MCNP model with a peaking factor of 3.0.
13. Core decay heat is modeled using ANSI/ANS 5.1-1979 – Decay Heat Power in Light Water Reactors,[13-12] and is multiplied by a factor of 1.0.

While investigating the consequences of such an accident, it was assumed that the reactor had been operating with a 25 fuel assembly core at a power level of 2.0 MW for an infinite time so that fission product concentrations had attained equilibrium. Assessments were performed for three break locations:

1. Reactor outlet: 10-inch diameter pipe break in the primary valve pit.
2. Pool inlet: 10-inch diameter pipe break in the primary valve pit.

3. Beamtube No.6: 12-inch × 12-inch square beamtube break to the reactor bay.

All break scenarios assume catastrophic failure. Pipe break scenarios are double guillotine breaks that cannot be isolated. The Beamtube No.6 break scenario assumes a complete guillotine break on the pool side and a completely unobstructed opening on the reactor bay side.

Table 13-6 – Loss of Pool Water – Initialization

Parameter	Nominal	LCO	LSSS
Reactor Power (MW)	1.8	2.0	2.0
Primary Coolant Flow Rate (gpm)	1000	900	900
Pool Temperature (°F)	105	117	117
Pool Level from Top of Core	240	204	204

Table 13-7 – Break Flow Parameters – Loss of Pool Water

Event	Break Location				Remarks
	Reactor Outlet	Pool Inlet	Beamtube No.6		
			Flapper Open	Flapper Closed	
Cross-sectional Break Area (in^2)	78.5	78.5	144	144	
Initial Break Flow Rate (lbm/sec)	505	996	1301	1301	
Core Starts to Uncover (sec)	227	101.5	95.0	95.0	
Core Fully Uncovered (sec)	229.0	113.5	N/A	N/A	Beamtube No.6 break leaves the bottom 4 inches of the fuel covered by water

Table 13-8 – Sequence of Events – Loss of Pool Water

Event	**Break Location**				**Remarks**
	Reactor Outlet	**Pool Inlet**	**Beamtube No.6**		
			Flapper Open	**Flapper Closed**	
Break Initiation	0.0	0.0	0.0	0.0	Initiating Event
Primary Pump Trip (sec)	0.0	0.0	0.0	0.0	Primary pump trips on LSSS for pool level scram
Reactor Scram signal (sec)	2.0	2.0	2.0	2.0	Reactor scram on LSSS
Scram Rods starts to drop (sec)	2.05	2.05	2.05	2.05	Scram time delay of 0.05 second
Full Insertion of Scram Rods	3.05	3.05	3.05	3.05	1.0 second rod drop time
Flapper Valve fully Open sec)	747.5	92.5	0.9	N/A	Opens on loss of differential pressure
Core starts to Uncover (sec)	227.0	101.5	95.0	95.0	77.25 inches from the pool bottom
Core fully voided (sec)	229.0	113.5	N/A	N/A	53.25 inches from the pool bottom
Time to Equilibrium Pool Level	~1200.0	~160.0	~200	~200	
Peak Fuel Temperature (°F)	3165.0 sec (1983.1 °F)	2985.0 sec (2031.7 °F)	2022.5 sec (1635.6 °F)	2502.5 sec (1849.8 °F)	Safety Limit – 4352 °F
Peak Cladding Temperature (°F)	3215.0 sec (1964.7 °F)	2995.0 sec (2011.8 °F)	2065.0 sec (1619.1 °F)	2500.0 sec (1832.5 °F)	Safety Limit – 2200 °F[a]

In the event of a LOCA, natural-convection air cooling will be sufficient to ensure that the peak fuel cladding temperature will not exceed the safety limit of 2200 °F for PULSTAR fuel, which ensures the integrity of the fuel cladding.[13-3][a] Figure 13-6 details the break flow for the modeled LOCAs, Figure 13-7 details the hot fuel pin centerline temperatures, and Figure 13-8 details the fuel cladding temperatures.

The above analysis indicates that no loss of cladding integrity or fuel damage would result from a loss of coolant accident.

[a] Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

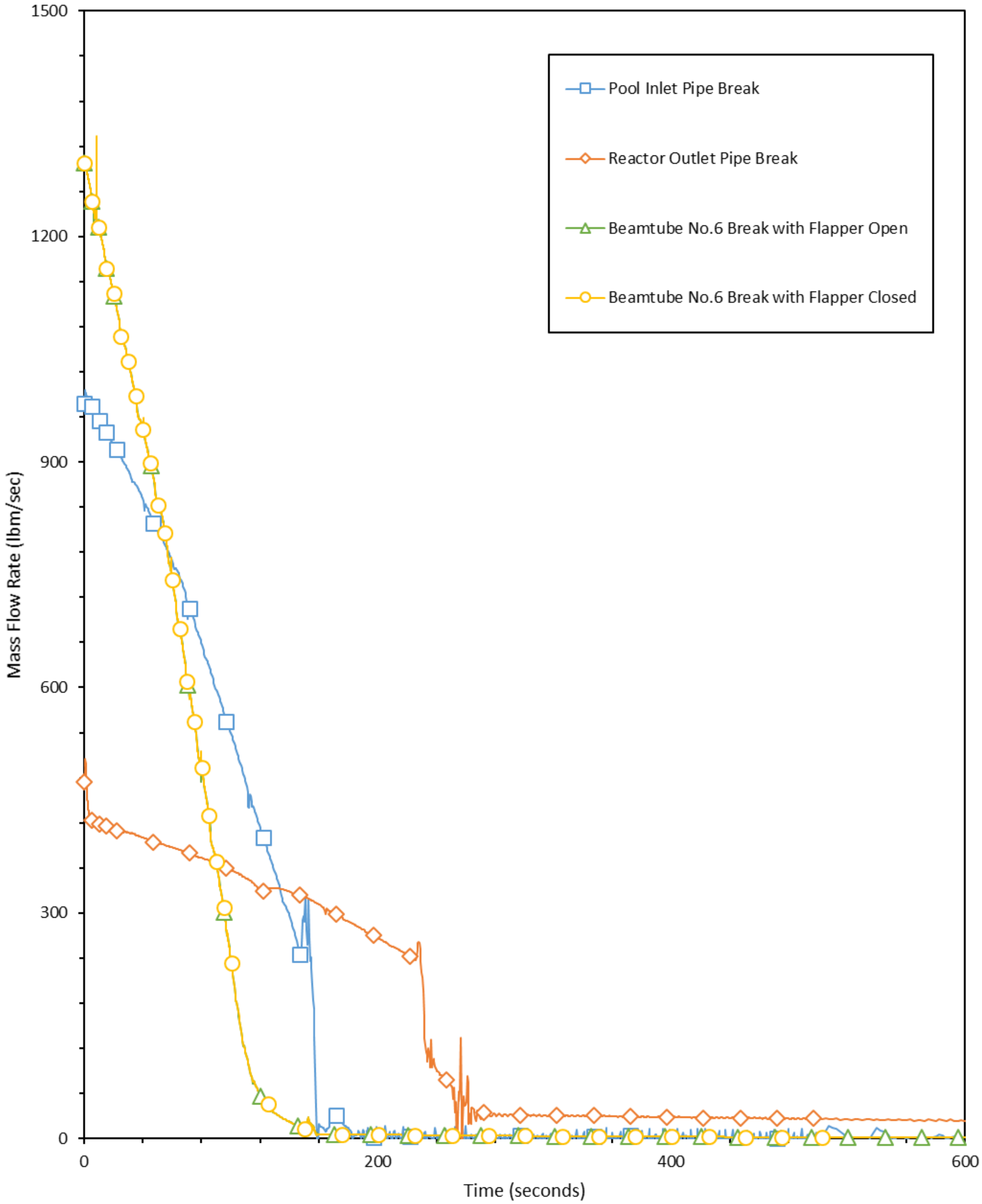

Figure 13-6 – Break flow resulting from a LOCA at the three break locations: reactor outlet piping, reactor inlet piping and Beamtube No.6.

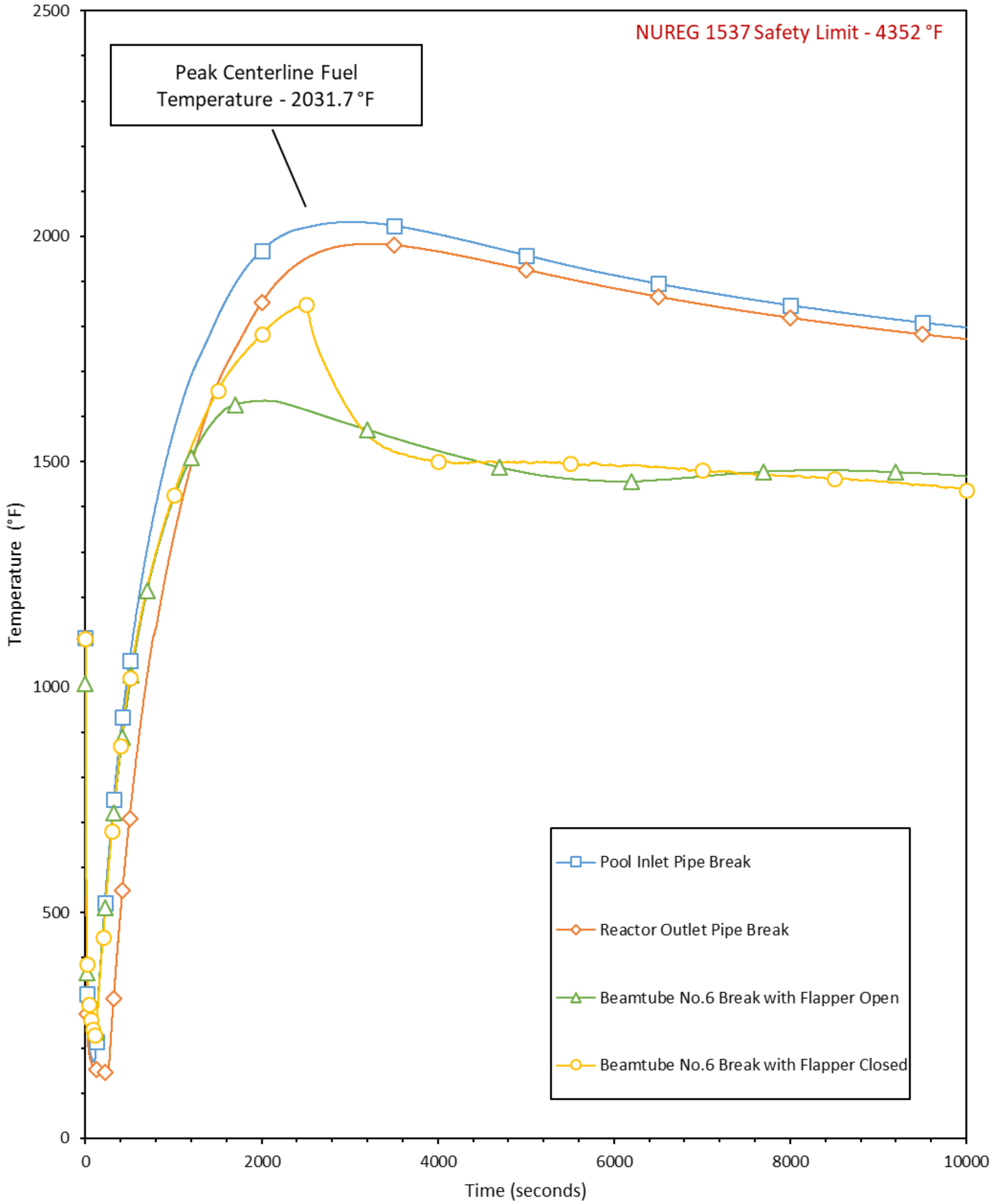

Figure 13-7 – Fuel centerline temperatures in the hot channel resulting from a LOCA at the three break locations: reactor outlet piping, reactor inlet piping and Beamtube No.6.

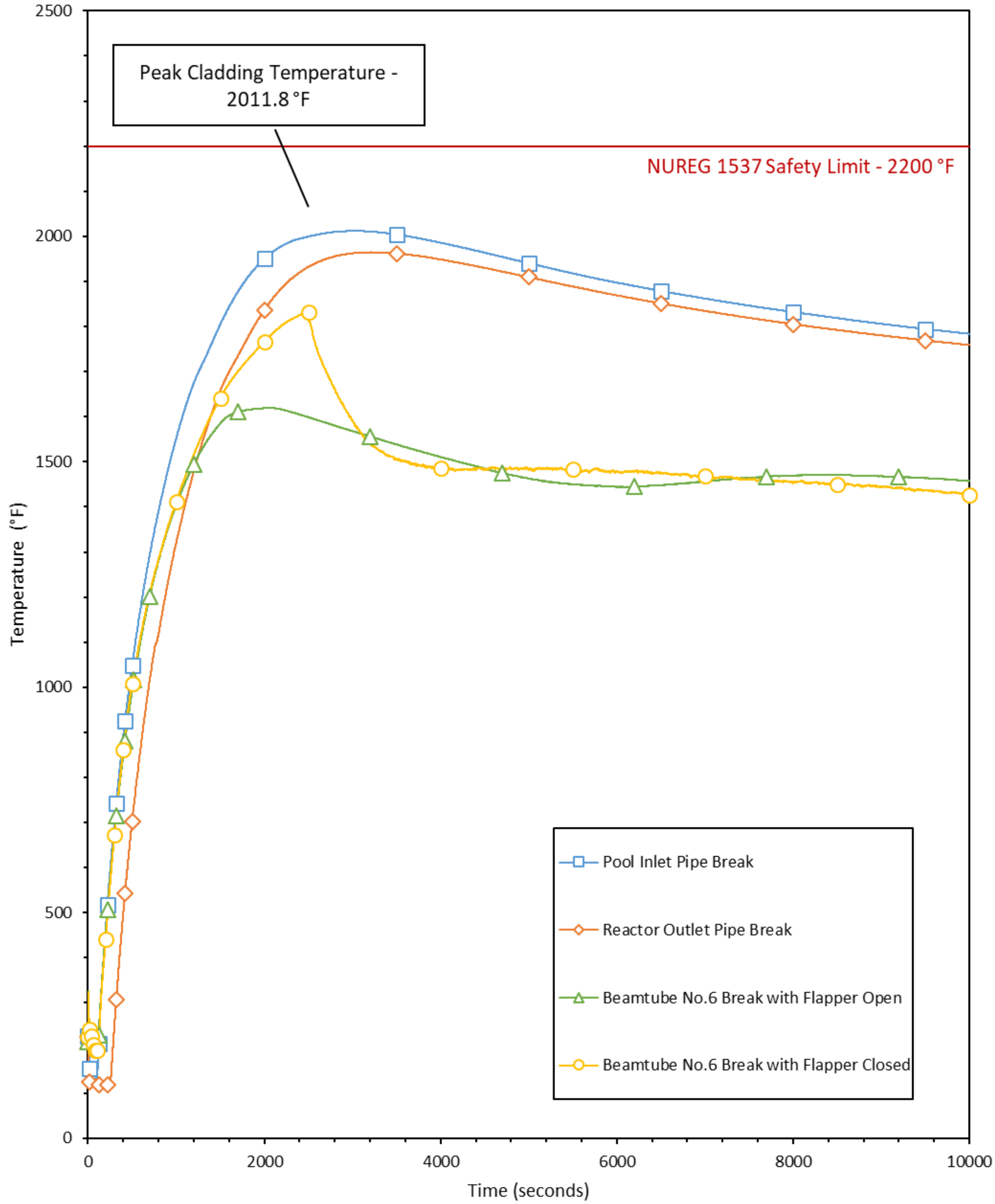

Figure 13-8 – Fuel cladding temperatures in the hot channel resulting from a LOCA at the three break locations: reactor outlet piping, reactor inlet piping and Beamtube No.6.

13.2.3.1. Hazard from Graphite

As detailed in Section 4, the NCSU PULSTAR has previously incorporated the use of up to 10 graphite reflectors to increase the excess reactivity of the core. The potential for a large release of Wigner energy and/or a graphite fire following a loss of pool water has been analyzed by Brookhaven National Laboratories for all research reactors in the United States. This study as documented in NUREG/CR-4981[13-13] concluded that there is no new evidence associated with either the Windscale Accident or the Chernobyl Accident that indicates a credible potential for a graphite burning accident in any of the US research reactors, such as the PULSTAR. Furthermore, the study indicates there is no new evidence that suggests that detailed case-by-case analyses of the role of graphite in NRC licensed reactors are warranted.

13.2.3.2. Hazard from Radiation

The possibility of a LOCA causing a complete loss of water cover over the reactor core is considered minimal due to the design of the reactor structure, primary coolant system, and experiment beamtubes. The primary coolant system has secured piping and components and several isolation valves. Experiment beamtubes are secured and sealed to prevent leakage or sudden loss of water. A heat exchanger breach, as described later, would leave the reactor core covered by water but may result in water drainage from the reactor pool leaving fuel in the linear storage racks uncovered. The radiation hazard resulting from uncovered fuel in the reactor core is more significant of the two scenarios since fuel in storage has decayed.

In the event a LOCA did cause the reactor core to become uncovered, alarms and administrative controls would clear affected areas of personnel in ample time to prevent excessive personnel exposure. The reactor pool level alarm would occur first, and then the over the pool radiation monitor alarm would occur prior to the reactor pool level draining to the reactor core level. Reactor building evacuation time is estimated to be 1 to 3 minutes after the reactor evacuation alarm is activated. The worst case would have the evacuation alarm activated by the over the pool radiation monitor alarm leaving personnel with less than 2 minutes to evacuate the reactor building until the reactor core is exposed. The drain times are given in Table 13-8. If the facility is secured (e.g. after hours), then notification of the low pool level is made to the reactor staff either directly or by the NCSU Police Department. If a LOCA were to occur while reactor staff are away from the reactor facility, it is assumed that the complete draining of the pool would occur.

The major hazard associated with a LOCA that results in the complete loss of pool water is the high radiation levels from the uncovered reactor core. The corrective measure would be to repair the leak and refill the pool without subjecting personnel to excessive radiation dose. If radiation levels and time permit, reactor pool isolation, leak repair in areas outside the biological shield, and pool filling may be initiated in response to a LOCA to maintain water shielding over the reactor core by the reactor staff and emergency personnel. Large scale refilling would be accomplished by using a local fire hydrant and fire department hoses connected to an auxiliary pool fill line. Reactor staff unblock and make ready the auxiliary pool fill line, which has a fire hose connection outside the reactor building to minimize personnel dose. The auxiliary fill line supplies water to the upper level of reactor pool below the top of the biological shield elevation. If a LOCA were to occur while reactor staff are away from the reactor facility, it is assumed that the complete draining of the pool occurs.

[redacted] fuel storage pits are located in the reactor pool [redacted] which may be used to store the [redacted] fuel assembly core. The storage pits are designed to ensure a subcritical condition with [redacted] fuel assemblies loaded in each. If the LOCA was to occur and time and radiation levels permit, fuel from the reactor core and storage racks along the reactor pool walls may be moved to the storage pits. The storage pits may be covered with additional shielding as well. The storage pit would be water filled at all times since it is [redacted] and depending on the breach location there may be water above the storage pits. The use of the storage pits would allow repairs to be made with the reactor pool partially drained.

Radiation levels for this scenario were calculated previously by Monte Carlo techniques and benchmarked using reported Livermore Pool Type Reactor (LPTR) data for the NCSU PULSTAR reactor at a power of 1 MW.[13-14,13-15] These results are doubled for 2 MW since the radiation dose rates are associated with short-lived radionuclides that have saturated to an equilibrium level. Radiation dose rates for a LOCA resulting in an uncovered reactor core are given in Table 13-9 for 10 minutes after reactor shut down.

Table 13-9 – Dose Rates at Various Locations Following a LOCA

Location	1 MW	2 MW
	rem/h	rem/h
Over pool	2.3E+02	4.6E+02
Side of pool	5.0E+01	1.0E+02
Walkway to pool	1.0E+00	2.0E+00
Control Room entry	5.0E-02	1.0E-01
Control Room door to reactor	2.5E-01	5.0E-01
Loading Dock platform	2.3E-01	4.5E-01
Reactor building floor – North	2.0E-01	4.0E-01
Reactor building floor – South	2.0E-01	4.0E-01
Reactor building floor – East	1.8E-01	3.5E-01
Reactor building floor – West	1.8E-01	3.5E-01
Reactor building outer wall	4.0E-03	8.0E-03
Roof Top – above reactor core	2.5E+01	5.0E+01
Loading Dock outer door	1.5E-01	3.0E-01
Emergency Generator	1.0E-02	2.0E-02
Outer wall of reactor building	4.0E-03	8.0E-03
Site Boundary (streets)	2.0E-03	4.0E-03
Loading Dock driveway	3.0E-03	6.0E-03

	24 hours dose rem
Site Boundary (streets)	4.8E-02

The reactor pool top and reactor building roof are the areas with major radiation dose concerns. The radiation dose rates at the reactor bay floor (elevation [redacted]) would result in high radiation areas (HRA). Personnel dose to reactor staff and emergency responders would be controlled as stated in the emergency plan. Based on the anticipated repairs and radiation fields, personnel doses would be monitored and controlled to levels less than the annual limits given in 10 CFR Part 20 for occupational workers. Dose rates outside the reactor building are predicted to be less than 0.010 rem/h, primarily from skyshine rather than direct exposure through the reactor building wall. For 24 hours without any shielding in place, the dose is expected to be less than 0.05 rem at the site boundary (surrounding streets).

Because of the high radiation levels inside the reactor building and on the reactor building roof from uncovered fuel in the reactor core, this accident is considered to be the MHA for external radiation dose.

The above analysis indicates that a loss of coolant accident and subsequent uncovering of the core would not result in dose limits to personnel or members of the public being exceeded.

13.2.4. Heat Exchanger Pressure Boundary Breach

In this scenario, primary coolant leaks into the secondary coolant since the reactor pool water elevation is higher than that of the secondary cooling tower water basin which creates a pressure gradient. This gradient provides the driving force in a pressure boundary breach for a primary side to secondary side water leak.

The only path resulting in a primary to secondary leak from the heat exchanger would be if a hole were to develop in a plate. During operation of the secondary pump, the secondary side pressure exceeds that of the primary side pressure. A heat exchanger pressure boundary breach while the secondary pump is operating would result in a mass transfer from the secondary side to the primary side. This type of leak is easily identified since the pool water level would start to increase and secondary water contaminants would be activated and collected in the primary water demineralizer system causing an abnormal radiation level increase. If the leak happened during low power operations, or when the secondary pump was secured, then primary water would transfer into the secondary system. Detection of this type of heat exchanger boundary breach would be from a decrease in the pool water level. If an entire plate fails, the leak rate would be approximately 16 gpm (1000 gpm / 63 plates). The design of the facility prevents the core from becoming uncovered due to a heat exchanger pressure boundary breach. This is due to the cooling tower basin elevation being approximately [redacted] higher than the reactor core.

A heat exchanger leak is unlikely due to the construction and material of the heat exchanger, the proper maintenance of primary and secondary water chemistry, and the reactor surveillance program. Only in the unlikely event of a failure in a plate could water transfer between the secondary and primary systems. If a boundary breach were to suddenly appear during power operation, secondary water would enter the primary system, due to the secondary system being at a slightly greater pressure than the primary.

The surveillance program includes monitoring of heat exchanger efficiency for reduced heat transfer caused by fouling and radioactivity analysis of the secondary coolant. Primary water is kept at a high quality minimizing the potential for corrosion. The secondary coolant system has a corrosion control system to minimize corrosion. Filters and screens are also used in the secondary coolant system to minimize fouling of the heat exchanger.

The pressures resulting from dead-heading either the primary or secondary pump by the closing of a valve downstream of the heat exchanger are well below the design pressure limit of the heat exchanger, therefore the pressure boundary of the heat exchanger would not be compromised.

As a result of a worst case heat exchanger leak, the reactor pool water level will decrease at a maximum rate of 16 gpm or 19 inches per hour. The abnormal (low) pool water level alarm occurs at -12 inches and the low pool water level scram and primary pump trip setpoint occurs at -36 inches. The estimated time between the first abnormal pool level indication and the second is 75 minutes. A radiation alarm from the over the pool radiation monitor may also occur after the reactor pool level reaches -36 inches. Reactor staff should be able to respond within the 75 minutes to secure the leak resulting in a maximum primary coolant loss of 2200 gallons. Maximum time to drain the reactor pool to the cooling tower water basin elevation ranges from 4 hours 20 minutes to 5 hours 10 minutes. The maximum primary coolant loss would be approximately 6300 gallons.

It can be concluded that a heat exchanger primary pressure boundary breach will not result in the uncovering of the core. The concern would be potentially increased external radiation levels inside the reactor building and a release of radioactive materials to the unrestricted area. Reactor fuel in storage in the pool racks may be uncovered if the reactor staff does not respond in a timely manner or is not able to secure the leak and is not able to relocate the fuel from the storage racks to the storage pits.

Radiation dose rates for this scenario are significantly lower than those for a complete reactor pool drain. Radiation dose from the release of radioactive materials for this accident are negligible, as described later in this section.

13.2.5. Loss of Coolant Flow

13.2.5.1. Blocked Flow Accident

One type of loss of flow accident which has resulted in partial meltdown of MTR plate type fueled cores has been due to blockage of the fuel assembly flow by foreign objects such as gaskets, plastic sheets, etc. This type of condition may not cause a scram and can go undetected.

The PULSTAR fuel assembly has an engineered safety feature in its design to mitigate the blocked flow type of accident. [redacted] Since all 25 fuel pins in a fuel assembly have a common flow channel, the four holes will provide a path for the coolant in the unlikely event that a foreign object (e.g., gaskets, plastic sheets, etc.) falls over the upper coolant inlet of the fuel assembly.

When local flow blockage at the top of the core inlet happens, main coolant flow to the blocked fuel channel is redirected from the fuel channel inlet at top to the side bypass flow holes. Since there is no monitoring that can detect flow blockage, proper operator actions may not take place and the reactor may erroneously continue operating at full power. Therefore, it is important to show that core safety parameters are not challenged in the case of core flow blockage.

To demonstrate core safety, a complete blockage of the hot fuel assembly at the top is conservatively assumed for analysis. The reactor is at full power operation at the limiting conditions and neither reactor scram nor operator actions are credited. The RELAP simulation was run until a stable operating condition was achieved provided by the redirected flow through the side flow holes.

Table 13-10 summarizes the flow distribution and temperatures in the hot fuel channel in normal and blocked conditions. Hot channel exit flow decreases from 4.95 lbm/sec to a stable 4.13 lbm/sec for the blocked condition, where there is zero flow from the channel top and main flow is redirected through the side bypass flow holes. Flow through the side bypass flow holes increases from 2.08

lbm/sec to 4.13 lbm/sec, which becomes the sole contributor to fuel cooling in the blocked channel. Peak fuel centerline temperature increases slightly from 1110.8 °F to 1124.2 °F, which implies that the flow through the side bypass flow holes provides sufficient fuel cooling. From the analysis, it can be concluded that core safety is ensured by the flow through the side bypass flow holes in the event of a complete blockage of a fuel assembly.

Table 13-10 – Hot Channel Parameters – Blocked Flow Accident

Parameters	Unblocked Fuel Assembly	Blocked Fuel Assembly
Hot Channel Inlet Flow (lbm/sec)	2.87	0.0
Hot Channel Flow Hole Bypass Flow (lbm/sec)	2.08	4.13
Hot Channel Exit Flow (lbm/sec)	4.95	4.13
Hot Channel Exit Coolant Temperature (°F)	137.5	141.5
Peak Fuel Centerline Temperature (°F)	1110.8	1124.2
Peak Fuel Cladding Temperature (°F)	226.9	239.9

13.2.5.2. Loss of Forced Flow Accident

This accident is one of the most limiting for forced convection cooled non-power reactors, where the forced flow is downward through the reactor core. Upon loss of forced downward coolant flow through the core, coolant flow in the core must reverse to establish upward natural convection cooling. During the flow reversal, heat transfer may be inadequate in the core which may challenge important safety parameters and ultimately the integrity of the fuel cladding. Some initiators of loss of coolant flow could be loss of electrical power or failure of the primary pump or other component in the primary coolant system.

The postulated initiating event at the PULSTAR is an electrical power failure resulting in a total loss of forced flow. The scenario assumes that the reactor has been operating at full power and fission product decay rates have reached equilibrium, therefore decay heat following scram will be at the maximum.

This accident was analyzed using the RELAP5/MOD3.3 code of the US Nuclear Regulatory Commission (NRC).[13-11] The report Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor[13-4] describes the RELAP5 model of the PULSTAR systems, initializations, and results of the accident analyses in greater detail. Key modeling features are summarized below and transient analysis is carried out until stable natural convection cooling is achieved:

1. RELAP5 nodalization of the PULSTAR system is given in the report Safety Analysis for Assessing 2 MW Power Upgrade for the NCSU PULSTAR Reactor.[13-4]

2. The transient is initiated by the complete stop of the primary coolant pump from full power limiting safety system settings (LSSS) as given in Table 13-11.
3. Coast down of the pump is not assumed.
4. Reactor scram occurs with transient initiation with total time delay of 2.05 sec
5. Flapper valve is modeled to open by differential pressure across the valve, that is, differential pressure between the reactor plenum and the adjacent pool, or to remain closed depending on the scenario.
6. Flow communication occurs between the fuel channels and the pool through the fuel channel top opening and four flow holes at upper part of a fuel channel.
7. Convection heat transfer is modeled from the fuel to the fuel channels to the fuel box then to the pool.
8. Core power distribution provided by the MCNP model with a power peaking factor of 3.0.
9. Core decay heat is modeled using ANSI/ANS 5.1-1979 – Decay Heat Power in Light Water Reactors,[13-12] multiplied by 1.0.

In the event that the primary flow is interrupted by loss of the pump or other causes, a flapper valve on the plenum, which is located directly under the grid plate, will open and provide a path for natural convection cooling to be established within the reactor pool. The flapper valve is held in a closed position by the differential pressure created by the coolant flowing through the core and the pool static head at the plenum level. The normal operating position of the flapper valve is closed during forced convection flow cooling. Refer to Section 6.2.4 for more details on the flapper system.

The reactor is assumed to scram when a limiting safety system setting (LSSS) signal is received by the reactor safety system (RSS). It is assumed that there is a 2.0 second sensor delay and a 0.05 second scram delay processing time for a total delay of 2.05 seconds. When the pump stops, a conservative assumption is that forced flow stops instantly. The initial conditions are given in Table 13-11 and are the technical specification limits for operation at 2.0 MW. The event sequence is summarized in Table 13-12.

Table 13-11 – Initialization – Loss of Flow Accident

Parameter	Nominal	LCO	LSSS
Reactor Power (MW)	1.8	2.0	2.0
Primary Coolant Flow Rate (gpm)	1000	900	900
Pool Temperature (°F)	105	117	117
Pool Level from Top of Core (inches)	240	204	204

Table 13-12 – Sequence of Events – Loss of Flow Accident

Event	Flapper Valve Open (seconds)	Flapper Valve Closed (seconds)	Remarks
Primary Pump Stops	0.0	0.0	Initiating event
Maximum Hot Pin Temperature	0.0 (1110.8 °F)	0.0 (1110.8 °F)	
Flapper Valve Opens	1.5	NA	Due to loss of differential pressure
Scram Signal	2.0	2.0	Low flow signal + 2.0 second delay
Control Rods Start to Drop	2.05	2.05	0.05 second scram circuit time
Rods Reach Bottom	3.05	3.05	1.0 second rod drop time
Onset of Core Flow Reversal	3.5	5.0	
Hot Channel Boiling	4.5	5.5	
Maximum Cladding Temperature	4.5 (323.6 °F)	5.5 (329.6 °F)	
Time to Stable Natural Convection	~100	NA	

Figure 13-9 shows the transient behavior of reactor power. During the first two seconds of the transient, the reactor power initially decreases as the result of negative reactive feedback caused by the reduction in moderator density as the coolant heats up prior to the reactor scram. The reactor scrams when the LSSS setpoint for low coolant flow rate is reached (900 gpm).

Figure 13-10 shows the transient behavior of the natural convection flow rates. For the flapper open scenario, natural convection is fully established through the flapper valve and primary loop within 30 seconds from the start of the transient and ensures long term stable cooling of the core with no excessive heat up of the fuel. Negative flow rates signifies upward flow.

Figure 13-11 shows the transient behavior of the counter-current flow between fuel channels for the flapper closed scenario. Counter-current flow is fully established through the core within 30 seconds from the start of the transient even when there is no flow through the flapper valve.

Figure 13-12 and Figure 13-13 shows the transient behavior of the hot spot fuel and cladding temperatures. Fuel pin temperature during the transient does not exceed the maximum fuel pin temperature of 1110.8 °F seen during steady-state operation at limiting conditions for both the flapper open and flapper closed scenarios. Maximum cladding temperature does increase to 323.6 °F and 329.6 °F for the flapper open and closed conditions, respectively, however, all temperatures are well below established safety limits.

The above analysis indicates that no loss of cladding integrity or fuel damage would result from a loss of flow accident.

Table 13-13 – Summary – Loss of Flow Accidents

Parameter	Flapper Open	Flapper Closed	Remarks
Peak Fuel Temperature (°F)	1110.8 °F	1110.8 °F	Safety Limit – 4352 °F
Peak Cladding Temperature (°F)	323.6 °F	329.6 °F	Safety Limit – 2200 °F[a]

[a] Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

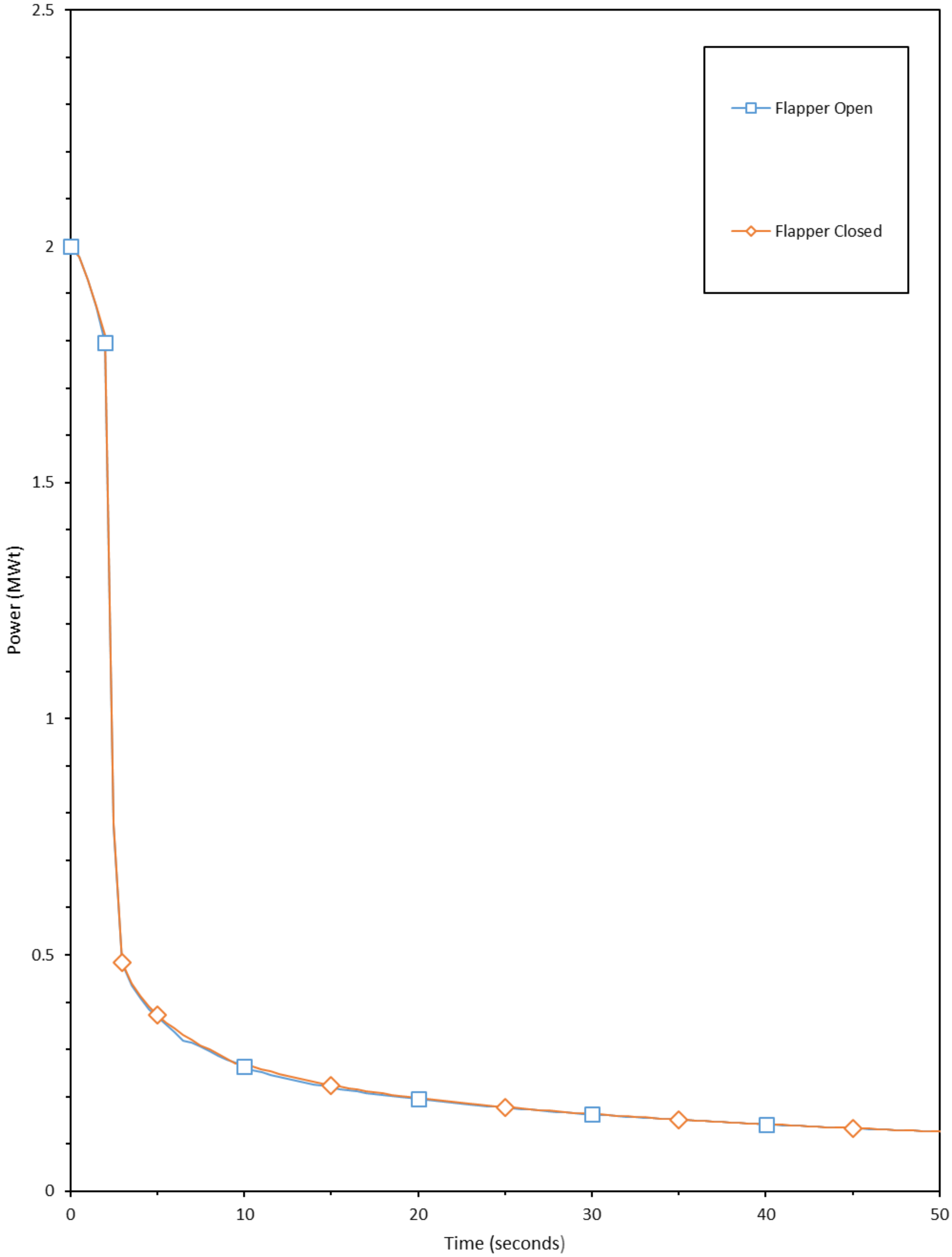

Figure 13-9 – Reactor power level resulting from a loss of flow accident for an initial condition of hot full power with forced flow cooling.

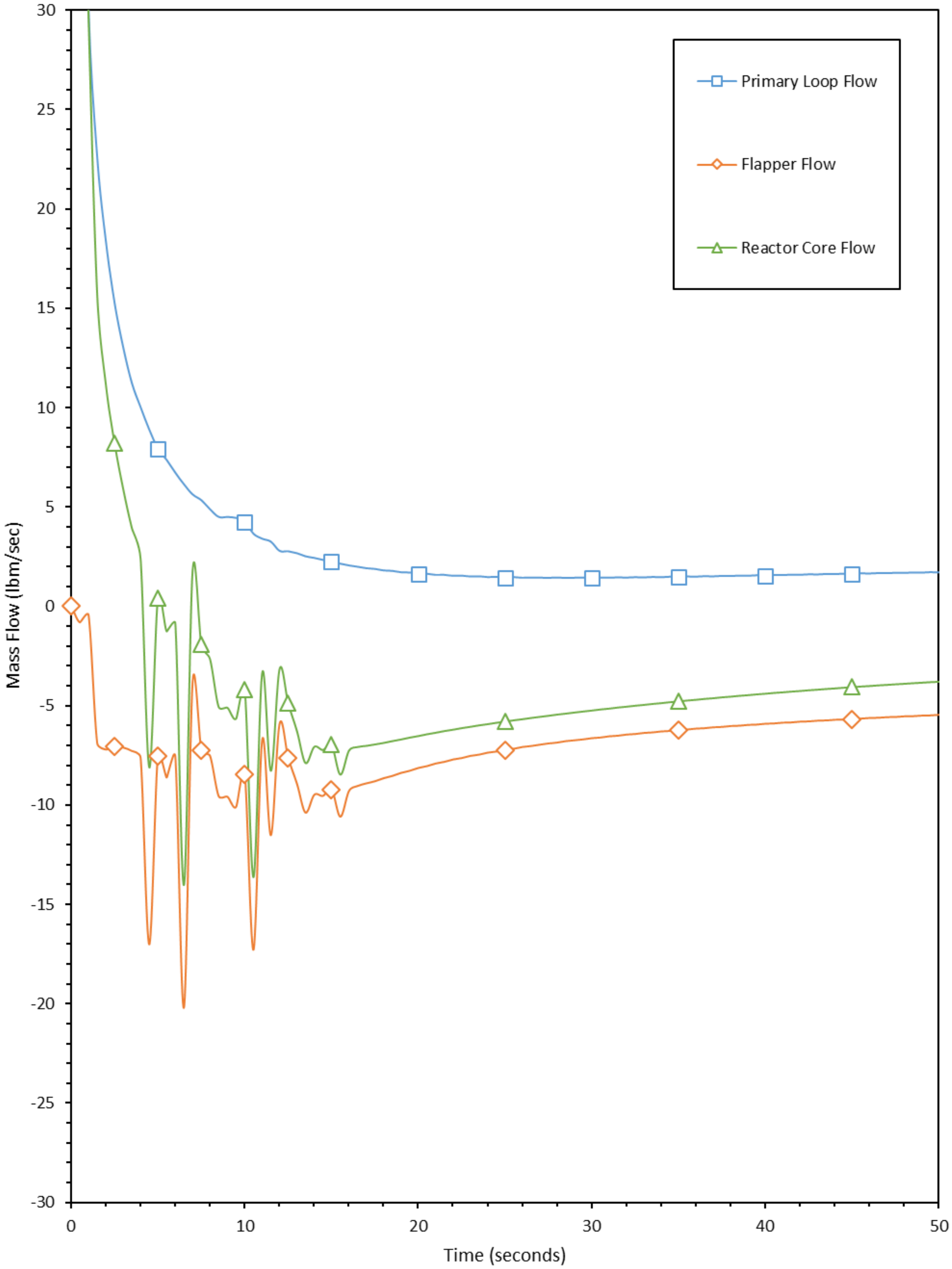

Figure 13-10 – Natural convection flow rate resulting from a loss of flow accident for an initial condition of hot full power with forced flow cooling and the flapper open.

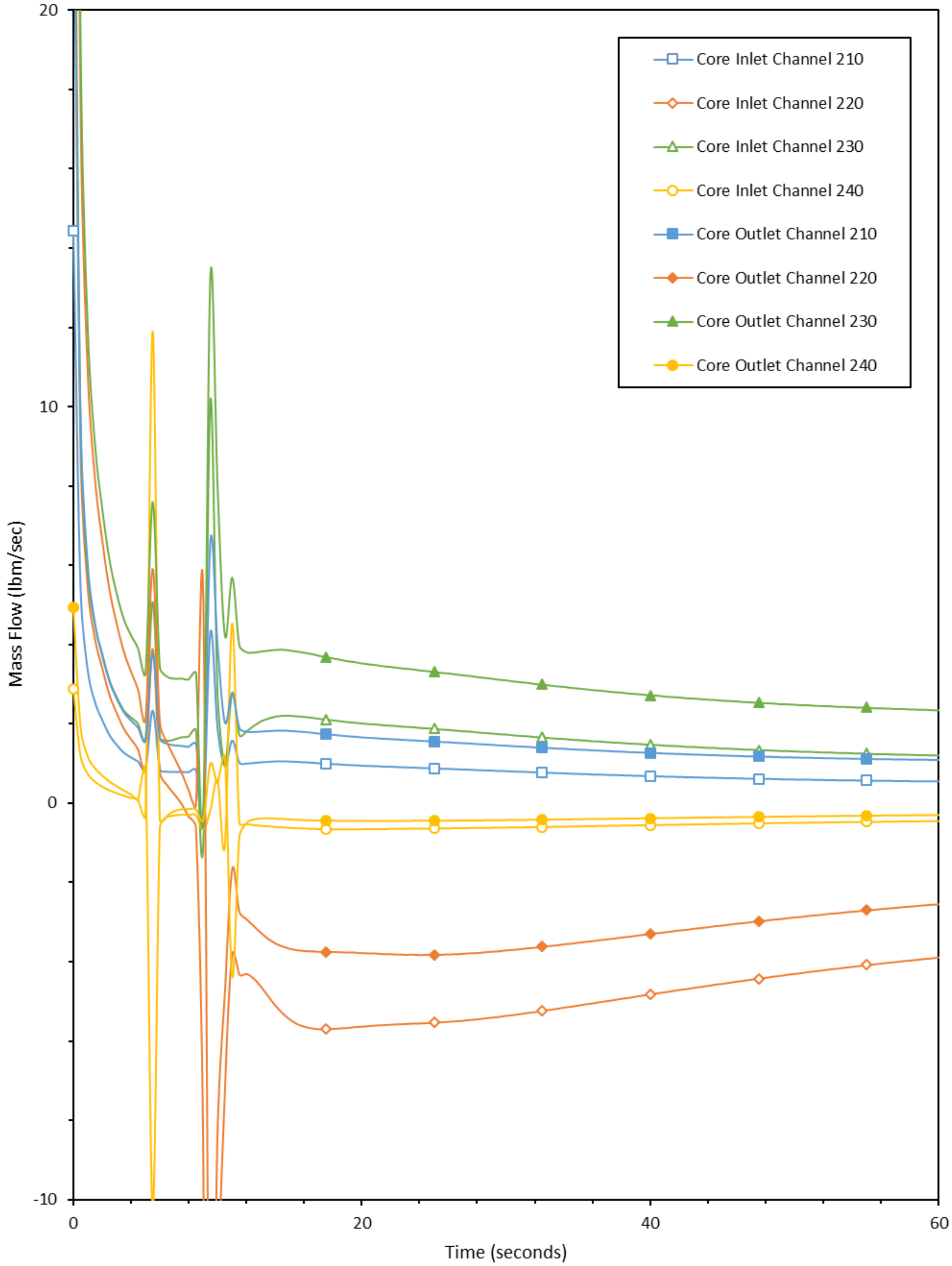

Figure 13-11 – Fuel channel flow rate resulting from a loss of flow accident for an initial condition of hot full power with forced flow cooling and the flapper closed.

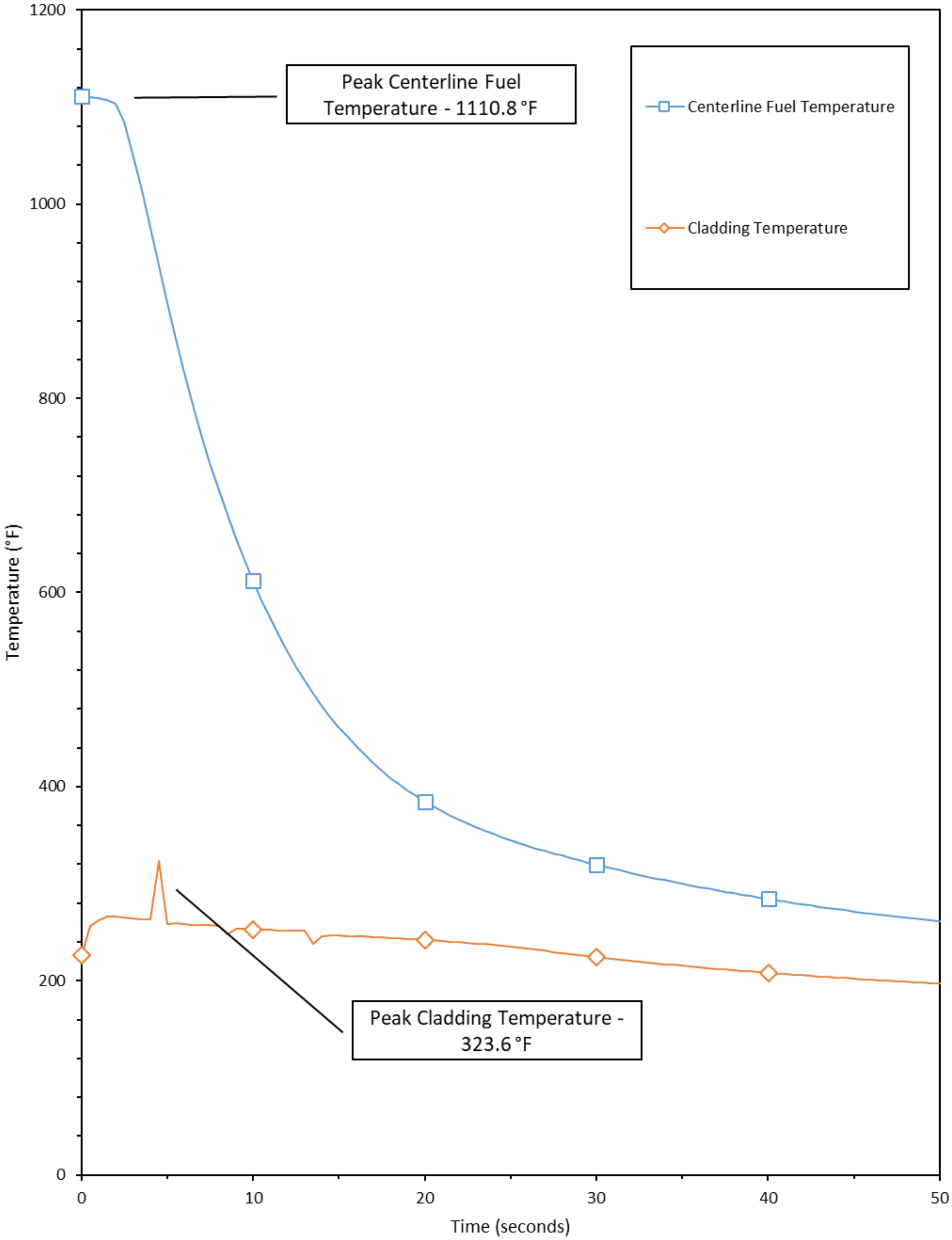

Figure 13-12 – Maximum temperatures in the hot channel resulting from a loss of flow accident for an initial condition of hot full power with forced flow cooling and the flapper open.

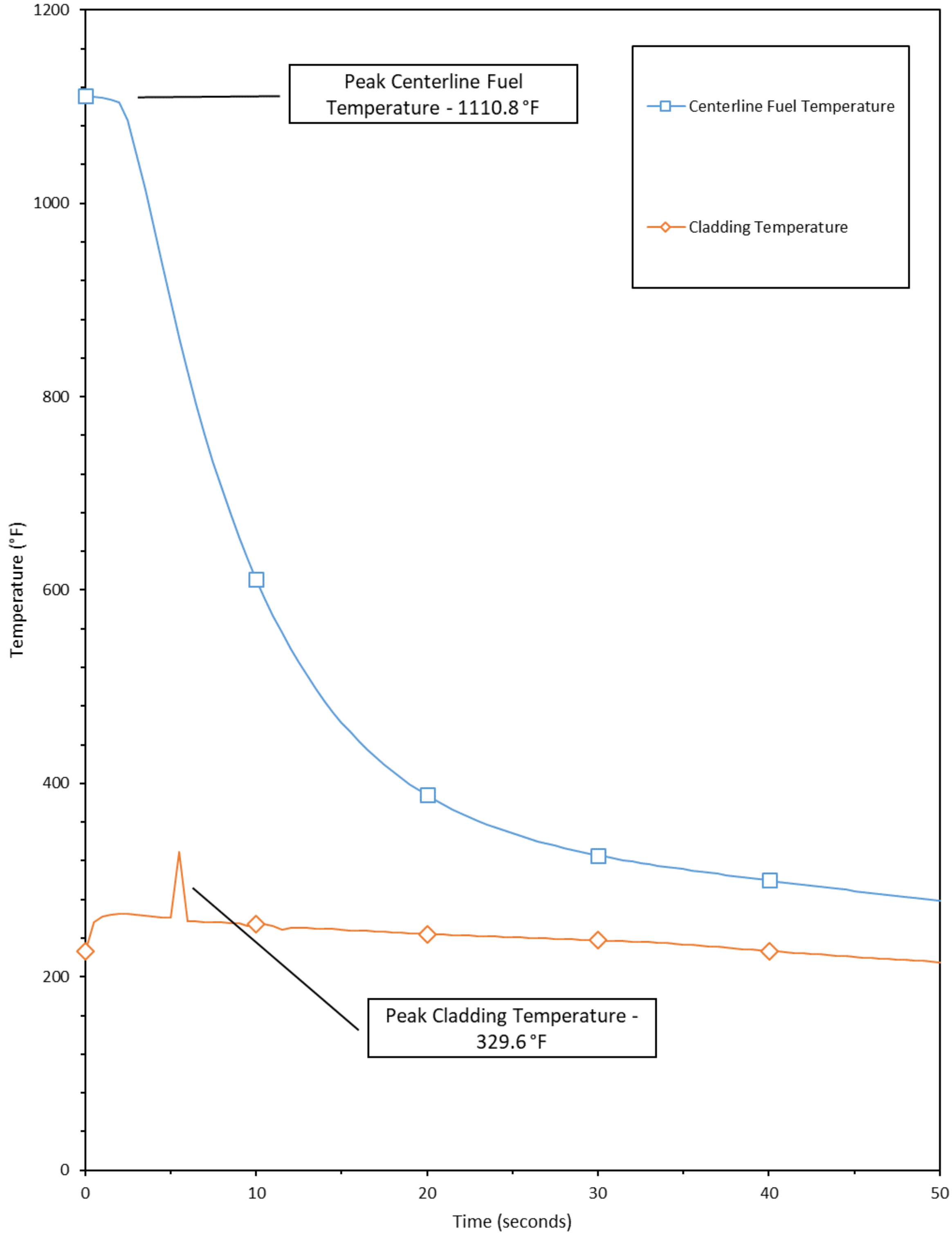

Figure 13-13 – Maximum temperatures in the hot channel resulting from a loss of flow accident for an initial condition of hot full power with forced flow cooling and the flapper closed.

13.2.6. Mishandling or Malfunction of Fuel

Events which could cause accidents of this type at the NCSU PULSTAR include:

1. Fuel handling accidents where an element is damaged severely enough to breach the cladding.
2. Failure of the fuel cladding due to manufacturing defect or corrosion.

13.2.6.1. Cladding Failure Accident

In the cladding failure accident (CFA), submersion and internal dose to occupants inside and members of the public outside the reactor building are of concern. Credit is taken for the retention of many fission products and other radionuclides present in the primary coolant system. For the CFA, the fission product inventory in the NCSU PULSTAR reactor core is based on 6320 MW·days of operation at 2 MW (equivalent to the fuel burnup limit of 20,000 MWd/MTU).

The activities of the fission products which would be found in the fuel pin annulus gap between the zircaloy-2 cladding and the UO_2 pellets are estimated using the data for commercial reactor fuel given in NUREG 1887[13-16] and NUREG/CR2507[13-17] by adjusting for power level and fuel temperature.

NUREG 1887 provides specific core isotopic inventory for commercial reactor fuel in Bq/MW. Additional radionuclide inventory data given in IAEA Publication 53[13-18] for research reactors were also included in the CFA analysis for the NCSU PULSTAR reactor fuel.

The activity in the fuel gap is dependent on the type of fuel, fuel dimensions, fuel burnup, linear power generation, radionuclide half-life, and centerline fuel temperature. PULSTAR fuel and commercial power reactor fuel have similar dimensions, cladding, chemical form, and ^{235}U enrichment. The major difference is the fuel temperature. Low temperature data on low enriched UO_2 sintered pellets of low enrichment is provided in NUREG/CR2507. NUREG/CR2507 gives data for low temperature fuel and burnup similar to that for NCSU PULSTAR fuel. For this analysis, the gap release fraction is based on data and equation 14 given in Section III of NUREG/CR2507 for low temperature UO_2 commercial fuel (density of 10.0 to 10.6 grams per cm^3 with a surface area to volume value of 6 cm^{-1} and for a fission energy release of 200 MeV per fission):

$$Gap\ Release\ Fraction = 1 \times 10^{-7} \cdot \lambda^{-0.5} + 1.6 \times 10^{-12} \cdot \left({}^{P}/_{\lambda}\right) \qquad \textbf{Equation 13-5}$$

where,

P is ${}^{2\ MW}/_{0.316\ MTU} = 6.25\ MW/MTU$ for the NCSU PULSTAR

λ is the radioactive decay constant in s^{-1}

For the CFA, it is assumed that 25 fuel pins at maximum burnup suffer a clad rupture and all the fission products contained in the annulus of the 25 fuel pins are released into the reactor pool with a minimum water depth of 14 feet. The fuel pins are well protected due to the design and construction of the fuel assembly, as described in Section 4. There is no credible scenario that could result in the failure of all 25 fuel pins, nevertheless, the analysis assumes that the damage occurs due to a handling error resulting in mechanical shock to the fuel pins.

All of the fission gases and 3 percent of the iodine and bromine isotopes are assumed to be released from the primary coolant and into the reactor bay air volume at the surface of the reactor pool. Short-lived radionuclides from the decay of airborne fission products are assumed to reach equilibrium values in the reactor air volume. Particulate fission products released to the reactor pool are retained in the reactor pool.

For the purpose of assessing the consequence of released fission products, it is assumed that all of the radioactive fission gases in the fuel gap for 25 fuel pins and 3 percent of the halogens activity in the fuel gap from 25 fuel pins escape into the reactor building air volume. The released activity is uniformly mixed in the free air volume over the next 24 hours given that the air exchange rate in confinement is 2.35 hours. Assuming immediate release of the gap inventory as described accounts for all the radioactivity that could be released. Continuing releases of activity into the reactor building is not considered to occur since the fuel temperature is low, which limits diffusion from the fuel, and the reactor pool evaporation rate of approximately 1 gallon per hour is negligible relative to the reactor pool volume of approximately 14,000 gallons.

The anticipated sequence of events which would occur for a postulated 25 fuel pin failure is as follows:

1. Short-lived fission products are assumed to decay to negligible levels prior to reaching the fuel pin gap due to the transit time through the fuel pellet. Even though fuel movements occur after a decay period following reactor operation, no credit is taken for this decay prior to beginning fuel movement as this time is variable. The remaining fission products in the fuel pin gap escape into the reactor primary coolant from the cladding failure.
2. 100 percent of the fission gases and 3 percent of the iodine and bromine isotopes are assumed to be released from the primary coolant to the reactor building air volume at the reactor pool surface. Solubility reduces the iodine and bromine fission products available for escape into the reactor building air volume. Particulate fission products released to the reactor pool are retained in the reactor pool. Short-lived fission products from the decay of fission gases are assumed to reach equilibrium values in the reactor air volume.
3. The radiation level increase in the stack gas or stack exhaust radiation monitors would cause the normal ventilation system to automatically place the reactor building into confinement and thus prevent release of unfiltered air. The Radiation Monitoring System alarm automatically initiates evacuation of the Reactor Building as well.
4. In confinement mode the normal ventilation fan is shut down and a 600 cfm confinement fan automatically starts and passes the reactor building air through a filtration system prior to its discharge from the 100-foot stack. The filtration system has a high efficiency particulate air (HEPA) filter with a removal efficiency of 99.97% for particulates and an activated charcoal filter with a removal efficiency of 99% for halogens. Prior to use, and as a routine surveillance after installation, the HEPA and charcoal filters are tested for leakage. Retention factors of 99.97 percent for particulates and 90 percent for halogens are used in the CFA analysis.
5. After passing through the confinement filters, and prior to discharge from the stack, 600 cfm flow would normally be diluted again with exhaust flow from the ventilation system serving the south wing of the Burlington Engineering Laboratories; however, no allowance for this additional dilution is assumed in this analysis.
6. The reactor building free air volume is 2.4×10^{9} cm^3 as indicated in Section 11. The fission product gases escaping from the pool are assumed to be uniformly distributed throughout the reactor building. Activity passes through the confinement filter system

and is exhausted out the reactor stack at the rate of 600 cfm.

7. The fission product release scenario is calculated to last approximately 24 hours with allowance for decay. Various offsite locations downwind of the stack would be exposed to the plume during this time. Therefore, an average release concentration for the entire 24 hours is used with no allowance for the R-63 ventilation dilution. Since the stack exhaust exit is elevated and not accessible by the public, a dispersion factor for various offsite locations is calculated to quantify radiation dose to the public associated with the 25 fuel pin failure scenario.

The assumption of mixing in the reactor building air volume is conservative since no dilution is assumed for the 600 cfm of fresh air which must leak into the reactor building as makeup. Based on the reactor building volume of 2.4×10^9 cm^3 and a purge rate of 600 cfm, this analysis makes the additional assumption that the entire fission product inventory is removed from the reactor building in 24 hours. 24 hours corresponds to approximately 10 complete air volume exchange of the free air volume. An average concentration for 24 hours was used in the dose calculations. Average concentration for 2 hours is also used as an exposure time as 2 hours allows for evacuation of nearby public areas. The average concentration is given by the following:

$$\int_0^t C(0)e^{-kt}dt = C(0)\frac{[1-e^{-kt}]}{kt} = C(ave) \qquad \textit{Equation 13-6}$$

where,

k is the removal rate constant given by the sum of the air exchange rate in confinement and the radioactive decay constant

t is time, or 24 hours for this analysis

$A(0)$ is the initial activity released from the reactor pool

V is the free air volume of 2.4×10^9 cm^3

$C(0)$ is the initial concentration in the reactor building $= {A(0)}/{V}$

$C(ave)$ is the average concentration over time t

Table 13-14 provides results from the above discussion for CFA. The following notes apply to Table 13-14:

a. Linearly scaled to 20,000 MWd/MTU from 30,000 MWd/MTU as stated in NUREG 1887[13-16] .
b. Scaled to 2 MW from previous FSAR value at 1 MW.
c. Assumed to be in equilibrium with parent nuclide.
d. Taken from IAEA Publication 1308[13-18] and adjusted for 2 MW.
e. Gap release fraction calculated using Equation 13-5.
f. Average concentration calculated using Equation 13-6. Released average concentrations were adjusted for confinement filter retention.

Table 13-14 – Reactor Core Inventory, Gap Inventory and Airborne Concentrations for a Cladding Failure Accident at 2 MW

Nuclide	Core Ci/MW	2 MW Core Ci	2 MW Design FHA Gap Ci	2 MW Gap Inventory Fraction[e]	Reactor Building *C(0)* µCi/ml	Reactor Bldg. 24h *C(ave)* µCi/ml	Stack 24h *C(ave)* µCi/ml	Reactor Bldg. 2h *C(ave)* µCi/ml	Stack 2h *C(ave)* µCi/ml
Kr83m[d]	4.49E+03	8.98E+03	3.57E-03	9.93E-06	1.49E-06	1.48E-06	1.48E-06	1.49E-06	1.49E-06
Kr85[a]	3.17E+02	6.34E+02	1.81E-01	7.15E-03	7.55E-05	7.54E-05	7.54E-05	7.55E-05	7.55E-05
Kr85m	8.00E+03	1.60E+04	9.91E-03	1.55E-05	4.13E-06	4.12E-06	4.12E-06	4.13E-06	4.13E-06
Kr87	1.60E+04	3.20E+04	1.05E-02	8.18E-06	4.36E-06	4.35E-06	4.35E-06	4.36E-06	4.36E-06
Kr88	2.30E+04	4.60E+04	2.26E-02	1.23E-05	9.42E-06	9.40E-06	9.40E-06	9.42E-06	9.42E-06
Kr89[d]	3.78E+04	7.56E+04	5.00E-03	1.65E-06	2.08E-06	1.99E-06	1.99E-06	2.08E-06	2.08E-06
Xe131m	3.30E+02	6.60E+02	3.61E-03	1.37E-04	1.50E-06	1.50E-06	1.50E-06	1.50E-06	1.50E-06
Xe133	5.70E+04	1.14E+05	3.99E-01	8.75E-05	1.66E-04	1.66E-04	1.66E-04	1.66E-04	1.66E-04
Xe133m	2.00E+03	4.00E+03	8.81E-03	5.50E-05	3.67E-06	3.66E-06	3.66E-06	3.67E-06	3.67E-06
Xe135m[d]	9.22E+03	1.84E+04	2.69E-03	3.65E-06	1.12E-06	1.11E-06	1.11E-06	1.12E-06	1.12E-06
Xe135	1.10E+04	2.20E+04	1.95E-02	2.22E-05	8.14E-06	8.12E-06	8.12E-06	8.14E-06	8.14E-06
Xe137[d]	5.14E+04	1.03E+05	7.49E-03	1.82E-06	3.12E-06	3.01E-06	3.01E-06	3.11E-06	3.11E-06
Xe138	5.70E+04	1.14E+05	1.60E-02	3.51E-06	6.66E-06	6.59E-06	6.59E-06	6.66E-06	6.66E-06
I131	2.80E+04	5.60E+04	2.47E-01	1.10E-04	3.08E-06	3.08E-06	3.08E-07	3.08E-06	3.08E-07
I132	4.00E+04	8.00E+04	3.53E-02	1.10E-05	4.41E-07	4.40E-07	4.40E-08	4.41E-07	4.41E-08
I133	5.70E+04	1.14E+05	1.55E-01	3.40E-05	1.94E-06	1.93E-06	1.93E-07	1.94E-06	1.94E-07
I134	6.30E+04	1.26E+05	3.42E-02	6.79E-06	4.28E-07	4.26E-07	4.26E-08	4.27E-07	4.27E-08
I135	5.00E+04	1.00E+05	7.53E-02	1.88E-05	9.41E-07	9.39E-07	9.39E-08	9.41E-07	9.41E-08
Br83[b]	4.49E+03	8.98E+03	4.06E-03	1.13E-05	5.07E-08	5.06E-08	5.06E-09	5.07E-08	5.07E-09
Br84m[d]	1.40E+02	2.80E+02	2.56E-05	2.28E-06	3.20E-10	3.12E-10	3.12E-11	3.19E-10	3.19E-11
Br84[b]	8.25E+03	1.65E+04	3.48E-03	5.28E-06	4.35E-08	4.33E-08	4.33E-09	4.35E-08	4.35E-09
Br85[d]	1.07E+04	2.14E+04	1.38E-03	1.61E-06	1.73E-08	1.64E-08	1.64E-09	1.72E-08	1.72E-09
Cs138[b,c]			1.60E-02		6.66E-06	6.62E-06	1.99E-09	6.66E-06	2.00E-09
Rb89[c,d]			5.00E-03		2.08E-06	2.06E-06	6.19E-10	2.08E-06	6.25E-10
Rb88 [c]			2.26E-02		9.42E-06	9.34E-06	2.80E-09	9.42E-06	2.83E-09

Table 13-15 lists the effective dose-equivalent and thyroid dose-equivalent to occupational personnel during evacuation and for 24 hours of response with a APR respirator. Effective dose is the sum of the deep dose-equivalent and committed effective dose-equivalent. Thyroid dose is the sum of the deep dose-equivalent and committed dose-equivalent. The 2 hour average concentrations were used for the 2 hour doses for response with a SCBA respirator.

The following notes apply to Table 13-15:

a. Correction of 0.1 for finite room size for noble gas as stated in Section 11.

b. Protection factor of 100 for a negative pressure air purifying respirator as stated in 10 CFR Part 20 Appendix A.

c. Dose conversion factors were used from 10 CFR Part 20 Appendix B Table 2 to convert concentration of airborne activity to dose. Corrections for exposure time were made.

For personnel in the reactor building at the time the CFA occurs, an exposure time of 6 minutes is used based detector response time and evacuation time.

- The time for the detector response to activate the building evacuation alarm is estimated as 4 minutes. Time for released activity to reach the ventilation system and detection by the radiation monitoring system is less than 2 minutes. The detector response time is based on a ventilation system flow rate of greater than 60 feet per minute and duct length of 100 feet and detector response time of 0.5 minutes. A response time of 4 minutes is assumed for conservatism.
- The time for personnel to physically exit the reactor building is estimated as 2 minutes. Evacuation time were measured from various locations inside the reactor building to the evacuation exit point and ranged from 10 seconds to 1 minute following initiation of the reactor building evacuation signal. Also, evacuation times were calculated for an average walking pace of 3 mph for the furthest distance in the reactor building to the evacuation exit point. Both the measured time and estimated walking time gave a time of 1 minute. A travel time of 2 minutes is assumed for conservatism.

If an emergency entry after a CFA into the reactor building is necessary, such entries would be made using respirators. Occupational dose to emergency response personnel inside the reactor building was calculated for the first 24 hours following the CFA release while wearing a negative pressure air purifying respirator (APR) with a protection factor of 100. 24 hours is recognized as being an excessive exposure period but is used as 24 hours accounts more than 10 air changes of the reactor building and therefore reduces the initial concentration of airborne activity to negligible levels.

Self-contained breathing apparatus (SCBA) are available with a protection factor of 10,000, as stated in 10 CFR Part 20 Appendix A, if needed. SCBA respirators have a limited air supply, so personnel would enter and exit for brief times. A total of 2 hours wear time was used for response with a SCBA respirator.

Results from Table 13-15 indicate effective and thyroid dose-equivalents are within 10 CFR Part 20 occupational limits. Shallow and lens dose-equivalents were not calculated since these doses are not controlling per 10 CFR Part 20 Appendix B.

Table 13-15 – 24 Hour Effective Dose-Equivalent to Occupational Personnel

Nuclide	Effective Dose from Evacuation (rem[g])	Thyroid Dose from Evacuation (rem)	Effective Dose For 24 Hours with Respirator (APR) (rem[g](h))	Thyroid Dose For 24 Hours with Respirator (APR) (rem[h])
Kr83m[d]	3.71E-09	3.71E-09	8.89E-07	8.89E-07
Kr85[a]	1.89E-05	1.89E-05	4.52E-03	4.52E-03
Kr85m	5.16E-06	5.16E-06	1.24E-03	1.24E-03
Kr87	2.18E-05	2.18E-05	5.22E-03	5.22E-03
Kr88	1.18E-04	1.18E-04	2.82E-02	2.82E-02
Kr89[d]	5.21E-04	5.21E-04	1.20E-01	1.20E-01
Xe131m	9.41E-08	9.41E-08	2.25E-05	2.25E-05
Xe133	4.15E-05	4.15E-05	9.96E-03	9.96E-03
Xe133m	9.17E-07	9.17E-07	2.20E-04	2.20E-04
Xe135m[d]	3.12E-06	3.12E-06	7.41E-04	7.41E-04
Xe135	2.03E-05	2.03E-05	4.87E-03	4.87E-03
Xe137[d]	7.80E-04	7.80E-04	1.80E-01	1.80E-01
Xe138	4.16E-05	4.16E-05	9.88E-03	9.88E-03
I131	9.25E-03	3.85E-01	2.22E-02	9.24E-01
I132	2.65E-05	3.68E-04	6.34E-05	8.81E-04
I133	1.29E-03	4.84E-02	3.09E-03	1.16E-01
I134	5.35E-06		1.28E-05	
I135	1.41E-04	3.36E-03	3.38E-04	8.05E-03
Br83[b]	4.22E-07		1.01E-06	
Br84m[d]	7.99E-07		1.87E-06	
Br84[b]	5.44E-07		1.30E-06	
Br85[d]	4.32E-05		9.86E-05	
Cs138[b,c]	8.33E-05		1.99E-04	
Rb89[c,d]	8.68E-06		2.06E-05	
Rb88[c]	7.85E-05		1.87E-04	
Total	1.25E-02	4.39E-01	3.91E-01	1.41E+00
Total with SCBA Respirator			2 h dose with SCBA: 3.65E-02	2 h dose with SCBA: 3.75E-1

Public doses are calculated for occupied and other locations outside the reactor facility. The following exposure pathways and atmospheric conditions were used:

1. Submersion and inhalation doses were calculated for occupied locations listed in Table 11-13 for (i) fumigation conditions, (ii) calm winds, and (iii) normal (GPM) weather conditions for exposure periods of 2 hours and 24 hours.
2. Submersion and inhalation doses were also calculated at distances from 30 m to 5000 m for (i) fumigation conditions and (ii) normal weather conditions (GPM) for elevations up to 30 m for periods of 2 hours and 24 hours.
3. The maximum [X/Q] value for each weather condition at a given location was used. [X/Q] equations from Section 11 were used; equation 11-15 for 2 hours, equation 11-16 and 11-23 for 2 hours to 24 hours, equation 11-27 for fumigation, and equation 11-28 for calm winds. Equation 11-15 is on the plume centerline for all stability classes. Equation 11-16 is on the plume centerline for stability classes A, B, and C. Equation 11-23 is used for stability classes D, E, and F. Wind speed used was 1 m/s.
4. Dose conversion factors were used from 10 CFR Part 20 Appendix B Table 2 to convert downwind concentrations of airborne activity calculated to dose.
5. Average concentrations for exposure durations of 2 hours and 24 hours were used:
   - 2 hours allows for evacuation of the reactor site and other nearby locations.
   - 24 hours allows for a minimum of 10 air changes of the reactor building, which reduces the initial concentration of airborne activity to negligible levels.
6. Direct exposure from the reactor stack and overhead plume were calculated as depicted in Figure 11-9 and equations 11-43 through 11-46 for released activity from a CFA.

Fumigation conditions provide the least amount of atmospheric dispersion and therefore result in doses that are conservative. Results using the GPM are more typical since fumigation or calm wind conditions occur infrequently.

Results for locations outside the reactor facility are given in the following tables:

- Table 13-16 and Table 13-17 lists effective dose-equivalent from submersion and inhalation to occupied areas near the reactor facility.
- Table 13-18 lists effective dose-equivalent from submersion and inhalation to locations outside the reactor facility at distances from 30 m to 5000 m and elevations from 0 m (ground) to 30 m.
- Table 13-20 lists effective dose-equivalent from direct exposure at two locations outside the reactor facility within the 100 m from the reactor stack.

Table 13-16 – 24 Hour Effective Dose-Equivalent to Occupied Areas Outside the Reactor Facility

Building or Location	Distance x meters	Height z meters	[X/Q] s/m^3 Fumigation ≤ 24 hour	Calm Wind ≤ 24 hour	GPM 24 hour	24 hour Effective Dose-Equivalent Fumigation mrem	Calm Wind mrem	GPM mrem
All	30 to 100	up to 12	8.54E-03	3.99E-04	2.31E-04	5.94E+00	2.78E-01	1.61E-01
All	100 to 150	up to 12	2.46E-04	4.73E-05	2.39E-04	1.71E-01	3.29E-02	1.66E-01
All	150 to 5000	up to 30	2.00E-03	4.89E-05	2.15E-03	1.39E+00	3.40E-02	1.50E+00
Withers, Mann	50	12	5.38E-03	1.73E-04	9.73E-05	3.74E+00	1.20E-01	6.77E-02
Broughton, Riddick	70	12	3.97E-03	9.36E-05	2.26E-04	2.76E+00	6.51E-02	1.57E-01
Patterson, Ricks	90	12	3.17E-03	5.80E-05	2.41E-04	2.21E+00	4.04E-02	1.68E-01
DH Hill	150	30	2.00E-03	2.17E-05	2.15E-03	1.39E+00	1.51E-02	1.50E+00
Cox	175	12	1.73E-03	1.58E-05	2.17E-04	1.20E+00	1.10E-02	1.51E-01
Dabney	200	24	1.54E-03	1.22E-05	5.12E-04	1.07E+00	8.49E-03	3.56E-01
Hillsborough St.	200	15	1.54E-03	1.22E-05	2.59E-04	1.07E+00	8.49E-03	1.80E-01
Talley, Reynolds	200	12	1.54E-03	1.21E-05	2.05E-04	1.07E+00	8.42E-03	1.43E-01
Carroll, Syme	325	12	9.93E-04	4.61E-06	1.64E-04	6.91E-01	3.21E-03	1.14E-01
North	350	20	8.23E-04	3.99E-06	1.76E-04	6.46E-01	2.78E-03	1.22E-01

Table 13-17 – 2 Hour Effective Dose-Equivalent to Occupied Areas Outside the Reactor Facility

Building or Location	Distance x meters	Height z meters	[X/Q] s/m^3			2 hour Effective Dose-Equivalent		
			Fumigation 2 hour	Calm Wind 2 hour	GPM < 2 hour	Fumigation mrem	Calm Wind mrem	GPM mrem
All	30 to 100	up to 12	8.54E-03	3.99E-04	2.31E-04	5.06E-01	2.36E-02	1.37E-02
All	100 to 150	up to 12	2.46E-04	4.73E-05	2.39E-04	1.46E-02	2.80E-03	1.42E-02
All	150 to 5000	up to 30	2.00E-03	4.89E-05	7.57E-03	1.19E-01	2.90E-03	4.49E-01
Withers, Mann	50	12	5.38E-03	1.73E-04	9.73E-05	3.19E-01	1.03E-02	5.77E-03
Broughton, Riddick	70	12	3.97E-03	9.36E-05	2.26E-04	2.35E-01	5.55E-03	1.34E-02
Patterson, Ricks	90	12	3.17E-03	5.80E-05	2.41E-04	1.88E-01	3.44E-03	1.43E-02
DH Hill	150	30	2.00E-03	2.17E-05	7.57E-03	1.19E-01	1.29E-03	4.49E-01
Cox	175	12	1.73E-03	1.58E-05	2.17E-04	1.03E-01	9.36E-04	1.29E-02
Dabney	200	24	1.54E-03	1.22E-05	1.49E-03	9.13E-02	7.23E-04	8.83E-02
Hillsborough St.	200	15	1.54E-03	1.22E-05	2.59E-04	9.13E-02	7.23E-04	1.53E-02
Talley, Reynolds	200	12	1.54E-03	1.21E-05	2.05E-04	9.13E-02	7.17E-04	1.21E-02
Carroll, Syme	325	12	9.93E-04	4.61E-06	1.75E-04	5.88E-02	2.73E-04	1.04E-02
North	350	20	8.23E-04	3.99E-06	4.90E-04	5.51E-02	2.36E-04	2.90E-02

Table 13-18 – 24 Hour and 2 Hour Effective Dose-Equivalent to Locations Outside the Reactor Facility

Distance x	24 hour Public Effective Dose-Equivalent				2 hour Public Effective Dose-Equivalent			
	Height				Height			
	Ground	10 meter	20 meter[a]	30 meter[a]	Ground	10 meter	20 meter[a]	30 meter[a]
meters	mrem	mrem	mrem	mrem	mrem	mrem	mrem	mrem
30	2.59E-03	7.17E-02	1.15E-03	2.17E-02	2.21E-04	1.03E-04	1.15E-03	6.56E-02
50	8.01E-02	2.02E-01	1.17E-03	8.22E-03	6.82E-03	2.90E-04	1.17E-03	2.61E-02
70	1.61E-01	2.16E-01	9.96E-04	4.33E-03	1.37E-02	3.10E-04	1.17E-03	1.42E-02
90	1.81E-01	2.16E-01	7.68E-04	1.43E-05	1.54E-02	3.11E-04	1.11E-03	6.07E-05
100	7.33E-02	1.31E-01	4.54E-01	3.15E+00	6.25E-03	1.12E-02	3.87E-02	9.09E-01
150	1.09E-01	1.41E-01	3.46E-01	1.49E+00	9.25E-03	1.20E-02	3.40E-02	4.49E-01
200	1.18E-01	1.34E-01	2.45E-01	8.86E-01	1.00E-02	1.14E-02	3.45E-02	2.74E-01
250	1.15E-01	1.22E-01	1.80E-01	5.93E-01	9.77E-03	1.04E-02	3.42E-02	1.87E-01
300	1.02E-01	1.15E-01	1.41E-01	4.28E-01	8.70E-03	9.81E-03	3.11E-02	1.38E-01
350	1.01E-01	1.06E-01	1.22E-01	3.25E-01	8.58E-03	9.02E-03	2.90E-02	1.06E-01
400	9.42E-02	9.59E-02	1.06E-01	2.56E-01	8.03E-03	8.64E-03	2.96E-02	8.47E-02
450	8.61E-02	8.62E-02	9.26E-02	2.08E-01	7.34E-03	8.50E-03	2.89E-02	6.95E-02
500	7.79E-02	7.73E-02	8.19E-02	1.73E-01	6.74E-03	8.25E-03	2.77E-02	5.83E-02
600	6.31E-02	6.23E-02	7.12E-02	1.25E-01	6.90E-03	8.09E-03	2.45E-02	4.29E-02
700	5.14E-02	5.07E-02	6.10E-02	9.52E-02	6.59E-03	7.90E-03	2.13E-02	3.32E-02
800	4.24E-02	4.19E-02	5.24E-02	7.52E-02	6.15E-03	7.57E-03	1.85E-02	2.66E-02
900	3.55E-02	3.51E-02	4.52E-02	6.11E-02	6.27E-03	7.18E-03	1.62E-02	2.18E-02
1000	3.01E-02	2.98E-02	3.93E-02	5.08E-02	6.20E-03	6.83E-03	1.42E-02	1.83E-02
1200	2.31E-02	2.36E-02	3.05E-02	3.67E-02	5.77E-03	6.82E-03	1.12E-02	1.35E-02
1400	2.02E-02	2.05E-02	2.47E-02	2.83E-02	5.21E-03	6.50E-03	9.22E-03	1.06E-02
1600	1.79E-02	1.79E-02	2.07E-02	2.28E-02	5.25E-03	6.11E-03	7.83E-03	8.63E-03
1800	1.58E-02	1.57E-02	1.78E-02	1.90E-02	5.15E-03	5.70E-03	6.81E-03	7.28E-03
2000	1.41E-02	1.39E-02	1.56E-02	1.63E-02	4.95E-03	5.32E-03	6.03E-03	6.28E-03
2500	1.11E-02	1.14E-02	1.19E-02	1.19E-02	4.36E-03	4.49E-03	4.71E-03	4.69E-03
3000	9.46E-03	9.55E-03	9.64E-03	9.33E-03	3.80E-03	3.84E-03	3.87E-03	3.75E-03
3500	8.17E-03	8.16E-03	8.06E-03	7.67E-03	3.33E-03	3.33E-03	3.29E-03	3.13E-03
4000	7.12E-03	7.08E-03	6.91E-03	6.50E-03	2.94E-03	2.93E-03	2.85E-03	2.69E-03
5000	5.59E-03	5.53E-03	5.33E-03	4.97E-03	2.36E-03	2.34E-03	2.25E-03	2.10E-03

(a) Note:
No areas are occupied at 20 m and 30 m elevations at distances (x) from 30 m to 150 m.

For direct exposure from distances within 100 m of the stack, two lines of activity in the overhead plume and upper 20 m of the stack are used as described in Section 11.1.1.1 Figure 11-9. Table 13-19 list the source term and Table 13-20 lists 24 hour effective dose-equivalent results at 2 locations within 100 m of the stack.

Table 13-19 – Fission Product Activity within 100m of the Stack

Nuclide	Overhead Line	Stack Line
	curie	curie
Kr83m(d)	4.19E-05	5.82E-06
Kr85(a)	1.42E-03	1.97E-04
Kr85m	1.17E-04	1.62E-05
Kr87	1.23E-04	1.71E-05
Kr88	2.66E-04	3.69E-05
Kr89(d)	5.64E-05	7.82E-06
Xe131m	4.25E-05	5.90E-06
Xe133	4.70E-03	6.51E-04
Xe133m	1.04E-04	1.44E-05
Xe135m(d)	3.14E-05	4.36E-06
Xe135	2.30E-04	3.19E-05
Xe137(d)	8.51E-05	1.18E-05
Xe138	1.86E-04	2.59E-05
I131	8.71E-06	1.21E-06
I132	1.25E-06	1.73E-07
I133	5.47E-06	7.58E-07
I134	1.21E-06	1.67E-07
I135	2.66E-06	3.69E-07
Br83(b)	1.43E-07	1.99E-08
Br84m(d)	8.83E-10	1.22E-10
Br84(b)	1.22E-07	1.70E-08
Br85(d)	4.65E-08	6.45E-09
Cs138(b,c)	5.62E-08	7.80E-09
Rb89(c,d)	1.75E-08	2.43E-09
Rb88(c)	7.93E-08	1.10E-08

Table 13-20 – 24 Hour Effective Dose-Equivalent from Direct Exposure within 100m of the Stack

Source	24 h mrem Location 1 (x,y,z) = (50 m, 0 m, 0 m)	24 h mrem Location 2 (x,y,z) = (50 m, 0 m, 20 m)
Stack Line	3.13×10^{-5}	3.65×10^{-5}
Overhead Line	4.87×10^{-4}	2.11×10^{-3}
Total	5.18×10^{-4}	2.15×10^{-3}

The maximum public dose to occupied areas for 24 hours are estimated to be less than 0.01 mrem from direct exposure and less than 6 mrem from submersion and inhalation. Public dose is therefore well below 10 CFR Part 20 limits for members of the public and emergency action levels based on radiation dose. No protective action recommendation is necessary.[13-19]

Activity in the reactor building air also represents a source of direct radiation. For this calculation, it is assumed the sources remain in the reactor building and are uniformly distributed within the air volume. The reactor building walls were assumed to be equivalent to 12 inches of ordinary concrete. Steel doors to the reactor are ¼ inch thick. Using the above assumptions, the direct radiation level outside the reactor building wall was calculated to 0.03 to 0.6 mrem/h, or 0.7 to 13 mrem in 24 hours. The steel door locations have the higher dose rate. Reactor doors are located inside of Burlington Laboratory, i.e. there is no direct access from the reactor building to locations outdoor areas. This calculation was made using computer code Microshield 5 which uses the point kernel method.[13-20]

The above analysis indicates that a cladding failure accident (CFA) and subsequent release of fission products would not result in dose limits to personnel or members of the public being exceeded.

13.2.6.2. Water-logging

An important consequence resulting from the use of un-bonded fuel assemblies is the possibility of a water-logging failure. Such failure could occur in the event of a defect or leak in the fuel cladding, which would permit in-leakage of water during low power operations or at shutdown. During a subsequent pulse, pressure could be generated in the annulus between fuel pellets and cladding and due to the inability of the steam to escape rapidly through the defect, a failure of the pin could result.

Two cases of such water-logging failures have been documented at other facilities,[13-21] both of which occurred in conjunction with experiments in which holes were drilled through the cladding to accommodate thermocouples. The first case, in which a series of holes were drilled in a line, a pulse having a period of 7.5 milliseconds resulted in a rupture 12 inches in length, which followed the line of drilled holes. In a similar case involving only one thermocouple hole, in which the epoxy sealant deteriorated and allowed water to enter the tube, a 0.78-inch vertical crack developed in the center of a small blister which formed after a pulse.

Since pulsing the NCSU PULSTAR reactor is not permitted, failure of a water-logged fuel pin during pulsing is not applicable. It should be noted, however, that if a defect in the cladding should occur, the fission products would likely escape during the original in-leakage of water, and should serve to indicate the presence of the defective pin.

13.2.7. Experiment Failure

Of the experiments analyzed, direct exposure to a high activity source or beam tube is the worst case for external dose. This accident is prevented by use of engineering controls, experiment limits, and access controls. Radiation monitors and alarming dosimeters are used by personnel entering areas with potentially high external dose rates. Upon receipt of a radiation monitor alarm, the dose to personnel is reduced by evacuating the area and/or equipment actions to reduce the radiation levels. Warning alarms may be used to alert personnel of a status change, e.g. shutter opening, elevated radiation levels. Administrative controls include radioactive material authorizations, radiation work permits, and facility procedures. With controls in place, external dose from experiment failure would not exceed 10 CFR Part 20 limits.

Of the experiments analyzed, a fueled experiment failure is associated with the worst case airborne activity release. This accident is prevented by use of engineering controls and experiment limits. Encapsulation is required by technical specifications (TS). TS limits for fueled experiments are fission rate and total number of fissions to limit personnel and public doses to three percent of the limits given in 10 CFR Part 20 should an accidental release occur. TS 3.8 on fueled experiments is provided in Section 10.

A detailed analysis was performed in the *Safety Analysis in Support of Fueled Experiments for the NCSU PULSTAR Reactor* for accidental releases from fueled experiments.[13-24] Accidental release of fission products associated with a fueled experiment failure is postulated to occur from a non-violent, encapsulation failure, e.g. a rupture or large leak. If encapsulation fails, all fission gas activity and halogen activity is assumed to be immediately released into the reactor building. The design basis for this accident assumes the fission gas and halogen activity is saturated. A release of fission gases and halogens from the design bases accident would be detected shortly after occurring by the radiation monitoring system with activation of the evacuation alarm and confinement system. A less sudden release of fission product activity would be detected by routine air sample analysis of the reactor building and airborne effluent and by higher readings on the radiation monitoring system.

Sections 1 through 9 and Section 14 of the report[13-24] provide information on accidental releases from fueled experiments. Section 2 of the report provides information on the saturation activity calculation. Section 5 of the report provides information on calculations used for released activity and release rates for fueled experiment accidents. Section 8 of the report provides information on calculation of external dose for accidental releases from fueled experiments.

The report[13-24] analyzed accidental releases and radiation dose from fueled experiments. Since the same fission rate limit applies to any fueled experiment, the accident source term of fission gases and halogens at saturation activities is the same for encapsulated or vented fueled experiments. For accidental releases, the radioactive material inventory of fission gases and halogens at saturation activities are assumed to be completely released and then instantaneously and uniformly distributed throughout the entire reactor building free air space. The released materials are then exhausted by the ventilation system and reactor stack to the environment. Exposure times of 2 hours and 24 hours were used in the dose calculations. Two hours is sufficient time to evacuate the reactor building, Burlington labs, and other campus locations. Twenty-four hours is sufficient time to provide more than ten air changes of the reactor building air space thereby removing all of the released activity. The concentration inside the reactor building, decay inside the reactor building, filter retention, exhaust ventilation rate, and atmospheric dispersion are considered in the analysis. Radiation dose from submersion and inhalation and external dose from an overhead plume, reactor stack, and reactor building with contaminated air were analyzed.

Calculation 3 and Calculation 4 of the report[13-24] analyze accidental releases from fueled experiments. Tables 14-3 through 14-6 of the report give data and results for radiation dose from accidental releases from fueled experiments using ^{239}Pu. Tables 14-7 through 14-9 give data and results for radiation dose from accidental releases from fueled experiments using ^{235}U. For accidental releases, the thyroid TODE dose criteria for fueled experiments of 1.5 rem was limiting for both ^{235}U and ^{239}Pu. Fission rates of 1.90×10^{10} f/s for ^{239}Pu and 2.32×10^{10} f/s for ^{235}U were calculated as giving the TODE dose limit to the thyroid for personnel inside the reactor building.

Calculation 5 of the report[13-24] gives radiation doses (TEDE) for accidental releases from fueled experiments in occupied areas outside the reactor building under various weather conditions.

Tables 14-10 of the report lists [X/Q] values calculated as described in Section 6 of the report[13-24]. Maximum [X/Q] values were 8.54×10^{-3} s/m^3 for periods up to 24 h, 7.79×10^{-4} s/m^3 for 96 h, and 9.15×10^{-5} s/m^3 for 520 h.

Tables 14-11 and 14-12 of the report give the TEDE for public areas from ^{235}U and ^{239}Pu for the limiting fission rate of 9.6×10^9 f/s. TEDE in publicly occupied areas outside the reactor building were calculated at a fission rate of 9.6×10^9 f/s to be 3.0×10^{-4} rem or less for an accidental release from a fueled experiment.

Calculation 7 of the report[13-24] gives external dose (DDE) from the reactor building, overhead plume, and reactor stack for accidental releases from fueled experiments. The reactor building air space was assumed to be uniformly contaminated with the accidental release and ventilated by the confinement system for periods of 2 hours and 24 hours. Tables 14-15 and 14-16 give the ^{235}U and ^{239}Pu source terms at a fission rate of 9.6×10^9 f/s. Tables 14-17 through 14-19 give the external dose rates resulting from ^{235}U. Table 14-20 gives the combined external doses for ^{239}Pu. DDE from the reactor building, overhead plume, and reactor stack were calculated to give 2.4×10^{-5} rem or less to publicly occupied areas outside the reactor building. Most of the dose is associated with the reactor building. DDE to occupants inside Burlington labs, but outside the reactor building, was calculated based on an exposure time of 15 minutes to account for time to reach areas outside the site boundary. Exposure time is based on the time for the reactor staff to exit the reactor building and verify personnel within the building have evacuated. This gives a dose of less than 1.4×10^{-5} rem based on a minimal distance of 1 m to 10 m occupied for 15 minutes. It is noted that University personnel notify reactor staff if roof top access is being made. Reactor facility procedures require the reactor staff to clear the roof top if an evacuation alarm occurs. Initial dose rate on the roof top was calculated to be 8.6×10^{-4} rem/h. Exposure time on the building roof is estimated as being less than 15 minutes based on the time for the reactor staff to exit the reactor building and notify personnel on the roof top. This gave a dose of approximately 2.2×10^{-4} rem or less to roof top occupants.

Based on this analysis, radiation dose for accidental releases from a fueled experiment are below three percent of the applicable limits given in 10 CFR Part 20 for occupied areas inside and outside the reactor building. Accidental releases from failure of a fueled experiment do not require an emergency declaration since radiation doses as calculated are below Emergency Action Levels given in the facility Emergency Plan or result in a reportable event as defined in TS 1.2.24.b.

13.2.8. Loss of Normal Electrical Power

The design of the PULSTAR reactor is such that the reactor can be shut down and remain safely in a shutdown condition under a complete loss of electrical power. There are no electrical power supplies that are critical for maintaining the facility in a safe shutdown condition, even for extended periods of time. Since the NCSU PULSTAR does not require emergency backup systems to safely maintain core cooling, there are no credible reactor accidents associated with the loss of electrical power. A backup power system is present at the NCSU PULSTAR that provides powers to the reactor console and radiation alarm panel and to the confinement fans. This system is described in Section 8.

13.2.9. Release of Radioactive Water to Unrestricted Areas

For the analysis on the postulated release of contaminated water, compliance with unrestricted area limits per unit volume is analyzed. The volume giving the 10 CFR Part 20 limit to unrestricted areas is then calculated. The public dose limit of 0.1 rem is associated with the unrestricted area limit if the pubic were to consume water from that location.

Typically, long-lived radionuclides in the reactor pool and waste water meet 10 CFR Part 20 limits for unrestricted areas with no need for dilution. Waste water activity is lower than the primary coolant due to decay and dilution from tap water used at the reactor facility and associated labs.

Storm water and ground water both empty into surface waters as described in Section 2. The water eventually drains to the Neuse River. If the reactor coolant were to enter the Swift Creek catchment basin, Lake Wheeler and Lake Benson have a combined storage capacity exceeding 1 billion gallons as reported in Section 2.

For either case, short-lived radionuclides would decay during transit before entering the nearest supply of pubic water.

The nearest location in which the released water is publicly accessible is the Rocky Branch creek and the nearest drinking water location is the Neuse River. Minimum flow rates given in Section 2 were used to calculate gallons per year and dilution factor.

Using the long-lived radionuclide concentrations given in Section 11 for primary coolant and the surface water data from Section 2 and 10 CFR Part 20, concentrations and public dose were calculated for the Rocky Branch Creek and Neuse River.

A loss rate of 1 gallon per hour of radioactive water at a postulated 10 CFR Part 20 Appendix B fraction of 4.4 gives the following:

Rocky Branch Creek:

Minimum gallons per year = 3.14×10^{8}

Dilution factor = 3.14×10^{8} gallons / 8760 gallons = 3.58×10^{4}

10 CFR Part 20 Appendix B fraction = 4.4/3.58E4 = 1.2×10^{-4}

Neuse River:

Minimum gallons per year = 1.0×10^{11}

Dilution factor = 1.0×10^{11} gallons / 8760 gallons = 1.1×10^{7}

10 CFR Part 20 Appendix B fraction = $4.4/1.1\times10^{7} = 3.8\times10^{-7}$

A continuous loss rate to the unrestricted area exceeding 8000 gallons per hour would result exceeding the unrestricted area limit (i.e. 10 CFR Part 20 fraction of 1 at Rocky Branch creek (8100 gal/h = (1 gal/h) / (1.2×10^{-4}).

The maximum10 CFR Part 20 Appendix B fraction has historically been measured at a value of 10. Using that fraction as the worst case, then the loss rate to unrestricted areas that may reach unrestricted area limits at Rocky Branch Creek is in excess of 3500 gallons per hour (8100 gal/h*4.4/10).

Water loss rates of the magnitude calculated that exceed regulatory limits at the nearest location that is publicly accessible are not credible for the PULSTAR reactor facility.[13-23]

13.2.10. External Events

There are no postulated external events that would adversely affect the reactor as described in Section 2. An Emergency Plan and Security Plan are in place to prevent, mitigate, and respond to external events.

13.2.11. Mishandling or Malfunction of Equipment

Situations involving an operator error at the reactor controls, a malfunction or loss of safety related instruments or controls, and an electrical fault in the control rod system were anticipated during the reactor design stage. As a result many safety features such as control system interlocks and automatic reactor shutdown circuits were designed into the overall control system.

The confinement system consists of two independent and redundant filter trains to help mitigate the consequences of accident scenarios that involve the release of fission products such as fuel cladding failure or failure of a fueled experiment. In the unlikely event that both confinement trains malfunction during one of these accident scenarios then the reactor bay would be isolated until the system can be restored.

No credible accident initiating events were identified for this accident class.

13.3. Summary and Conclusions

This section contains conservative analyses of different types of accidents that are related to the PULSTAR reactor. There is no projected damage to the reactor core as an outcome of the accidents evaluated, except when the core damage is assumed to be part of the accident scenario, such as the fuel handling accident discussed in Section 13.2.6. The robustness of the PULSTAR reactor is the result of the use of passive and engineered safety features in its design.

A review of those accidents which could occur indicates that the maximum hypothetical accident is due to the radiation hazard resulting from a loss of pool coolant accident. Even in this event, there is no core melting or loss of cladding integrity. The associated hazard is related only to the vertical radiation beam emanating from the unshielded shutdown core. Corrective measures can be taken to repair the leak and refill the reactor pool without exceeding occupational dose limits or limits for members of the public.

Therefore, operation of the PULSTAR reactor will present no undue hazard to any member of the general public or to personnel.

13.4. *References*

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14. TECHNICAL SPECIFICATIONS

The current set of NCSU PULSTAR Technical Specifications (TS) are contained in Appendix A to License No. R-120.

The technical specifications satisfy 10 CFR Part 50.34 and 10 CFR Part 50.36 and are consistent with ANSI/ANS 15.1 *The Development of Technical Specifications for Research Reactors* and NUREG 1537 *Appendix 14.1 Format and Content of Technical Specifications for Non-Power Reactors*.

Normal operation of the NCSU PULSTAR Reactor within the limits of these technical specifications will not result in offsite radiation exposure in excess of 10 CFR Part 20 guidelines. Also, observance of these technical specifications limits the likelihood and consequences of malfunctions.

15. FINANCIAL QUALIFICATIONS

15.1. Financial Ability to Construct a Non-Power Reactor

This is not applicable for a renewal application.

15.2. Financial Ability to Operate a Non-Power Reactor

North Carolina State University is a government institution of the State of North Carolina. Pursuant to 10 CFR Part 50.33(f)(2), none of the provisions of 10 CFR Part 50.33(d) apply. The University of North Carolina is incorporated under North Carolina General Statute (NCGS) Chapter 116 Article 1 Section 3.[15-1] NCGS 116-4 provides that North Carolina State University at Raleigh is a constituent institution of the University of North Carolina. NCGS 116-11 (9)(a) provides that the university Board of Governors shall develop, prepare and present to the Governor and the General Assembly a single, unified recommended budget for all of the constituent institutions of The University of North Carolina. The State Budget Allocation line projecting estimated state funding of the NCSU PULSTAR reactor facility (as detailed in Table 15-1 below) is appropriated under this provision.

Pursuant to 10 CFR Part 50.33(f)(2), the estimated annual funding for the first 5 year period after the projected license renewal are given in Table 15-1. The annual operating expenditures for the first 5 year period after the projected license renewal are given in Table 15-2. As indicated, the projected annual funding levels of the reactor facility are equal to the projected annual operating expenditures.

A recent annual financial statement for North Carolina State University detailing the financial solvency of the institution is appended to the facility Financial Qualifications Report.[15-2]

Table 15-1 – PULSTAR Reactor Facility Annual Funding

Fiscal Year	2018	2019	2020	2021	2022
State Budget Allocation [2]	$532,800	$538,100	$543,500	$548,900	$554,400
Services Cost Recovery [3]	$418,700	$428,300	$438,200	$448,300	$458,600
Federal Contracts and Grants [4]	$1,303,800	$1,333,800	$1,364,500	$1,395,900	$1,428,000
Total Funding	$2,255,300	$2,300,200	$2,346,200	$2,393,100	$2,441,000

Table 15-2 – PULSTAR Reactor Annual Operating Expenditures

Fiscal Year	2018	2019	2020	2021	2022
Total Personnel [5]	$1,273,800	$1,303,100	$1,333,100	$1,363,700	$1,395,100
Total Operating and F&A[6]	$981,500	$997,100	$1,013,100	$1,029,400	$1,045,900
Total Expenditures	$2,255,300	$2,300,200	$2,346,200	$2,393,100	$2,441,000

Bases and Assumptions:

1. The budgeted figures detailed in Table 15-1 and Table 15-2 above represent likely funding and expenditures for the specified fiscal years (which run from July 1 of the preceding

year through June 30 of the year indicated). Expenses are projected to increase at an estimated 2.3% annual rate of inflation.

2. The state allocation as appropriated is provided via the NC State University 2-15461 account and is estimated for future years based on a 1% rate of growth.
3. Operating funds required over and above the annual state allocation are provided through reactor services cost recovery and are budgeted for reactor operating expenditures via the university trust account 3-76676.
4. Federal funding for the development of experimental capabilities has been included in these budget estimates. The development of experimental capabilities is funded through sponsored research programs.
5. The budget estimate for 'Personnel' includes salaries and wages for all facility administrative, operating and research staff, graduate students, and student reactor operators.
6. The estimates for total Operating and Facilities and Administrative (F&A) costs include all direct and in-direct (non-personnel) costs of operating and supporting the reactor facility.

Funding for internal N.C. State academic R&D is included under the Federal Contracts and Grants line in Table 15-1 above. The NCSU PULSTAR reactor is also a user facility and as such is utilized by external academic, governmental, and commercial entities for research and development and other irradiation and testing activities. The total projected revenue from all service user activities is estimated under the Services Cost Recovery line of Table 15-1. While the facility does perform some service work for commercial users (amounting to less than 50% of the annual Services Cost Recovery totals given in Table 15-1), the revenue generated from these activities represents significantly less than 50% of the total annual cost of operating the facility as detailed in Table 15-2.

Therefore, pursuant to the requirements of 10 CFR 50.21(c) and 10 CFR 50.22 concerning a class 104(c) research facility, no more than 50 percent of the annual cost of owning and operating the PULSTAR facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training, and as such, the NCSU PULSTAR Reactor should be licensed as a Class 104 facility.

15.3. Financial Ability to Decommission a Non-Power Reactor

Pursuant to 10 CFR Part 50.33(k) and 10 CFR Part 50.75(d), a Decommissioning Report has been submitted as part of the facility Financial Qualifications Report.[15-2] The decommissioning report contains the following information:

1. A cost estimate for decommissioning the facility.
2. Indication of the funding method to be used to provide funding assurance for decommissioning.
3. A means of adjusting the cost estimate and associated funding level periodically over the life of the facility.

Information and detailed analyses from the Decommissioning Report are summarized in sections 15.3.1 through 15.3.3 below.

15.3.1. Decommissioning Cost Estimate

In accordance with the provisions of NUREG 1537 Part 1 Section 15.3,[15-3] a decommissioning cost estimate for the NCSU PULSTAR reactor was obtained through review and comparison with the historical decommissioning costs of similar representative academic research reactor facilities. Decommissioning costs for the reactor facilities at SUNY Buffalo (BMRC) and the University of Michigan were reviewed and analyzed.

The review concluded that the decommissioning cost for the NCSU PULSTAR would be very conservatively bounded by the $14.1M cost of the BMRC PULSTAR decommissioning effort (given in 2014 dollars). Making corrections for the slightly reduced scope of the projected decommissioning effort for the NCSU PULSTAR relative to the BMRC PULSTAR yielded an estimated 2014 decommissioning cost of $12.6M as detailed in Table 15-3 below.

Table 15-3 – Estimated 2017 Decommissioning Costs for the NCSU PULSTAR Reactor

Reactor Facility	Primary Year of Deconstruction	Total 2014 Cost	Labor Adj. L_x	Energy Adj. E_x	Waste Burial Adj. R_x	Estimated 2016 Decom. Cost[1]	Corrected 2017 Decom. Cost[2]
BMRC PULSTAR	2014 *(actual)*	$14.1M	1.047	0.892	1.011	$14,400,000	$14,600,000
NCSU PULSTAR	2014 *(basis)*	*$12.6M est.*	1.047	0.892	1.011	$12,800,000	$13,100,000

[1]The estimated decommissioning cost is corrected from the primary year of deconstruction to 2016 based on the formulas of NUREG-1307 Section 3[15-4] and applying the labor, energy, and waste burial correction factors as given.

[2]The Corrected 2017 Decommissioning Cost is calculated by applying an OECD projected inflation factor of 1.9% from 2016 US dollars.

Applying the methodology of NUREG 1307[15-4] in Table 15-3, the estimate for decommissioning the NCSU PULSTAR is $12.8 million in 2016 dollars as corrected from 2014. Applying an OECD inflation estimate of 1.9%[15-5] to correct from 2016 to 2017 dollars yields an estimated PULSTAR facility decommissioning cost of $13.1 million in 2017 dollars. This total may be subdivided into labor, materials, and services (65%) at $8.5 million, energy and transportation (13%) at $1.7 million, and waste burial (22%) at $2.9 million. Adding a 25% contingency factor per NRC requirements would increase the total estimated cost to $16.4 million. This estimate assumes that NC State will utilize the DECON method of decommissioning, removing and disposing of all radioactive waste offsite. The estimate does not include salaries of PULSTAR reactor personnel contributing to the decommissioning effort.

15.3.2. Funding Method

Pursuant to 10 CFR Part 50.75(e)(1)(iv), North Carolina State University intends to use a Statement of Intent (SOI)[15-6] as the method to provide decommissioning funding assurance. The SOI letter, signed by the University's Vice Chancellor for Finance and Administration, is appended to the Financial Qualifications Report[15-2] and summarized below.

North Carolina State University is a State government organization and decommissioning funding obligations are fully backed by the State government of North Carolina. Decommissioning funds, currently estimated in 2017 dollars in the amount of $13.1 million plus a contingency of $3.3 million, will be appropriated when necessary following the provisions of North Carolina General Statute 116-11(9)(a).[15-1]

In accordance with the provisions of 10 CFR Part 50.75(e)(1)(iv), the signatory of the Statement of Intent, the Vice Chancellor for Finance and Administration, is authorized to execute said document that binds North Carolina State University. Section 3.2 of University Regulation 01.20.02[15-7] states that "The Vice Chancellor for Finance and Administration is delegated the authority to sign all contracts for which the Chancellor has signature authority that may be further sub-delegated." Section 4.7.1.5 of REG 01.20.02 further specifies that "Authority to sign University contracts for use by campus departments, or contracts originating from vendors where the monetary consideration or the value of the agreement, whether monetary consideration is paid or not, or the revenue generated for the University exceeds two hundred fifty thousand dollars ($250,000) is delegated to the Vice Chancellor for Finance and Administration."

15.3.3. Adjustment of Decommissioning Cost Estimate

The 2016 Decommissioning cost estimate for the NCSU PULSTAR reactor facility is $12.8 million as detailed in section 15.3.1 above. This estimate will be updated periodically as required using the methodology described in NUREG 1307[15-4] and detailed below:

From NUREG 1307 Section 3:

$$Estimated\ Cost\ (Year\ X) = (2016\$Cost)(AL_x + BE_x + CB_x)$$ ***Equation 15-1***

where,

$2016\$Cost$ = $12.8 million

A is the labor fraction (65%)

B is the energy fraction (13%)

C is the burial fraction (22%)

L_x is the labor Cost adjustment (REF BLS Code CUI2010000000220I for South region)

E_x is the energy Cost adjustment

P_x is the industrial Electric Power Index (REF BLS Code WPU0543)

F_x is the light Fuel Oils Index (REF BLS Code WPU0573)

Adjustment factors would be calculated as follows (as referenced to 2016 Bureau of Labor Statistics Labor and Producer Price Indices:[15-8]

$$L_x = \frac{\text{Average ECI } (South\ for\ Year\ X)}{\text{Average ECI } (South\ for\ Year\ 2016)}$$ ***Equation 15-2***

$$= \frac{\text{Average ECI } (South\ for\ Year\ X)}{125.7}$$

$$E_x = 0.58P_x + 0.42F_x(PWR)$$ ***Equation 15-3***

where,

$$P_x = \text{(average Year X value of code 0543)/(average 2016 value of code 0543)}$$

$$= \text{(average Year X value of code 0543)/(214.1)}$$

$$E_x = \text{(average Year X value of code 0573)/(average 2016 value of code 0573)}$$

$$= \text{(average Year X value of code 0573)/(134.5)}$$

$$B_x = \text{(Table 2-1 value for genrators in unaffliated states } (PWR\ Year\ X))/(PWR\ 2016)$$

$$= \text{(Table 2-1 value for genrators in unaffliated states } (PWR\ Year\ X))/(12.471)$$

15.4. References

15-1 North Carolina General Statutes, Chapter 116, Higher Education, Article 1, University of North Carolina, http://www.ncga.state.nc.us/enactedlegislation/statutes/html/bychapter/chapter_116.html.

15-2 North Carolina State University PULSTAR Reactor, *Financial Qualifications Report*, March 2017.

15-3 NUREG 1537 Part 1, *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors-Format and Content,* February 1996, https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1537/.

15-4 NUREG-1307, *Report on Waste Burial Charges; Changes in Decommissioning Waste Disposal Costs at Low-Level Waste Burial Facilities Revision. 15*, January 2013, https://www.nrc.gov/docs/ML1302/ML13023A030.pdf.

15-5 Organization for Economic Cooperation and Development (OECD) 2017 U.S. Inflation Forecast, https://data.oecd.org/price/inflation-forecast.htm.

15-6 North Carolina State University PULSTAR Reactor, *Financial Qualifications Report-Attachment 1: Statement of Intent regarding Decommissioning Funding for the PULSTAR Reactor Facility at NC State University*, February 24, 2017.

15-7 North Carolina State University Regulation 01.20.02, *Delegation of Authority to Sign Contracts*, March 29, 2016, https://policies.ncsu.edu/regulation/reg-01-20-02/.

15-8 2016 Bureau of Labor Statistics Labor and Producer Price Index, http://www.bls.gov/data/.

16. OTHER LICENSE CONSIDERATION

16.1. Prior Use of Reactor Components

The NCSU PULSTAR reactor initiated operations in 1972. Section 1 discusses the facility history and major modifications made to the facility. With the exceptions of the modifications described, all the systems, structures, and components installed and operated in the past will continue to be used in the NCSU PULSTAR reactor. Design and operating limits, the potential for age-related degradation, inspections to assess ageing, and a discussion on the suitability for continued use have been included for specific systems, structures, and components in the appropriate section as required.

There are no reactor components in use at the NCSU PULSTAR reactor that have had prior use at any other facility or organization. It is conceivable that prior use components could be integrated into the reactor systems at some future time. Appropriate analysis and reviews of component replacement will be conducted in accordance to applicable standards, regulations and facility license, technical specifications and procedures.

16.2. Medical Use of Non-Power Reactors

The PULSTAR reactor has not been used for medical purposes. Future medical use of the NCSU PULSTAR reactor would be conducted pursuant to appropriate license applications and approvals as authorized by the Atomic Energy Act of 1954 as amended.