USNRC Project No. 99902094

UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN

PRELIMINARY SAFETY ANALYSIS REPORT

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

PREFACE

Nuclear power is undergoing a generational revitalization that is driving innovation across all aspects of the industry. The technology is uniquely suited to the intense demand for clean, reliable, resilient, and accessible energy. Certainly, demand for grid-scale electricity is rising faster than it has in thirty years. At the same time, expanding energy markets are shifting development towards new applications for energy including remote markets, behind the meter deployment, independent microgrids, industrial process integration, and many more. Successfully capturing this opportunity will require a paradigm shift in how advanced reactors, particularly microreactors and small modular reactors, are deployed, operated, and maintained.

The University of Illinois Urbana-Champaign ("U. of I."), through the Department of Nuclear, Plasma & Radiological Engineering, plans to meet this moment and deploy the Nano Nuclear Energy ("Nano") KRONOS™ Micro Modular Reactor (MMR) technology as a microreactor demonstration project. This project will provide a one-of-a-kind research facility specifically targeted at the pursuit of knowledge, experience generation, and technology development that will usher in the next generation of advanced reactor deployments.

The KRONOS MMR is a high temperature gas-cooled reactor (HTGR) design utilizing TRi-structural ISOtropic (TRISO) particle fuel, helium gas as a primary coolant, and graphite as a moderator. The reactor will be located on U. of I.'s campus in Urbana-Champaign, Illinois, and has a molten salt secondary loop providing high temperature heat for electrical power conversion and district heating capability for campus use. The unit will integrate with existing energy infrastructure.

It will be designated as a Class 104(c) non-power utilization facility in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Part 50.21(c), and will be licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

The mission of the U. of I. research reactor is to de-risk advanced reactor deployment and enable a new paradigm of nuclear power through education, research, and at-scale demonstration of next generation technologies and applications. Realization of this mission is through the deployment of advanced nuclear technology in a setting representative of next generation nuclear energy applications. Historically, university leadership in research and development has paved the way for optimized deployment of new technologies. The U. of I. continues this rich tradition to improve operational and deployment characteristics of emerging nuclear technologies.

The first core objective of the facility is to help cultivate the future workforce and address public perception of nuclear power through *Education, Training and Engagement*. This includes educating the next generation of engineers and scientists, training operators in reactor operations, preparing technicians for installation and maintenance, and engaging the general public.

The second core objective is to enable a new paradigm of nuclear energy—characterized by enhanced safety, efficiency, and sustainability—through *Research and Development*. This includes the optimization of reactor components, critical enabling technologies (like high temperature materials, advanced instrumentation, and cybersecurity) and synergistic applications (including integration with additional energy systems).

The cross-cutting objective is to demonstrate the future of nuclear power by *At scale Demonstration*. Many vendors are proposing co-location of Small Modular Reactors and Microreactors with industrial applications which need both electricity and high temperature heat as a means of enhancing the economic proposition of their technology. This includes generation of electricity, district heat, integrated thermal energy storage, and potential hydrogen production. By showcasing large-scale applications—such as reliable electricity generation, district heating solutions, advanced thermal energy storage, and hydrogen production—the project aims to highlight nuclear power's versatility and future potential.

U. of I. will collect operational data to strengthen its nuclear research and educational programs and to provide the State of Illinois with a technical resource that supports its transition toward a zero-carbon energy system.

Additionally, Nano will collect high-resolution performance and systems-level data to guide reactor optimization and inform design evolution for future commercial deployment scenarios, and the facility may be used to test design options for non-safety related systems, as part of the iterative development and testing approach. The U. of I. research reactor will also serve as a new operator training facility for future advanced reactors and provide a platform for familiarization with an at-scale facility for all maintenance crews.

This Preliminary Safety Analysis Report describes the facility in detail and is being submitted to the U.S. Nuclear Regulatory Commission (NRC) in support of a Construction Permit Application (Project 99902094), concluding the significant preapplication effort. The format and content of this document follows the guidance of NUREG-1537, Part 1, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content,” Part 2, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Standard Review Plan and Acceptance Criteria,” and Final Interim Staff Guidance augmenting NUREG-1537.

Efforts were directed on ensuring that each chapter contains the technical content and necessary information stated in NUREG-1537, including the regional demographic, hydrological, seismological, and meteorological data, and provides preliminary system descriptions and accident analyses using current computer codes and methodologies.

The preparation of this document represents the collective cooperation, support and efforts of many individuals, including members of the facility staff and external organizations.

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
ALARA	As Low As (is) Reasonably Achievable
AVR	German Arbeitsgemeinschaft Versuchsreaktor, prototype pebble bed reactor
CFR	Code of Federal Regulations
CPA	Construction Permit Application
CSP	Concentrated Solar Power
DOE	Department of Energy
ECCS	Emergency Core Cooling System
ESF	Engineered Safety Feature
FHSS	Fuel Handling and Storage System
FPS	Fire Protection System
HTGR	High Temperature Gas Cooled Reactor
HTTR	High Temperature Test Reactor
HVAC	Heating, Ventilation & Air Conditioning
I&C	Instrumentation and Control
LWR	Light Water Reactor
MHTGR	Modular High Temperature Gas Cooled Reactor
MMR	Micro Modular Reactor
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
NSR	Non-Safety Related
NWPA	Nuclear Waste Policy Act
OLA	Operating License Application
PBMR	Pebble Bed Modular Reactor
PDC	Principal Design Criteria
PSAR	Preliminary Safety Analysis Report
RG	Regulatory Guide
RPP	Radiation Protection Program
RPS	Reactor Protection System
SE	Safety Evaluation
SR	Safety Related
SSCs	Structures, Systems and Components
TEDE	Total Effective Dose Equivalent
THTR	Thorium High Temperature Reactor
TRISO	Tri-structural Isotropic
U. of I.	University of Illinois Urbana-Champaign
ULTB	Uranium Lease and Take-Back
UMSs	Unit Monitoring Systems

CHAPTER 1 THE FACILITY

1.1 INTRODUCTION

Nano Nuclear Energy Inc. (referred to hereafter as "Nano") is developing the KRONOS^{TM} Micro Modular Reactor (MMR) technology. The project to deploy the KRONOS MMR at University of Illinois Urbana-Champaign (U. of I.) is referred to hereafter as the "U. of I. research reactor". The KRONOS MMR proposed for construction is a Non-Power Research Reactor located on the U. of I. campus. It is designated as a Class 104(c) non-power utilization facility in accordance with Title 10 of the Code of Federal Regulations (10 CFR) Part 50.21(c), and licensed under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

U. of I., the applicant, is requesting a Construction Permit for a research reactor. This is a 45 MW_{th} non-power research reactor that will be located on the U. of I. campus near the intersection of South Oak Street and East Gregory Drive on the western edge of the campus. The entire reactor site is part of the university campus and wholly owned by the university. Details of the site location are further described in Section 2.1. The purpose of the research reactor is to demonstrate commercial readiness of the advanced reactor technologies, integration with variable energy needs, and operational flexibility through education, research, and at-scale demonstrations via this deployment and operation.

This Preliminary Safety Analysis Report (PSAR) is submitted in accordance with the requirements stated in 10 CFR 50.34(a) in support of the Construction Permit Application (CPA). The PSAR generally follows the content and organization of guidance provided in NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content," and Part 2, "Standard Review Plan and Acceptance Criteria," as augmented by the "Final Interim Staff Guidance Augmenting NUREG-1537," Parts 1 & 2 (Reference 1-1).

The content of this PSAR has been organized and formatted specifically as described in the "Proposed Contents of PSAR Using NUREG-1537 Guidance for the Micro Modular Reactor (MMR)" White Paper (Reference 1-2) and the resulting NRC Staff Observations (Reference 1-3). Unless otherwise specifically exempted, the U. of I. research reactor PSAR shall comply with the regulations listed in the approved "Applicability of Nuclear Regulatory Commission Regulations" Topical Report, subject to the limitations and conditions listed in the Safety Evaluation (SE) (Reference 1-4).

A detailed evaluation of applicable regulations and other necessary requirements for licensing the U. of I. research reactor will be conducted following the completion of the facility's design and submission of the CPA. A final Nuclear Regulatory Commission (NRC) staff determination regarding the applicable regulations will await review at the time of the Operating License Application (OLA) based on the regulations that are in effect at that time.

An overview description of the inherent and passive safety features of the facility is addressed in Section 1.2.2.

1.2 SUMMARY AND CONCLUSIONS ON PRINCIPAL SAFETY CONSIDERATIONS

U. of I. has a long history of supporting nuclear technology innovation, which includes operating a TRIGA® research reactor from 1960 until 1998. The proposed U. of I. Research Reactor is based on the KRONOS MMR technology which is a helium-cooled, graphite moderated prismatic block high-temperature gas-cooled reactor (HTGR) utilizing tri-structural isotropic (TRISO) fuel particles in ceramic annular fuel pellets.

The KRONOS MMR technology is considerably different from Light Water Reactor (LWR) technologies. The design relies on passive decay heat removal and does not require an emergency core cooling system to replenish primary coolant to maintain fuel cooling in the event of a rupture of the primary coolant pressure boundary. The KRONOS MMR is designed for a passive safety response to postulated events and relies on functional containment of the TRISO particles and ceramic fuel pellets to limit release of radioactive material to the environment. Large safety margins and negative reactivity feedback are achieved as a result of both the TRISO-based ceramic fuel and the reactor design. These inherent safety characteristics simplify the system, reduce the need for active safety systems to maintain fuel integrity, and limit the potential release of radioactive material during normal operation and postulated events.

The research reactor uses helium gas as the primary coolant. A circulator moves helium up along the outer annulus of the reactor vessel into an upper plenum where it turns and flows downward through the core, through and around the annular fuel pellets and eventually to the intermediate heat exchanger where heat is transferred to a molten salt intermediate loop. Helium is a chemically inert gas, thereby preventing adverse reactions with the fuel and structures of the reactor core system. Because helium remains gaseous under all normal and accident conditions, no phase changes of the coolant are possible, pressure measurements are accurate, and circulator (pump) cavitation cannot occur. Helium is also transparent to neutrons, and as such does not impact core reactivity nor does it become highly activated under irradiation.

The reactor is located below ground. Although it does not have nor need a credited, leak-tested containment building, it is surrounded by the Citadel building. This is a concrete structure that serves as an additional layer of defense-in-depth to minimize the release of radioactive material to the environment and provide protection against external hazards.

The main research reactor Systems, Structures and Components (SSCs) are listed in Table 1-1.

Table 1-1 System Designators and System Figures

PSAR SSCs	PSAR Section
Access Control	12.8
Backup Electrical Power System	8.3
Chemical Dosing System	9.7.3
Chilled Water System	9.7.1.3.2
Citadel Building Monitoring System	7.8
Citadel Radiation Monitors and Helium Detection Systems	7.8.1.2
Communications and Information System	9.4
Demineralized Water Distribution System	9.7.1.3.1
Electrical Power Systems	8.1
Ex-core Instrumentation System	7.8
Facility-to-off-site Radiotelephone System	9.4.4.2.2
Fire Detection and Alarm Systems	9.3.3.1
Fire Protection Systems	9.3
Fire Water Supply and Water Based Systems	9.3.3.4
Fixed Fire Suppression Systems	9.3.3.2
Fresh Fuel Handling and Temporary Storage System	9.2.4.1
Fuel Handling and Storage System	9.2
{{ }}a(4)	9.5.4.2.1
Fuel Handling Vertical Shaft	9.2.4.2.2
Heat Tracing System	8.2.3.1
Heat Transport System	5.2
Helium Pressure Relief System	4.3
Helium Purification System	5.4
Helium Recovery System	5.4.2.2
Helium Activity Monitoring/Failed Fuel Detection System	7.8
Hot Water System	9.7.1.3.3
Instrumentation and Control System	7.2
Intercom System	9.4.4.4
Intra-Facility Radiotelephone System	9.4.4.2.1
Leakage Monitoring System	7.8
Lifting and Rigging Equipment	9.7.2
Liquid Radioactive Waste System	11.1.1, 2.6.1.2
Low Voltage AC Distribution System	8.2.2.1
Low Voltage Direct Current (DC) Power System	8.2.2.2
Portable Monitoring Systems	8.3.2.1
Post-Accident Monitoring System	7.8
Potable Water Distribution System	9.7.1.3.4

Table 1-1 System Designators and System Figures (Continued)

PSAR SSCs	PSAR Section
Primary Coolant Make-up System	5.5
Primary Coolant System	5.2
Public Address System	9.4.4.3
Radioactivity Monitoring System	7.8
Radiotelephone System	9.4.4.2
Reactivity Control and Shutdown System	4.2.3
Reactor Cavity Cooling System	6.3
Reactor Control System	7.3
Reactor Core System	4.2.2
Heating, Ventilation, and Air Conditioning System	9.1
Reactor Internals System	4.2.5
Reactor Protection System	7.4
Safety Parameter Display System	7.7.1
Salt Activity Monitoring System	7.8
Secondary Coolant System	5-3
Seismic Monitoring System	7.8
Sensors and Instrumentation	7.6
Sewage Water System	9.7.1.3.5
Solid Radioactive Waste System	2.6.1.2
Spent Fuel Handling and Interim Storage System	9.2.4.3
Supervisory Control System	7.3
Telephone System	9.4.4.1
Thermal Energy Storage System	5.3
Uninterruptible Power Supply	8.3.2.1
Unit Monitoring System	7.8
Vessel System	4.3
Waste Processing	11.2
Water Services System	9.7.1

1.2.1 Consequences from the Operation and Use of the Facility

A key measure of safety and consequence from the operation of the facility is the magnitude of the potential source term associated with off-normal events. The source term represents the amount, timing, and nature of radioactive material released and available for release to the environment following a postulated event. The functional containment represents an Engineered Safety Feature (ESF) of the reactor design and is ensured by the TRISO-based ceramic fuel. The operating temperatures of fuel pellets in the reactor core do not challenge the temperature limits of TRISO particles or of the SiC matrix. The fuel design and performance are discussed further in Section 4.2.

The facility operating staff are subject to occupational radiation exposure from working in a facility that contains radioactive materials. Members of the public are potentially subject to limited exposure from radiological effluent releases during normal operations. For normal operation, such exposures are maintained below the dose limits of 10 CFR 20.1201 (Subpart C) and 10 CFR 20.1301 (Subpart D) for the operating staff and members of the public, respectively. Potential doses to the public resulting from postulated events are maintained by design to be well within the criterion of 10 CFR 50.34. The radiation protection program which addresses the limits in 10 CFR 20 as described in Chapter 11 will be provided in the OLA consistent with 10 CFR 50.34(b)(3).

1.2.2 Inherent and Passive Safety Features

The research reactor design includes several safety features that ensure large margins and enable simplified system design:

- The fuel uses TRISO particles, whose coatings can withstand much higher temperatures than zircalloy cladding before degrading, giving a significantly larger margin than LWR fuel to normal operating temperatures. Unlike LWR fuel (e.g., potential for leaking full-length rod), the discrete TRISO fuel particles limit large releases of inventory. The fuel design is described in Section 4.2.
- The reactor design relies on a functional containment approach as described in SECY-18-0096 (Reference 1-5). Functional containment is defined in Regulatory Guide (RG) 1.232 (Reference 1-6) as a barrier or set of barriers that effectively limit physical transport of radioactive material to the environment. A zero-maintenance functional containment served by the TRISO-based ceramic fuel is fundamental to the superior fission product retention capability of the reactor design, in contrast to a large, high maintenance containment building. The functional containment approach is described in Section 6.2.
- The primary heat transfer fluid is helium, an inert, chemically stable, single-phase gas that is not highly activated by radiation. Neutron moderation is provided separately from the coolant. The reactor coolant is discussed in Section 5.1.
- The overall negative temperature reactivity coefficient inherent to the reactor core design provides inherent safety. The core design is described in Section 4.5.
- For postulated events, the reactor is designed to perform safety functions without helium, electrical power, or operator action. Decay heat removal requirements are small due to core size and power density. The evaluation of postulated events is described in Chapter 13.
- Secondary heat transfer is ensured by a molten salt loop that effectively isolates the reactor from transients and upset conditions that could occur in the steam generation and power conversion systems.
- The reactor is surrounded by a concrete structure (the Citadel building) below ground, which serves as an additional layer of defense-in-depth against release of radioactivity to the environment and provides protection against external hazards. The building structural design and natural phenomena protective qualities are described in Chapter 3.

1.2.3 Design Features and Design Bases

The Principal Design Criteria (PDC) for the facility Structures, Systems and Components (SSCs) are described in Section 3.1 and are based on the approved Topical Report (Reference 1-7). The various sections of this PSAR provide descriptions of how the design criteria, including the PDCs, are incorporated into the design bases of the relevant reactor SSCs.

The research reactor design bases support passive safety responses to postulated events and rely on a functional containment to limit release of radioactive material to the environment. Unlike LWRs, the reactor does not require a credited containment structure nor active decay heat removal systems. Following loss of forced cooling events, the reactor passively reaches a safe and stable state without the need for electrical power or operator intervention. Large safety margins are provided by both the fuel and the reactor design.

Chapter 3 describes all Safety Related (SR) SSCs required for safe shutdown and decay heat removal, which are located below grade and protected from the potential consequences of tornadoes, hurricanes, floods, explosions, and toxic gases.

The research reactor SSCs are classified per the approved Event Sequence Identification and SSC Safety Classification Methodology Topical Report (Reference 1-8). U. of I. assigned the following safety classification groups for SSCs as described in Sections 3.2.1 and 3.2.2 of the Topical Report.

- Safety Related (SR): SSCs that have an impact on safety and are relied upon to remain functional to meet the three safety functions described in Section 3.6.1 during and following all event sequences as part of the plant design basis.
- Non-Safety Related (NSR): SSCs not required to remain functional to meet the three safety functions described in Section 3.6.1.

Note that some of the NSR SSCs may be considered as an additional layer of defense-in-depth to support the ALARA dose approach. A summary of SSC safety classifications is provided in Table 3-2.

1.2.4 Potential Events at the Facility

Chapter 2 provides potential events related to industrial facilities, transportation, and weather.

Chapter 13 provides an evaluation of postulated events. Postulated events are identified by the application of hazard analysis methodologies to evaluate the design of the facility and processes for potential hazards, initiating events, scenarios, and associated prevention and mitigation controls. This analysis is consistent with the accident analysis required for research reactors and demonstrates that even in the presence of highly conservative assumptions, no member of the public at the site boundary would receive a radiation dose in excess of 1 rem (10 mSv) Total Effective Dose Equivalent (TEDE) for the duration of the accident.

Accidents involving combustible gases are not applicable to the research reactor. Postulated events relevant to the research reactor do not feature phenomena that result in the generation of combustible gas (combustible hydrogen (tritium) is generated but in negligibly small amounts). As a result, combustible gases do not represent a hazard to the integrity of the functional containment barrier and its fission product retention capability.

1.3 GENERAL DESCRIPTION OF THE FACILITY

1.3.1 Geographical Location

The facility is located within the U. of I. campus in the city and county of Champaign, IL. The latitude and longitude, including an illustration of the site location, is provided within Section 2.1.

1.3.2 Principal Characteristics of the Site

The project site encompasses approximately 1.01 acres and is located east of the existing Abbott Power Plant and cooling towers, a hybrid natural gas and coal facility that serves the university-owned and operated electrical and steam distribution system. Currently, the reactor site functions as a parking lot.

Refer to Chapter 2 for further characteristics of the site.

1.3.3 Principal Design Criteria, Operating Characteristics, and Safety Systems

1.3.3.1 Principal Design Criteria

The PDC for the U. of I. research reactor are based on the criteria provided within the NRC SE issued for the approved topical report (Reference 1-7) and are described in Section 3.1.

1.3.3.2 Operating Characteristics

The research reactor is designed as a 45 MW_{th} reactor. The reactor design and operational characteristics are detailed in Chapter 4 and Table 4-1.

1.3.3.3 Safety Systems

Section 3.6 describes the fundamental safety functions and the safety classification of SSCs with respect to those functions. It identifies the SR SSCs and their design bases.

1.3.4 Engineered Safety Features

The ESFs are SSCs designed to mitigate the consequences of postulated events. The ESFs are described in Chapter 6. None of the research reactor ESFs are active systems as they relate to the containment of fission products and the passive removal of decay heat.

1.3.5 Instrumentation, Control, and Electrical Power Systems

The instrumentation and control (I&C) systems are described in Chapter 7. The I&C systems monitor and control the reactor system during normal operations and planned transients. They also monitor and actuate the Reactor Protection System (RPS) during postulated events.

The electrical power systems, described in Chapter 8, provide normal and backup power to the facility.

1.3.6 Cooling and Other Auxiliary Systems

The Heating, Ventilation, and Air Conditioning (HVAC) systems, discussed in Section 9.1, condition the interior environment and control airborne contaminants. It is divided into two pressure zones based on their potential for contamination.

Section 9.2 explains how the Fuel Handling and Storage System (FHSS) manages nuclear fuel before and after use in the reactor core.

The Fire Protection System (FPS) is described in Section 9.3. It is designed to detect, control and extinguish fires so that a continuing fire will not prevent safe shutdown or result in an uncontrolled release of radioactive material that exceeds acceptance criteria.

The remainder of Chapter 9 describes the other research reactor facility auxiliary systems.

1.3.7 Radioactive Waste Management and Radiation Protection

A Radiation Protection Program (RPP) will be established to protect the health and safety of workers from radiological effects, as necessary to comply with the regulatory requirements of 10 CFR Parts 19, 20, and 70. The RPP will include the elements of an as low as is reasonably achievable (ALARA) program, radiation monitoring and surveying, exposure control, dosimetry, contamination control, and environmental monitoring. The RPP is addressed in Section 11.1 and will be provided in the Operating License Application (OLA) consistent with 10 CFR 50.34(b)(3).

The facility also includes capabilities for the management of liquid, gaseous, and solid radioactive wastes produced by plant operations. The radioactive waste management systems are described in Section 11.2.

1.3.8 Experimental Facilities and Capabilities

The principal purpose of the non-power reactor is for research and at-scale demonstration of the advanced reactor technology implementation. The reactor includes the capability to irradiate material samples in experimental wells (outside of the reactor vessel), to support on-going and emerging research. The research reactor facility will support educational needs about advanced reactor technologies, demonstrate their operational flexibility, and demonstrate integration with existing and new power generation infrastructure having variable load demand.

1.3.9 Research and Development

Per 10 CFR 50.34(a)(8), the research and development needed to complete the design adequacy and safety evaluation of any plant SSCs will be completed prior to completion of construction of the facility.

U. of I. will collect operational data from the research reactor to strengthen its nuclear research and educational programs and to provide the State of Illinois with a technical resource that supports its transition toward a zero-carbon energy system.

Additionally, NANO will collect high-resolution performance and systems-level data to guide reactor optimization and inform design evolution for future commercial deployment scenarios, and the facility may be used to test design options for non-safety related systems, as part of the iterative development and testing approach. Additionally, the U. of I. research reactor will serve as a new operator training facility for future advanced reactors and provide a platform for familiarization with an at-scale facility for all maintenance crews.

The details of the research and development testing to be performed in the facility during research reactor operation will be provided in the OLA.

1.4 SHARED FACILITIES AND EQUIPMENT

The U. of I. research reactor is a standalone single-unit reactor facility that does not share any safety related SSCs with other facilities, other than those specifically identified, discussed, and evaluated in this report.

The reactor site may share infrastructure or non-safety related SSCs with nearby facilities. Those include, but are not limited to, the AC electrical power supply, water supply systems, warehousing and storage, and infrastructure related to site access roads. The detailed descriptions and safety implications of shared equipment or functions are provided in the respective PSAR chapters.

The U. of I. research reactor is designed to accommodate all uses or malfunctions of shared facilities without degradation of the reactor safety features, and to avoid the potential spread of contamination to the shared facilities or equipment, utilizing appropriate electronic or physical barriers where needed.

Hazards to safe operation originating from nearby industrial, transportation, military facilities and environmental hazards are discussed and evaluated in Chapter 2.

1.5 COMPARISON WITH SIMILAR FACILITIES

The research reactor design utilizes the operating experience of HTGRs accumulated worldwide and in the U.S. over the last 50 years. It builds upon the lessons learned from other HTGRs, both designed, operating and decommissioned. There have been seven HTGR types (Reference 1-9) built and operated in the world, in addition to research and development for future planned HTGR designs that have advanced the knowledge and industrial base. These are summarized in Table 1-2 and discussed below.

The operating conditions of the research reactor have been selected to be informed by other HTGR designs, as shown in Table 1-2 below. The reactor power, inlet and outlet temperature, and primary coolant pressure are well within industrial experience, thus allowing the use of well-characterized materials and systems.

Table 1-2 Key Specifications for Other HTGRs Compared to the KRONOS Reactor (Reference 1-9, Reference 1-10)

Reactors	Dragon	Peach Bottom	AVR (Germany)	Fort St. Vrain	THTR (Germany)	HTTR (Japan)	HTR-10 (China)	AGRs (UK)	HTR-PM (China)	MHTGR	KRONOS Reactor
Thermal Power (MWth)	21.5	115	46	842	750	30	10	1500	250/unit (2 units)	350	45
Power Density (MW/m^3)	14	8.3	2.6	6.3	6	2.5	2	6	---	5.9	{{ }}[a(4)]
Primary Coolant	He	He	He	He	He	He	He	CO_2	He	He	He
Secondary Coolant	Steam	Steam	Steam	Steam	Steam	He/ Pressurized Water	Steam	Steam	Steam	Steam	Molten Salt
Primary Coolant Pressure (MPa)	2	2.3	1.1	4.8	4	4	3	4.1	7	6.4	6.0
Primary Coolant Flow Rate (kg/s)	9.62	60	13	440	51.2	10.2-12.4	3.2-4.3	3790	96	158	24.1
Reactor Inlet Temperature (°C)	350	327	275	404	250	395	250	278	250	258	300
Reactor Outlet Temperature (°C)	750	700-726	950	777	750	850-950	700	635-675	750	686	660
RPV Material	Carbon Steel	Carbon Steel	Steel and concrete building	PCRV with liner	PCRV	2-1/4Cr-1 Mo Steel	C-Mn-Si Steel	PCRV	SA-508	Steel	Carbon Steel
Core Structure Type	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite	Graphite
Fuel Element Type	Prismatic Block	Prismatic Block	Pebble Bed	Prismatic Block	Pebble Bed	Prismatic Block	Pebble Bed	Prismatic Block	Pebble bed	Prismatic Block	Prismatic Block
Fuel	(U, Th, Pu) O_2 TRISO	(U, Th)C_2 BISO	(U, Th)O_2 BISO; then TRISO	UO_2 & ThO_2-UO_2 TRISO	(U, Th)O_2 BISO	UO_2 TRISO	UO_2 TRISO	UO_2	UO_2 TRISO	UCO TRISO	UCO TRISO
Enrichment (wt%)	93	93	93	93	93	3-10 (avg. 6)	17	2.5-3.5	LEU	19	Low-enriched uranium (LEU+) ≤9.90 wt%
Years Operational	1964-1975	1966-1974	1967-1988	1976-1989	1985-1991	1998-Present	2000-Present	1962-Present	2021-Present	Planned	Planned

Key observations and summaries of other HTGRs include:

- The use of graphite as a moderator has been demonstrated in the 14 Advanced Gas-Cooled Reactors (AGRs) operated in the United Kingdom, using CO_2 as coolant.
- The use of helium as the coolant has been demonstrated in several HTGRs. Specifically, the Peach Bottom Unit 1 and Dragon reactors have successfully demonstrated that helium is an excellent coolant medium in a high-temperature graphite-moderated reactor core. Peach Bottom 1 has also demonstrated load following capabilities and low operator dose achievable with this type of reactor.
- The AVR demonstrated a reactor outlet temperature of up to 950°C over 21 years of operation. This reactor also demonstrated the superior fission product retention of TRISO fuel over earlier types. The AVR successfully demonstrated a loss of coolant accident with Anticipated Transient without Scram with passive shutdown using the negative temperature coefficient of reactivity.
- The Fort St. Vrain reactor demonstrated excellent fuel performance and low operator dose. This reactor was also used to demonstrate the fuel handling and refueling approach for a prismatic fuel block reactor. The Fort St. Vrain facility also successfully implemented the "immediate dismantling" decommissioning strategy.
- The Thorium High-temperature Reactor (THTR) was a commercial-scale demonstrator of a pebble bed type reactor. The THTR also demonstrated co-location next to an industrial plant (coal fired power station) with a shared turbine hall. The THTR further demonstrated dry-cooling and achieved very low maintenance doses.
- The High Temperature Test Reactor (HTTR) and HTR-10 are still operating reactors used for research on future applications and for demonstrations of the safety concepts of HTGRs. The HTR-10 has successfully demonstrated the inherent shutdown capability via two safety demonstration experiments: (i) loss of forced cooling by tripping the circulator and (ii) reactivity insertion by means of control rod withdrawal. A safety demonstration test in HTTR demonstrated the passively safe response due to a loss of forced cooling event. HTTR and HTR-10 have also successfully demonstrated the operational aspects of small reactors, including staffing and procedures. The U. of I. research reactor is most similar to the HTTR.
- High temperature reactor design concepts in Germany (HTR-Modul), South Africa (PBMR), the U.S. (MHTGR, NGNP), and China (HTR-PM) have contributed to significantly improving the knowledge base and have laid the foundation for passive safety design, enhanced TRISO fuel performance, and the use of HTGRs for both power and heat applications. In the U.S., work on the MHTGR and NGNP concepts by government and industry have addressed many of the technology challenges identified in the design, construction, and operational experience of past reactor designs.
- Research and development programs led by U.S. national laboratories over the last several decades have advanced many of the capabilities being leveraged by the U. of I. research reactor design. Notably, TRISO fuel performance has been extensively tested by the Advanced Gas Reactor Fuel Development Program and forms the basis for the fuel qualification program. Other areas of advancement include:
  - TRISO-fueled core modeling & simulation
  - Enhanced reactor physics understanding
  - Nuclear analysis and performance data
  - Passive core cooling solutions

- Graphite irradiation behavior

The U. of I. research reactor design is based on the lessons learned from the operating reactors and the knowledge base that has been extended by the other designs. The design philosophy for this reactor has been to stay well within all known engineering limits (e.g., operating pressure, temperature, stresses...), make vast use of commercial off-the-shelf technology solutions and components, and to adhere to all long-established codes & standards and regulations.

Areas where the operating base used in the reactor design goes beyond previous HTGR designs is limited to: 1) Embedding TRISO fuel particles in a ceramic (SiC) matrix, instead of a graphite matrix. This fuel form will be tested through an extensive fuel qualification program, as described in Section 4.2. The use of the ceramic fuel matrix is expected to enhance the fission product retention even further than TRISO particles alone, resulting in increased margin to the dose consequence criteria, 2) Integration of Concentrated Solar Plant (CSP) technologies as part of the reactor's intermediate loop, which uses molten salt coolant and includes integrated thermal storage. This use of these technologies eliminates the potential for high pressure steam ingress into the pressure boundary, that could lead to chemical attack of the graphite, and effectively isolates the reactor from normal load fluctuations from the power conversion systems and therefore minimizing power change operations on the reactor side. The technologies used as part of the intermediate loop are based on the decades of experience of the CSPs operating across the world.

1.6 SUMMARY OF OPERATIONS

As discussed in Section 1.1, the purpose of the U. of I. research reactor is to demonstrate the commercial readiness of the KRONOS MMR technology, as well as its capabilities for operational flexibility and integration with existing energy infrastructure. Education, research, and at-scale demonstrations will be enabled via the deployment and operation of the research reactor. This includes obtaining operational insight into key technologies, design features, and collection of high-fidelity data for use in safety analysis and computational methodologies. These insights can support future enhancement and development of similar types of reactors, including load-following microreactors and HTGRs. The major programs to be performed in the facility will be provided in the OLA consistent with 10 CFR 50.34(b)(2).

The U. of I. research reactor is an HTGR designed by NANO that will be licensed to operate at a maximum power level of 45 MW_{th}. At the maximum rated capacity, the reactor will be refueled approximately every 3 years of equivalent full-power operation. The reactor plant is designed for an operational lifetime of 40 years. The research reactor will be operated over the approved range of power levels and conditions in order to significantly augment all aspects of the embedded technologies.

The research reactor includes the necessary features to monitor and assess plant performance as described in the respective areas of this report. The radioactive byproduct and fission product inventory from the normal operation of the facility and effluent release pathways to the environment is discussed in Chapter 11. An analysis of postulated events from operation of the facility, including the radiological consequences of unplanned releases, is addressed in Chapter 13.

The description of the radiation sources and effluent releases including fission product decay heat for the facility will be detailed in the OLA consistent with 10 CFR 50.34(b)(3).

1.7 COMPLIANCE WITH THE NUCLEAR WASTE POLICY ACT (NWPA) OF 1982

The U. of I. research reactor will be covered under the U.S. Department of Energy (DOE) Uranium Lease and Take-Back (ULTB) program which provides fresh fuel to domestic university research reactors and collects the spent fuel at the end of the operating life. Disposal of high-level nuclear waste and spent fuel will be covered by the ULTB program.

U. of I. intends to enter a ULTB contract with the DOE that commits the DOE to collect and manage the U. of I. high-level waste and/or spent nuclear fuel resulting from the research reactor operation. A letter of support for a contract between U. of I. and the DOE is included in Chapter 15. Details of the ULTB Program agreement between U. of I. and DOE, and required compliance by the licensee with the NWPA, will be provided in the OLA, consistent with Section 302(b)(1) of the NWPA.

1.8 FACILITY MODIFICATIONS AND HISTORY

This PSAR is provided as part of a CPA for a new non-power research reactor facility. As a new facility undergoing initial construction, commissioning, and operations, there are no facility modifications or prior history related to this application.

1.9 REFERENCES

1-1 NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Part 1, "Format and Content," (ADAMS Accession Number ML042430055) and Part 2, "Standard Review Plan and Acceptance Criteria," (ADAMS Accession Number ML042430048), February 1996; Final Interim Staff Guidance Augmenting NUREG-1537, "Guidelines for Preparing and Reviewing Applications for Licensing Non-Power Reactors:" Part 1, "Format and Content," (ADAMS Accession Number ML12156A069) and Part 2 "Standard Review Plan and Acceptance Criteria," (ADAMS Accession Number ML12156A075), October 17, 2012

1-2 U. of I. White Paper IMRDD-MMR-22-02, Release 01, "University of Illinois Research and Test Reactor: Proposed Contents of PSAR Using NUREG-1537 Guidance for the Micro Modular Reactor (MMR®)", dated August 25, 2022 (ADAMS Accession Number ML22220A252).

1-3 U.S. Nuclear Regulatory Commission, "Staff Observations Regarding University of Illinois at Urbana-Champaign White Paper- Proposed Contents of Preliminary Safety Analysis Report (PSAR) Using NUREG-1537 Guidance for the Micro Modular Reactor", dated November 22, 2022 (ADAMS Accession Number ML22321A314).

1-4 U. of I. Topical Report IMRDD-MMR-22-04-A, Release 04, "University of Illinois Urbana-Champaign High-Temperature Gas-cooled Research Reactor: Applicability of Nuclear Regulatory Commission Regulations", dated July 30, 2024 (ADAMS Accession Number ML24208A143).

1-5 U.S. Nuclear Regulatory Commission SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water Reactors," Enclosures 1 and 2, September 2018.

1-6 U.S. Nuclear Regulatory Commission Regulatory Guide 1.232, Revision 0, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors", April 2018, (ADAMS Accession Number ML17325A611).

1-7 U. of I. Topical Report IMRDD-MMR-23-06-A, Release 02, "University of Illinois Urbana-Champaign High-Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria", dated July 30, 2024 (ADAMS Accession Number ML24208A066)

1-8 U. of I. Topical Report IMRDD-MMR-23-03-A, Release 02, "University of Illinois Urbana-Champaign High-Temperature Gas-cooled Research Reactor: Event Sequence Identification and SSC Safety Classification Methodology", dated March 20, 2024 (ADAMS Accession Number ML24075A193).

1-9 Idaho National Laboratory, INL/EXT-10-19329, "High Temperature Gas-Cooled Reactors Lessons Learned Applicable to the Next Generation Nuclear Plant", Revision 1, April 2011.

1-10 International Atomic Energy Agency, "Small Modular Reactors Catalogue – A supplement to IAEA Advanced Reactors Information System (ARIS)," 2024.

USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 2 - SITE CHARACTERISTICS

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
°R	degree Rankine
ADIP	Airport Data and Information Portal
ALOHA	Areal Locations of Hazardous Atmospheres
ASCE	American Society of Civil Engineers
AST	aboveground storage tank
bgs	below ground surface
BLEVE	boiling liquid expansion vapor explosion
bpf	blow per foot
BTU	british thermal units
CEUS	central and eastern U.S.
CFR	Code of Federal Regulations
cfs	cubic feet per second
CONUS	continental United States
DOE	U.S. Department of Energy
DOT	Department of Transportation
EPZ	Emergency Planning Zone
ESPA	Early Site Permit Application
FAA	Federal Aviation Administration
FEMA	Federal Emergency Management Agency
FIRM	FEMA’s Flood Insurance Rate Map
GMM	ground motion models
GSL	Geological Survey Laboratory
HHFT	high-hazard flammable train
HSRG	heat recovery steam generators
IDLH	Immediately Dangerous to Life or Health
IL	Illinois
ISG	Interim Staff Guidance
ISWS	Illinois State Water Survey
kg	kilograms
km	kilometers
kPa	kilopascal
kW	kilowatts
lbm	per pound mass
LEL	lower explosive limit
LFL	lower flammability limit
m^2	square meters
m^3	cubic meters

LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
MASW	Multichannel Analysis of Surface Waves
mb	millibars
mph	miles per hour
NAVD 88	North American Vertical Datum of 1988
NCDC	National Climatic Data Center
NGA	Next Generation Attenuation
NGVD	National Geodetic Vertical Datum
NIOSH	National Institute of Occupational Safety and Health
NMSZ	New Madrid Seismic Zone
NOAA	National Oceanic and Atmospheric Administration
NRC	Nuclear Regulatory Commission
NWS	National Weather Service
OSHA	Occupational Safety and Health Administration
PGA	peak ground acceleration
PMH	probable maximum hurricane
PMP	probable maximum precipitation
PMWP	Probable Maximum Winter Precipitation
PSAR	Preliminary Safety Analysis Report
psf	per square foot
PSHA	probabilistic seismic hazard analysis
psi	lb per square inch
psig	pounds per square inch gauge
RG	Regulatory Guide
SCPTu	Seismic Cone Penetration Test
SPT	Standard Penetration Tests
SRP	Standard Review Plan
SSAR	Site Safety Analysis Report
SSC	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
TNT	trinitrotoluene
Tsf	tons per square foot
TWA	time-weighted average
U. of I.	University of Illinois Urbana-Champaign
U.S.	United States
UEL	upper explosive limit
UFL	upper flammability limit
USCS	Unified Soil Classification System

LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
USGS	United States Geological Survey
V	victor
Vs	sheer-wave velocity
WARM	Water and Atmospheric Resources Monitoring Network
WVSZ	Wabash Valley Seismic Zone
ΔT	temperature gradient

CHAPTER 2 SITE CHARACTERISTICS

This chapter provides information regarding site location and description, including a discussion of the population in the vicinity of the site, the distribution of infrastructure and natural features, as well as the basis for selecting the University of Illinois (U. of I.) Urbana-Champaign campus for a new research reactor based on the KRONOS™ Micro Modular Reactor technology. U. of I. at Urbana-Champaign is a public land-grant research university established in 1867 (Reference 2-1).

In accordance with Title 10 of the Code of Federal Regulations (CFR) CFR Part 100.10, which outlines the factors relevant to site selection for test reactors, the collection and assessment of site data presented in this chapter have been conducted with these considerations in mind. However, as documented in the United States (U.S.) Nuclear Regulatory Commission (NRC)-approved U. of I. Topical Report, "University of Illinois Urbana-Champaign High-Temperature Gas-cooled Research Reactor: Applicability of Nuclear Regulatory Commission Regulations" (Reference 2-2), the requirements of 10 CFR Part 100 do not apply to the U. of I. research reactor.

However, in the absence of specific regulatory guidance pertaining to site characteristics for research reactors, the approach adopted for this Construction Permit Application focuses on demonstrating consistency with the principles outlined in 10 CFR Part 100, rather than strict compliance with its requirements.

The site characteristics evaluated in this assessment include:

- Geography and demography: an analysis of the geographical features and population distribution surrounding the site.
- Nearby industrial, transportation, and military installations: a review of the proximity and potential impact of industrial facilities, transportation networks, and military establishments.
- Meteorology: an examination of local climate and weather patterns that may influence reactor operations.
- Hydrology: an examination of local hydrologic, hydrogeologic, and solute transport risk conditions surrounding the site.
- Geology, seismology, and geotechnical engineering: a comprehensive study of the geological and seismic characteristics, along with the geotechnical properties of the site.

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Site Location and Description

2.1.1.1 Description

The U. of I. research reactor Licensed Area covers approximately 1.01 acres and is situated less than one mile from the downtown area in Champaign, Illinois (IL). It is located near the intersection of South Oak Street and East Gregory Drive, on the western edge of the U. of I. campus. Figure 2-1 shows the site's location within Champaign County and state.

Figure 2-2 illustrates the prominent natural and urban features within approximately 5 miles (8 kilometers [km]) of the project site. Table 2-1 summarizes the distance and direction from the site's center point to major nearby features (Reference 2-3).

Figure 2-3 illustrates the topography within the vicinity of the site. The finished site grade elevation will be approximately 732 feet (223.1 meters) North American Vertical Datum of 1988 (NAVD 88). The site and adjacent ground within a radius of approximately 0.5 miles (0.8 km) is flat. Topographic elevations range from approximately 764 feet (232.8 meters) NAVD 88, to approximately 678 feet (206.6 meters) NAVD 88 to the east of the site. Therefore, the topography within a 5-mile (8-km) radius ranges from approximately 54 feet (16.5 meters) below to approximately 32 feet (9.8 meters) above the site grade elevation (Reference 2-4).

Figure 2-4 shows the Licensed Area of the reactor facility. The area includes a paved parking lot, paved driveways and sidewalks, and landscaped spaces. The Licensed Area is located east of the Abbott Power Plant, a hybrid power plant facility utilizing both natural gas and coal. The Abbott Power Plant supports the University-owned and operated electrical and steam distribution system and serves the surrounding area. State Route IL-45, a significant corridor that provides essential access to the U. of I. campus and to neighboring communities lies to the west of the Abbott Power Plant (Reference 2-5).

The Licensed Area is adjacent to the Geological Survey Laboratory (GSL) building (1116 South Oak Street) on three sides. However, the GSL is not included in the project scope as there are no current plans to incorporate the existing building into the research reactor facility. The U. of I. Personnel Services building (52 East Gregory Drive) is located east of the Licensed Area. Directly across Gregory Drive from the GSL and the Personnel Services building to the south are outdoor basketball and the U. of I. Campus Bike Center (Reference 2-5).

Historical aerial photos, dating back to 1940, indicate no prior development on the project site apart from the parking lot that currently occupies the area much of the Licensed Area. By 1969, the U. of I. Personnel Services building, the Abbott Power Plant, and the cooling towers located on the adjoining northern property were present. The area has remained relatively unchanged since then.

The project site is presently zoned "Multifamily 3 – High Density Multifamily Residential/Limited Business". This designation permits a blend of high-density multifamily housing, college-oriented dwellings, office space, and mixed-use buildings – land uses typically located in close proximity to major universities and colleges (Reference 2-6).

Immediately north of East Armory Avenue, land use classifications shift to predominantly residential areas identified as “University Neighborhood” on the City of Champaign’s Future Land Use Map. These neighborhoods are characterized chiefly by multi-family apartment complexes, though dormitories, fraternity or sorority houses, and other group-living facilities may also be present. Residents in this district enjoy convenient access to Campustown, Downtown, Midtown, and various U. of I. facilities via frequent public transit and a well-connected street grid that encourages walking and bicycling (Reference 2-7).

The environs of the proposed project site exhibit mixed land use, combining residential pockets, light industrial development, and a range of university-owned facilities – such as administrative offices and educational laboratories. U. of I. maintains an extensive recreational portfolio that encompasses campus facilities with over 450,000 square feet of indoor facilities distributed across six distinct venues. In addition to this substantial indoor space, the university also offers approximately 19 acres of outdoor playfields, providing a diverse range of recreational opportunities for students and the community (Reference 2-8).

Among the key amenities available at U. of I. are two state-of-the-art recreation centers. These centers are equipped with advanced aquatics complexes, dedicated fitness zones, and equipment rentals, as well as multi-activity courts designed for a variety of sports, including indoor soccer, hockey, racquetball and running tracks.

Furthermore, the university features specialized venues such as the Campus Bike Center, which promotes cycling as a sustainable mode of transportation, and a full-service Ice Arena. The outdoor recreational offerings include a well-developed network of complex fields and courts designed for basketball, tennis, baseball, soccer and volleyball. These outdoor facilities are complemented by picnic areas and open-air pavilions.

2.1.1.2 Specification and Location

The proposed Licensed Area encompasses approximately 1.01 acres and is situated within the Illinois/Indiana Prairie ecoregion. The designated reactor center point is approximately 40° 06’ 16” north latitude and -88° 14’ 28” west longitude. In metric coordinates, this location corresponds to Universal Transverse Mercator Zone 16N – Northing 381,613.40 m and Easting 394,210.04 m – as well as Illinois State Plane (East Zone) coordinates of Northing 307,853.54 m and Easting 381,613.40 m.

The Illinois/Indiana Prairie ecoregion is a landscape shaped by glaciation that occurred more than 12,500 years ago. It is typified by extensive flat to gently rolling plains underlain by deep, dark, fertile soil, and supported by a comparatively long growing season. Historically dominated by tall-grass prairies, the region has experienced almost complete conversion to intensive agriculture. Within the South Farms sector of the U. of I. campus, corn and soybeans constitute the principal crops, while cattle, sheep, poultry, and hogs are also raised. Such agricultural activity has measurably influenced local stream chemistry, increased turbidity levels, and altered aquatic habitats (Reference 2-9).

Across the broader U. of I. campus the prevailing landform is notably flat; though subtle undulations emerge around basins and drainage corridors (Reference 2-1). This flat, agrarian character extends throughout Champaign County, which spans approximately 998 square miles (2,585 square km) and ranges in elevations from about 620 to 860 feet (189 to 262 meters) above mean sea level (Reference 2-10).

Within a 50-mile (80-km) radius of the project site, several metropolitan areas are notable for their proximity and population. Decatur, IL, located approximately 43.2 miles (69.5 km) southwest of the site in Macon County, has a population of 70,522. Bloomington, IL, situated roughly 47.7 miles (76.8 km) northwest in McLean County, has a population of 76,680. Additionally, Danville, IL, lies 31.9 miles (51.3 km) to the west in Vermilion County and has a population of 29,204. While outside the 50-mile radius, Springfield, IL, located 86.1 miles (189 km) to the east in Sangamon County, is a significant metropolitan area with a population of 114,394, providing an additional regional context (Reference 2-11).

2.1.1.3 Boundary and Zone Area Maps

The research reactor will be housed entirely within the Reactor Facility building, which is illustrated on Figure 2-4. The same figure also delineates the Licensed Area and Site Boundaries. The Site Boundary, which in accordance with 10 CFR Part 20.1003, encompasses land that is owned, leased, or otherwise controlled by the licensee (U. of I.), extends beyond the Licensed Area.

The Emergency Planning Zone (EPZ) is established within the Site Boundary, as shown on Figure 2-4. In the event of an incident, all emergency measures will be confined to this EPZ. Radiation doses at the EPZ perimeter are projected to remain within the limits prescribed by the Environmental Protection Agency Protective Action Guides, consistent with recommendations contained in ANSI/ANS-15.16-2015, "Emergency Planning for Research Reactors," (R2024) and the guidance set forth in Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors." This configuration aligns with the provision in 10 CFR Part 50, Appendix E, Section I.3, which permits a reduced-radius EPZ for research and test reactors.

2.1.2 Population Distribution

This section provides population distribution data for resident and transient populations for the area within 5 miles (8 km) of the center point of the site for the following years (Reference 2-12, Reference 2-13):

- Most recent Decennial Census (2020)
- Beginning of the requested license period (approx. 2030)
- Five years after the beginning of the requested license period (approx. 2035)
- Approximate end of the requested license period (approx. 2070)

Estimates and projections of resident and transient populations around the site are divided into five distance bands (represented by concentric circles). The distances from the center point of the reactor are: 0 to 0.5 miles (0 to 0.8 km), 0.5 to 1 mile (0.8 to 1.6 km), 1 to 2 miles (1.6 to 3.2 km), 2 to 3 miles (3.2 to 4.8 km), and 3 to 5 miles (4.8 to 8 km). The distance bands are further subdivided into 16 directional sectors, each centered on one of the 16 compass directions and consisting of 22.5 degrees. For each segment formed by the distance bands and directional sectors, the resident population was estimated using the most recent and currently available decennial census year (2020) (Reference 2-12). The population data is used for environmental monitoring as part of the Radiation Protection Program discussed in Chapter 11.

Figure 2-3 identifies the nearest full-time residence, situated approximately 0.1 mile (0.16 km) northwest of the reactor at 1014 South Oak Street, as well as the closest part-time residence, located at 58 East Gregory Drive.

2.1.2.1 Resident Population

Resident-population characteristics within a 5-mile (8-km) radius of the reactor site are depicted in Figure 2-5 through Figure 2-8, which also delineate relevant town, city, and county boundaries.

For Figure 2-5, 2020 decennial-census block data – the smallest spatial resolution available from the U.S. Census Bureau (Reference 2-12) - were allocated to five concentric distance bands surrounding the site. Each band was further subdivided into four quadrants (NE, SE, SW, NW). Using ArcMap10.5, population density was calculated for every census block that intersects the 5-mile study area. The proportion of each block's land area lying within a given distance-and-direction segment was then determined, and an equivalent share of that block's population was assigned to the segment. Where multiple blocks overlapped a segment, their respective proportional populations were summed to yield the total estimate for that segment.

Figure 2-6, Figure 2-7, and Figure 2-8 present projected populations for the years 2030 (approximate initial license year), 2035 (5 years later), and 2070 (nominal license end). Currently available county projections were based on the 2020 decennial-census block data. It was determined that the best forecasting method would be to perform linear regression using 5 years of historical decennial population results to forecast future decennial periods, then interpolate the results to arrive at intercensal periods (Reference 2-14). The growth rate of the County was derived then applied to the base year of 2020 for each quadrant, which was the most recent decennial census data available, and projected forward for the years 2030, 2035, and 2070. The same rates were used to project population changes in each distance/direction segment in each county.

Table 2-2 shows the historical population for 2020 and the projected resident population for the years 2030, 2035 and 2070 that fall within the distance bands for Champaign County (Reference 2-12, Reference 2-14).

2.1.2.2 Transient Population

The U. of I. campus and Parkland College lie within 5 miles (8 km) of the site. Transient populations are temporary or seasonal populations residing in the area, such as in lodging accommodations, dormitories or classrooms on a college campus. The U.S. Census Bureau also tracks other transient populations, such as individuals residing in correctional facilities, nursing facilities, military quarters and noninstitutional facilities. However, in the decennial 2020 census, college students who live in college/university student housing were counted at the student housing rather than at their parents' or guardians' home (Reference 2-15). According to the 2020 decennial census, 13,255 students lived in college or university student housing in the County (Reference 2-13). Those students were included in permanent population figures.

Table 2-3 shows the historical transient population for 2020 and the projected transient population for the years 2030, 2035 and 2070 that fall within the distance bands for Champaign County (Reference 2-13, Reference 2-14).

Figure 2-9 shows the transient population as of the 2020 decennial census distributed into five distance bands based on distance from the center point of the reactor. Population estimates within each quadrant and band were derived from block data, a smaller geographic unit than block groups, also from the 2020 decennial census (Reference 2-13). To determine the population within each quadrant and band, a population density was calculated for every block within the 5-mile radius. The population was re-calculated based on the area within the quadrants and bands. For each segment formed by the distance bands and directional sectors, the percentage of each block area that falls, either partially or entirely, within that segment was calculated using the ArcMap10.5. The equivalent proportion of each block's population was then assigned to that segment. If portions of two or more blocks fall within the same segment, the proportional population estimates for the blocks were summed to obtain the population estimate for that segment.

Figure 2-10, Figure 2-11, and Figure 2-12 show the transient population estimates for 2030 (approximate beginning of the license period); 2035 (5 years later), and 2070 (approximate end of the license period). Currently available county projections were based on the 2010 census and thus were too dated to use for the purposes of this section. According to the Illinois State demographer, an updated projection set is currently unavailable due to necessary data from the 2020 census that are not yet released. It was determined that the best forecasting method would be to perform linear regression using 5 years of historical decennial population results to forecast future decennial periods, then interpolate the results to arrive at intercensal periods (Reference 2-14). The growth rate of the County was derived then applied to the base year of 2020 for each quadrant, which was the most recent decennial census data available, and projected forward for the years 2030, 2035, and 2070. The same rates were used to project population changes in each distance/direction segment in each county.

Table 2-1 Distances & Directions to Prominent Features

Feature	Distance from Site (mile)	Direction from Site	Notes
Abbott Power Plant	0.05	W	Located adjacent, east of the site
Abbott Power Plant Cooling Towers	0.01	W	Located adjacent to the site
United States Highway Route 45	0.1	W	Major roadway serving the area
United States Highway Route 150	1.1	N	Regional highway, major East-West route north of campus
Illinois Highway Route 10	0.6	N	State highway, principal East-West arterial through campus
Illinois Highway Route 130	3.2	E	State highway, runs North-South on Urbana's east side
United States Interstate Route 57	3.4	W	Interstate highway, North-South interstate west of city
United States Interstate Route 72	2.3	SW	Interstate highway, joins Interstate 57 south-west of site
United States Interstate Route 74	2.1	NW	Interstate highway, beltway north-west of campus
Canadian National North-South Railroad	0.1	W	Rail corridor, adjacent to Highway 45 corridor
Canadian National West-East Railroad	1.8	N	Rail corridor, passes through downtown Champaign
Norfolk Southern East-West Railroad	1.2	S	Rail corridor, tracks paralleling Windsor Road
Frasca Field	3.7	NW	Local general-aviation airfield
Willard Airport	4.8	S	Regional commercial airport near Savoy
Embarras River	1.6	E	Natural feature, meanders south of Urbana
Boneyard Creek	0.5	NE	Natural feature, urban stream through U. of I.
Phinney Branch	2.2	E	Natural feature, tributary of Embarras River

Table 2-2 Resident Population Distribution within 5 miles (8 km) of the Site in Champaign County

Year	Distance Band (miles)					
	0-0.5	0.5-1	1-2	2-3	3-5	Total
2020	15,434	15,030	37,977	39,899	39,911	148,251
2030	15,791	31,168	70,023	110,845	151,679	379,506
2035	16,701	32,966	74,061	117,237	160,425	401,390
2070	24,262	47,889	107,588	170,309	233,049	583,097

Sources: Reference 2-12, Reference 2-14

Table 2-3 Transient Population Distribution within 5 miles (8 km) of the Site in Champaign County

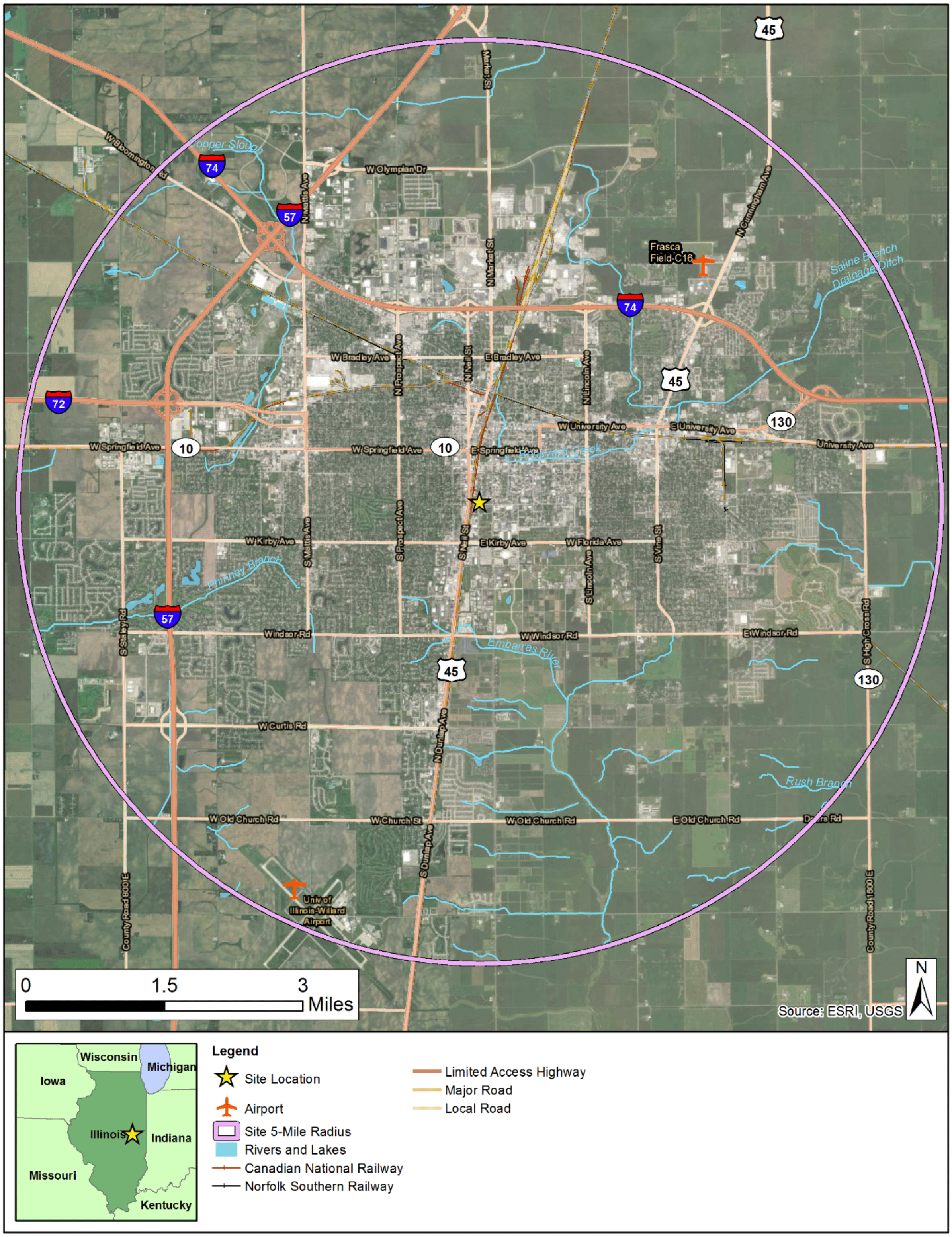
Year	Distance Band (miles)					
	0-0.5	0.5-1	1-2	2-3	3-5	Total
2020	5,825	7,570	8,113	4,280	2,957	28,745
2030	5,854	7,607	8,153	4,301	2,972	28,887
2035	5,868	7,626	8,173	4,312	2,979	28,958
2070	5,969	7,757	8,314	4,386	3,030	29,457

Sources: Reference 2-13, Reference 2-16

Figure 2-1 Location of the Site

Source: Reference 2-3

Figure 2-2 Prominent Features in Site Area



Source: Reference 2-3

Figure 2-3 Topography in the Site Vicinity and Nearest Residents

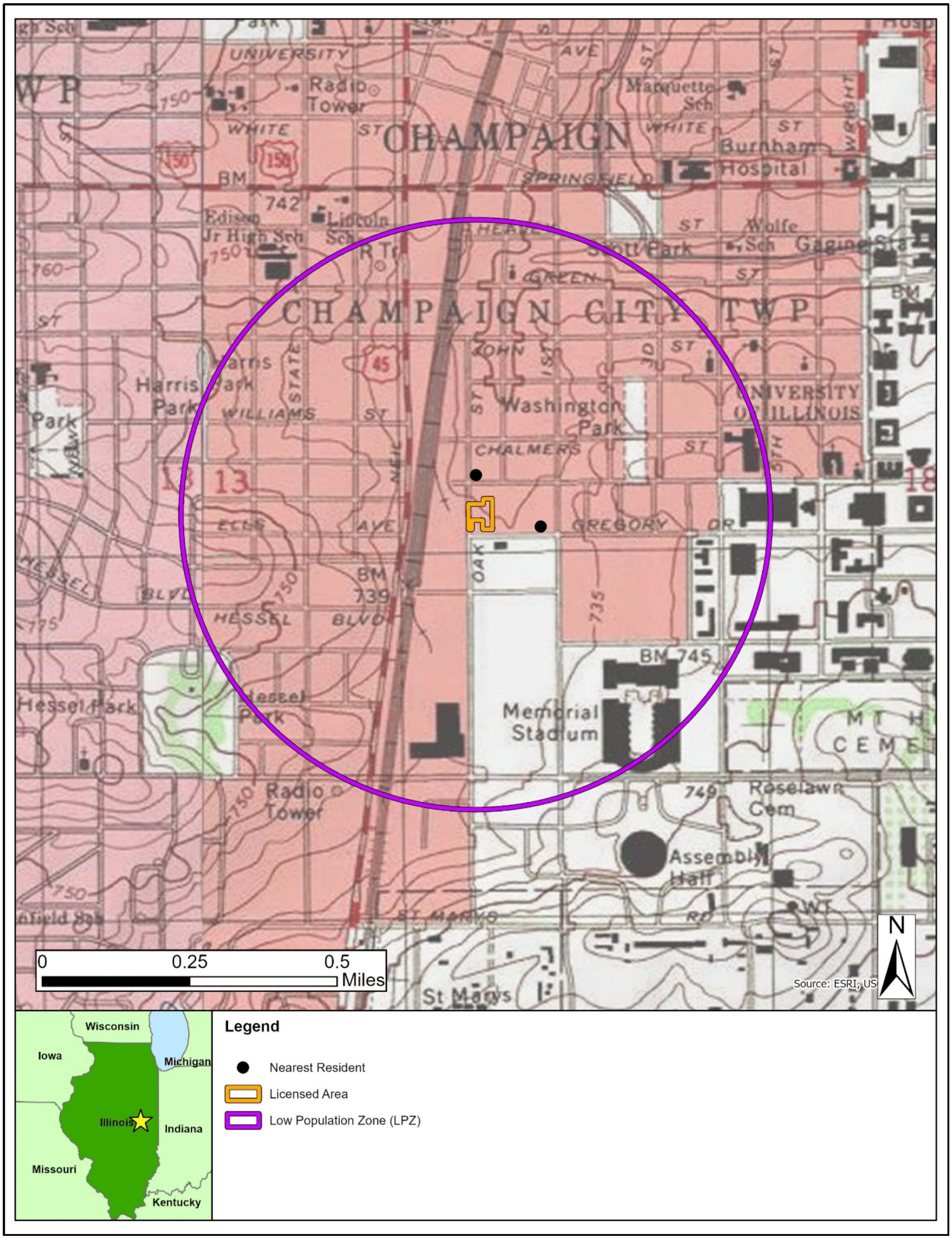


Figure 2-4 U. of I. Research Reactor Site Boundary, Licensed Area, and Facility Layout

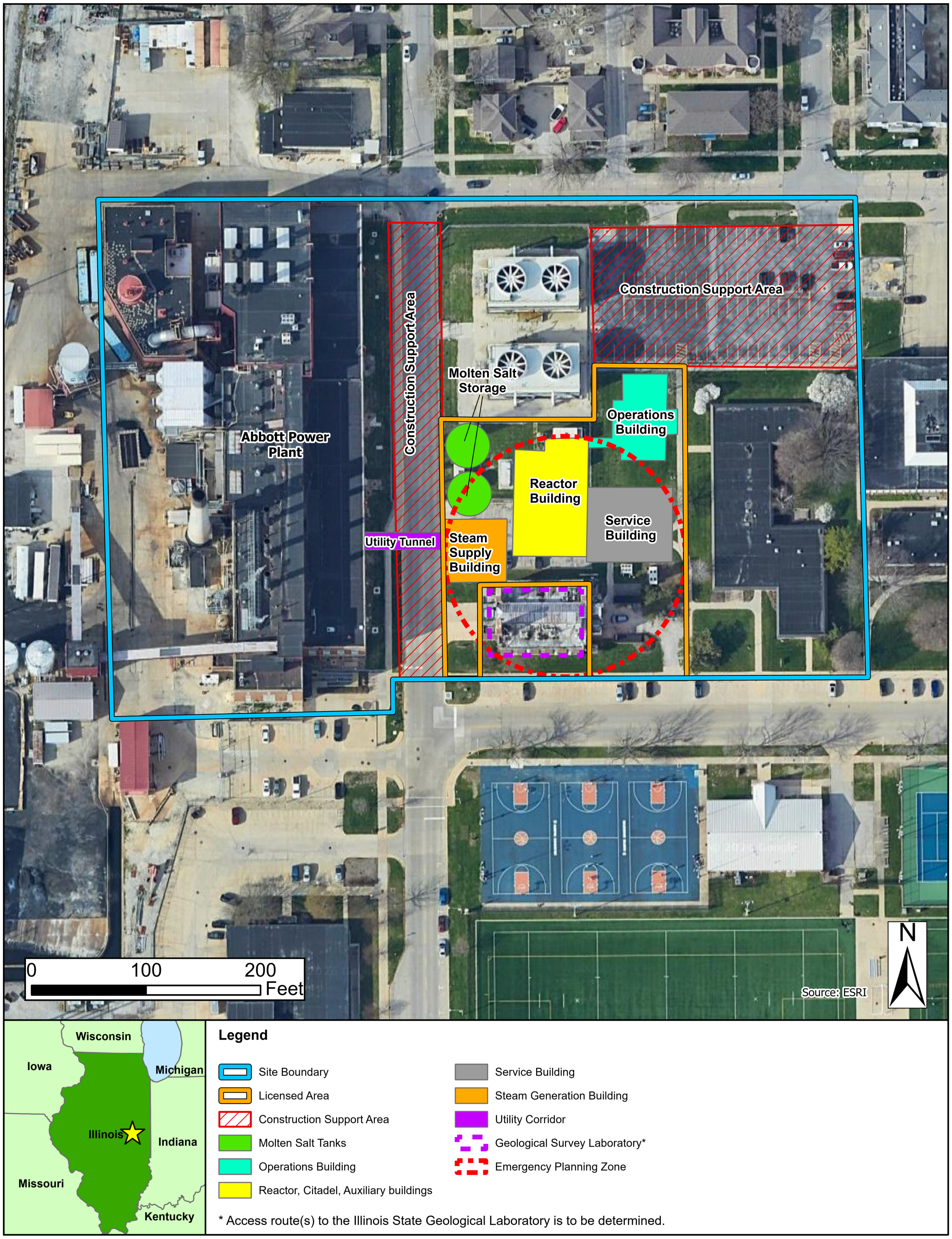
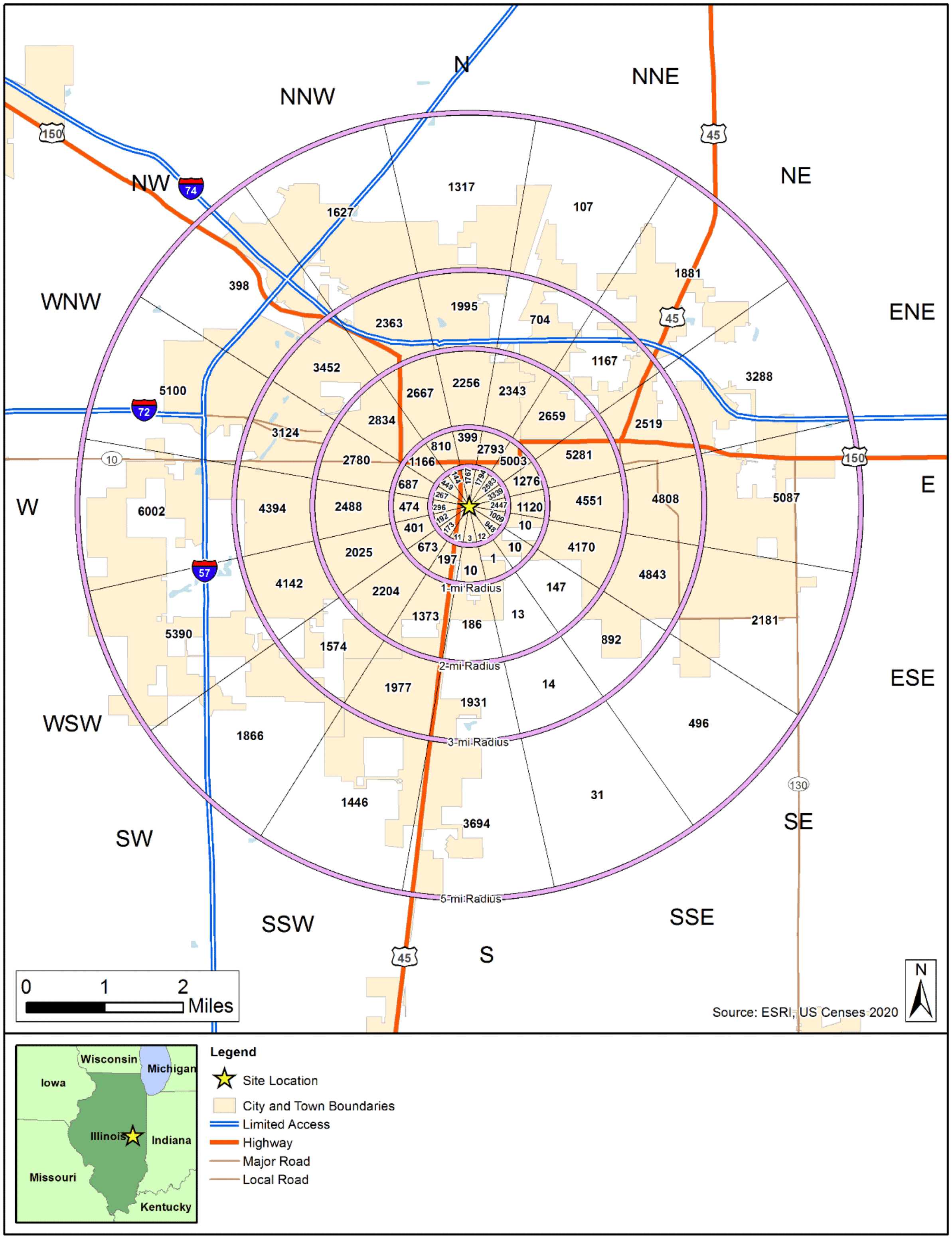
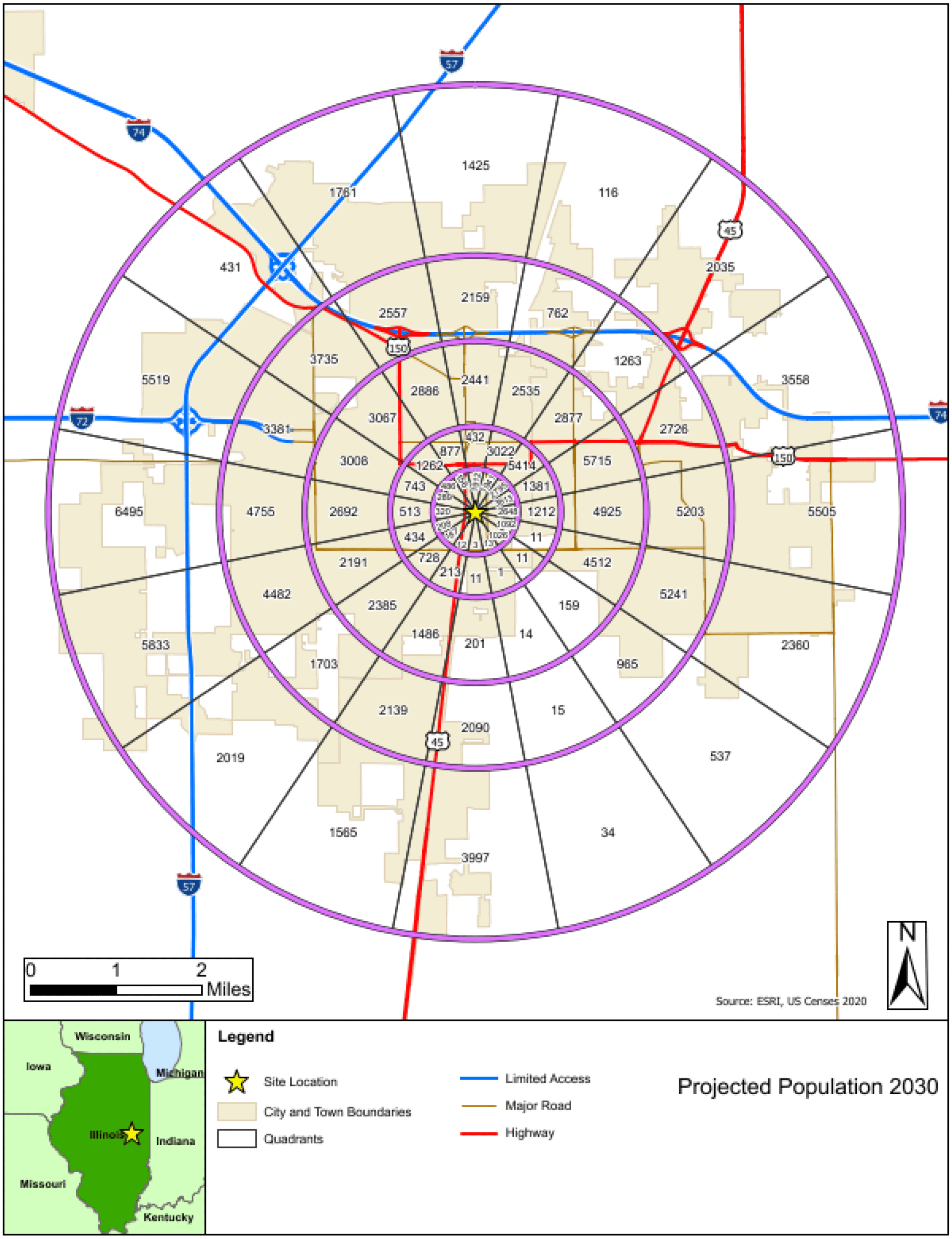


Figure 2-5 Resident Population Distribution – 2020



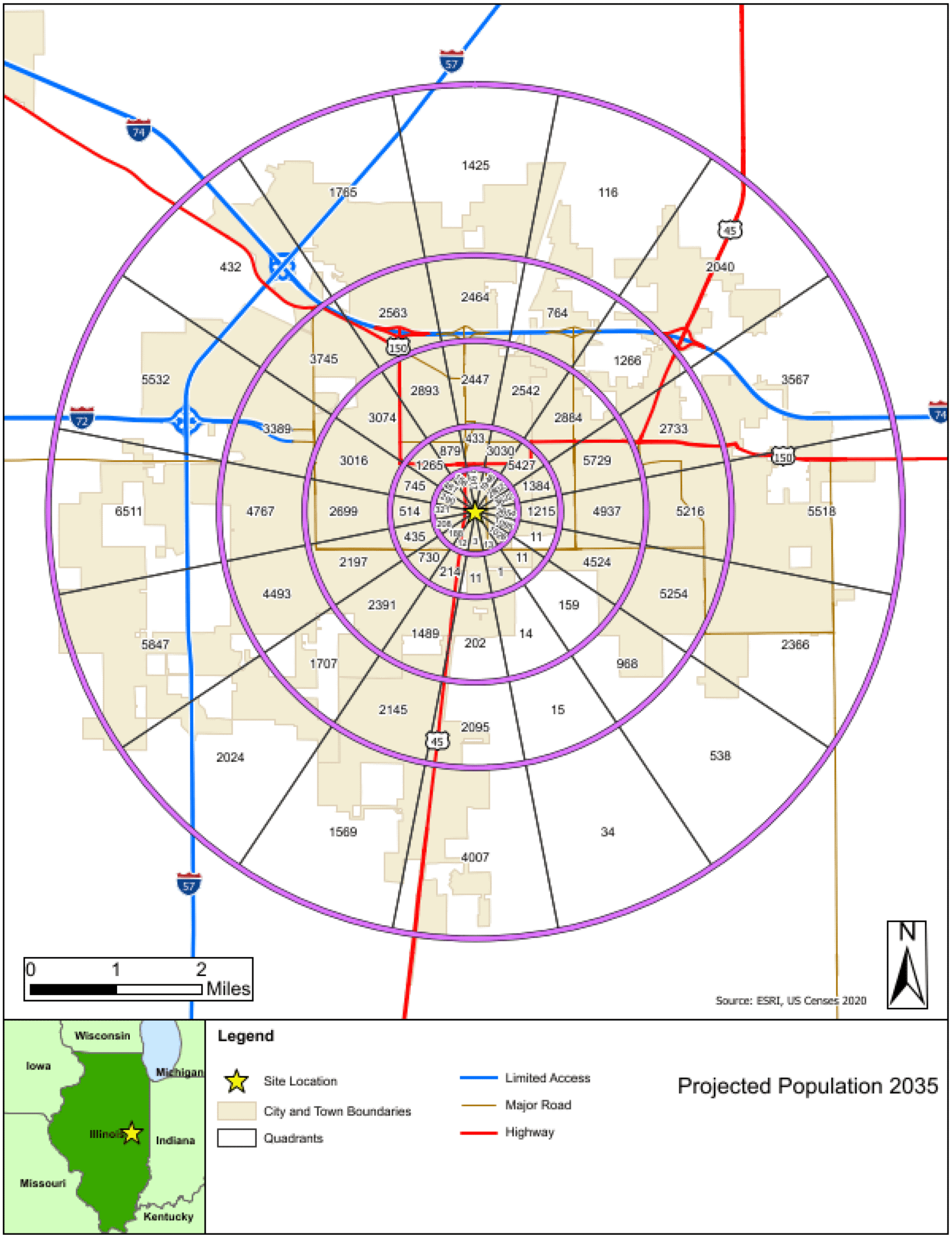
Source: Reference 2-3, Reference 2-12

Figure 2-6 Estimated Resident Population Distribution – 2030



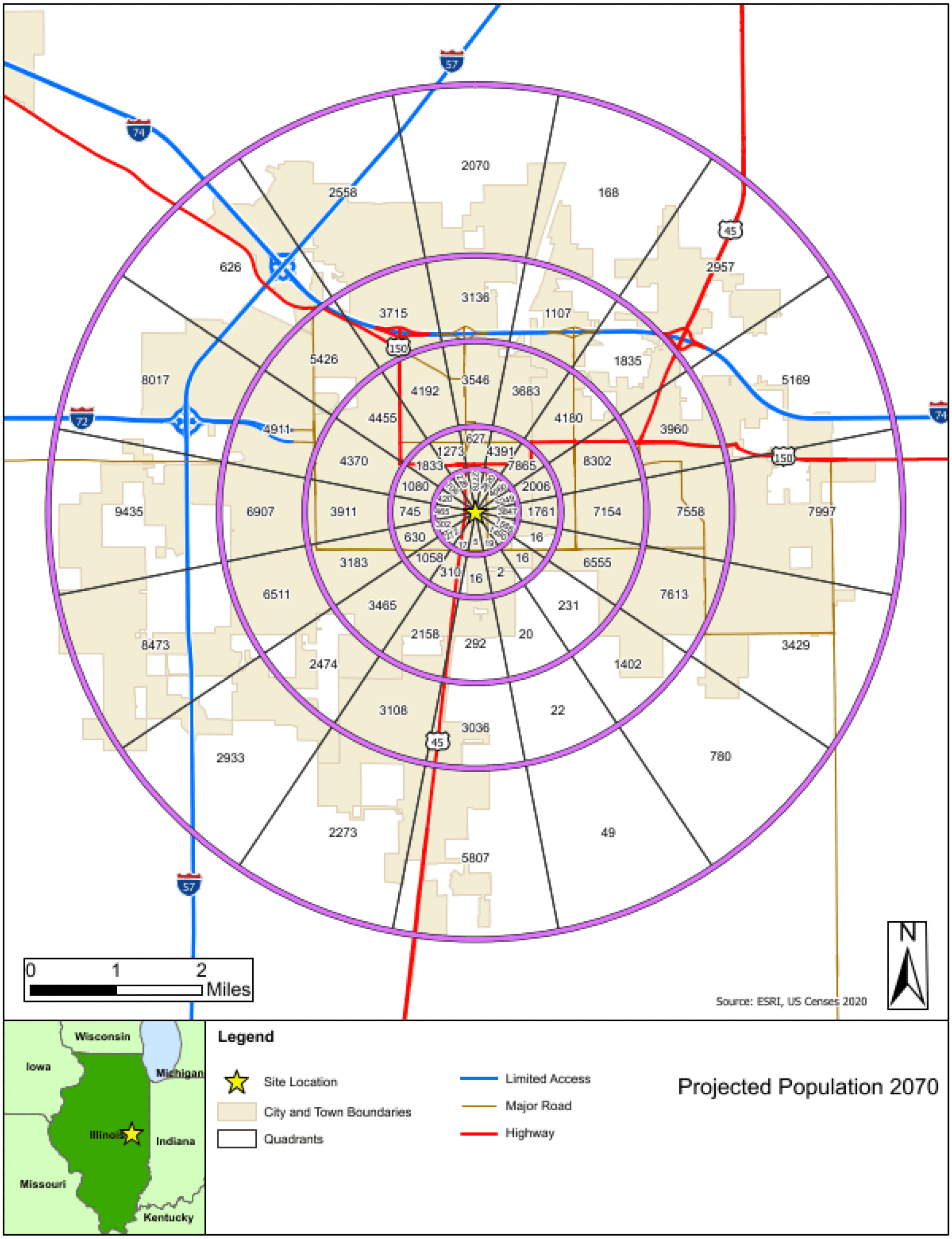
Source: Reference 2-3, Reference 2-12, Reference 2-14

Figure 2-7 Estimated Resident Population Distribution – 2035



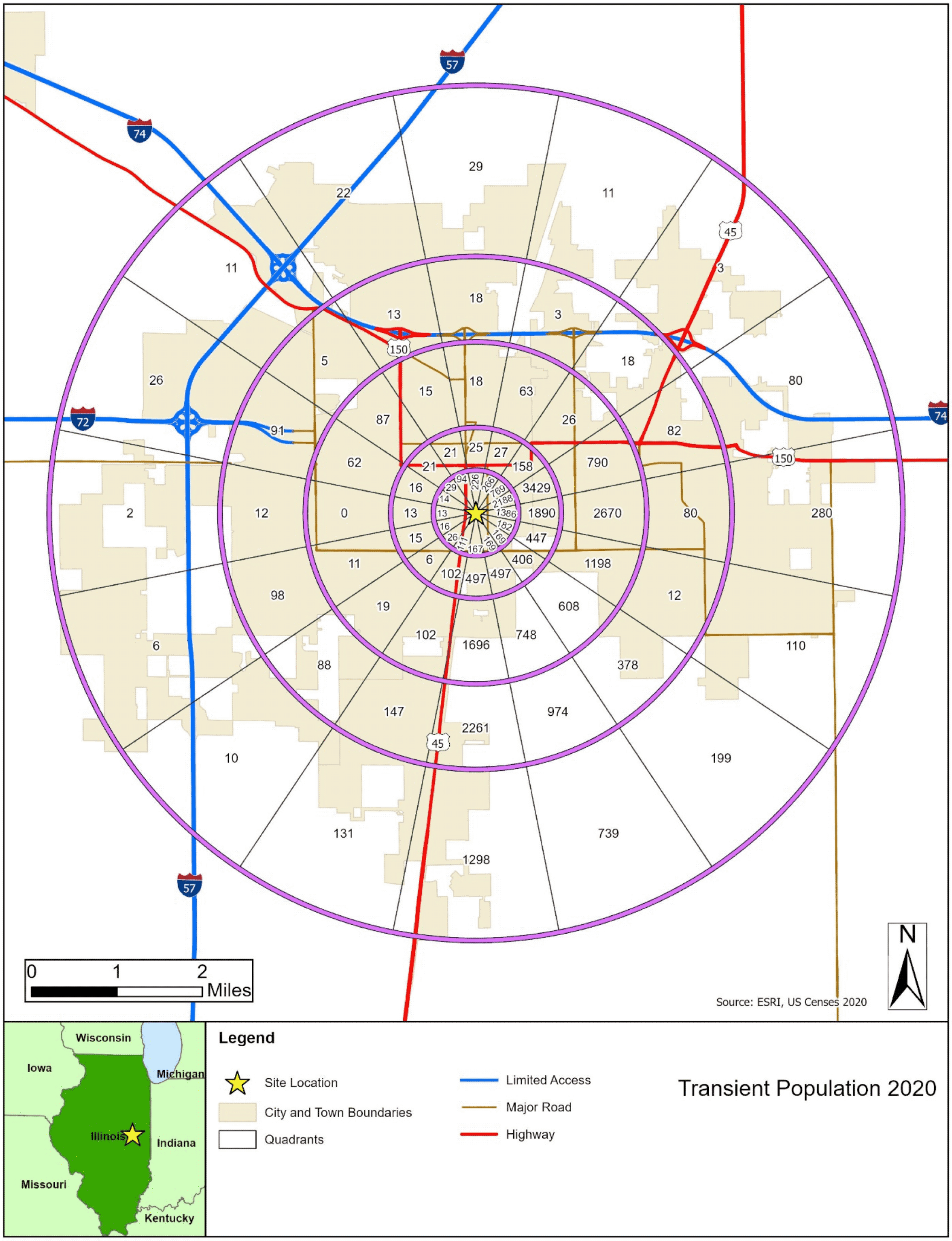
Sources: Reference 2-12, Reference 2-14

Figure 2-8 Estimated Resident Population Distribution – 2070



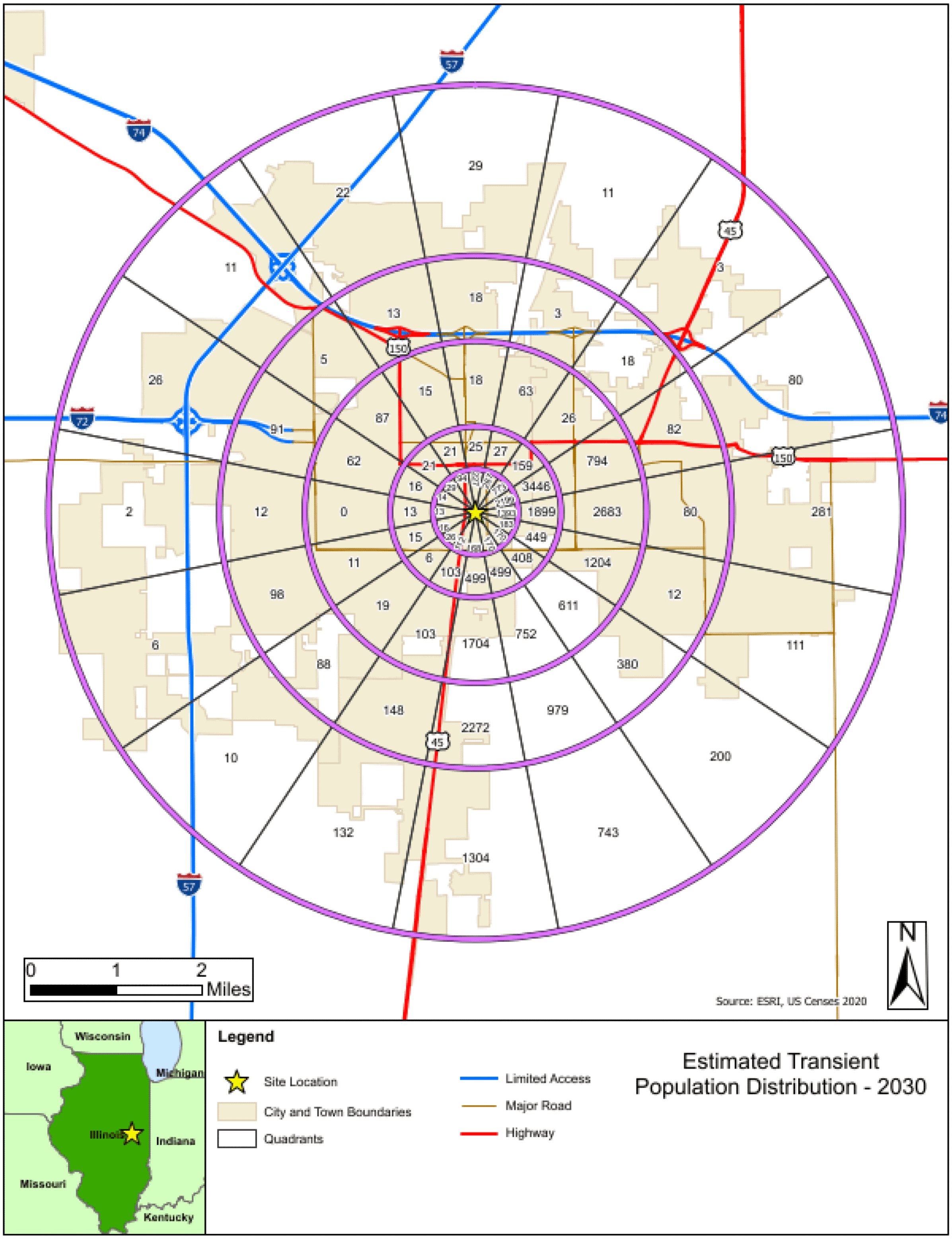
Sources: Reference 2-12, Reference 2-14

Figure 2-9 Transient Population Distribution – 2020



Source: Reference 2-13

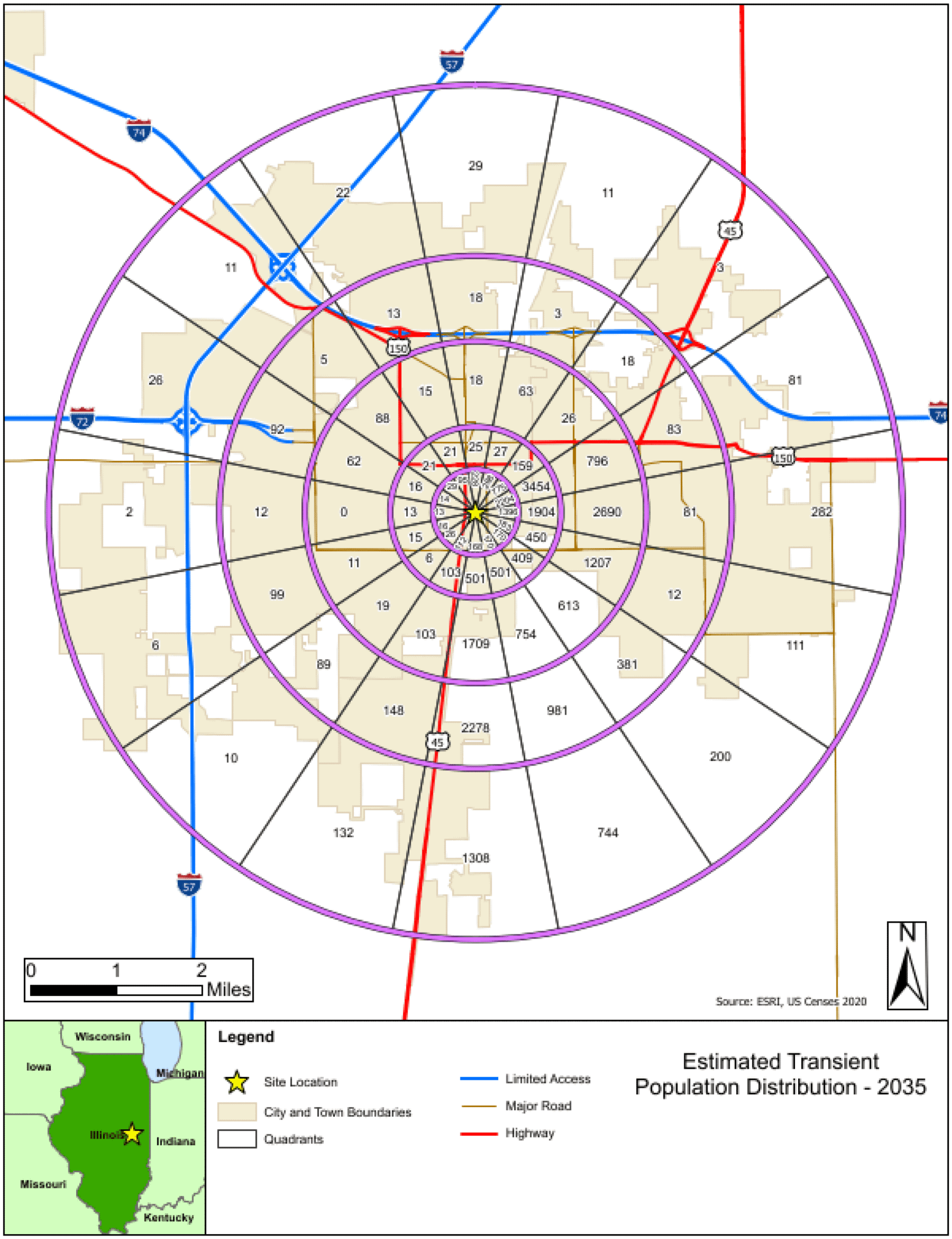
Figure 2-10 Estimated Transient Population Distribution – 2030



Sources: Reference 2-13, Reference 2-16

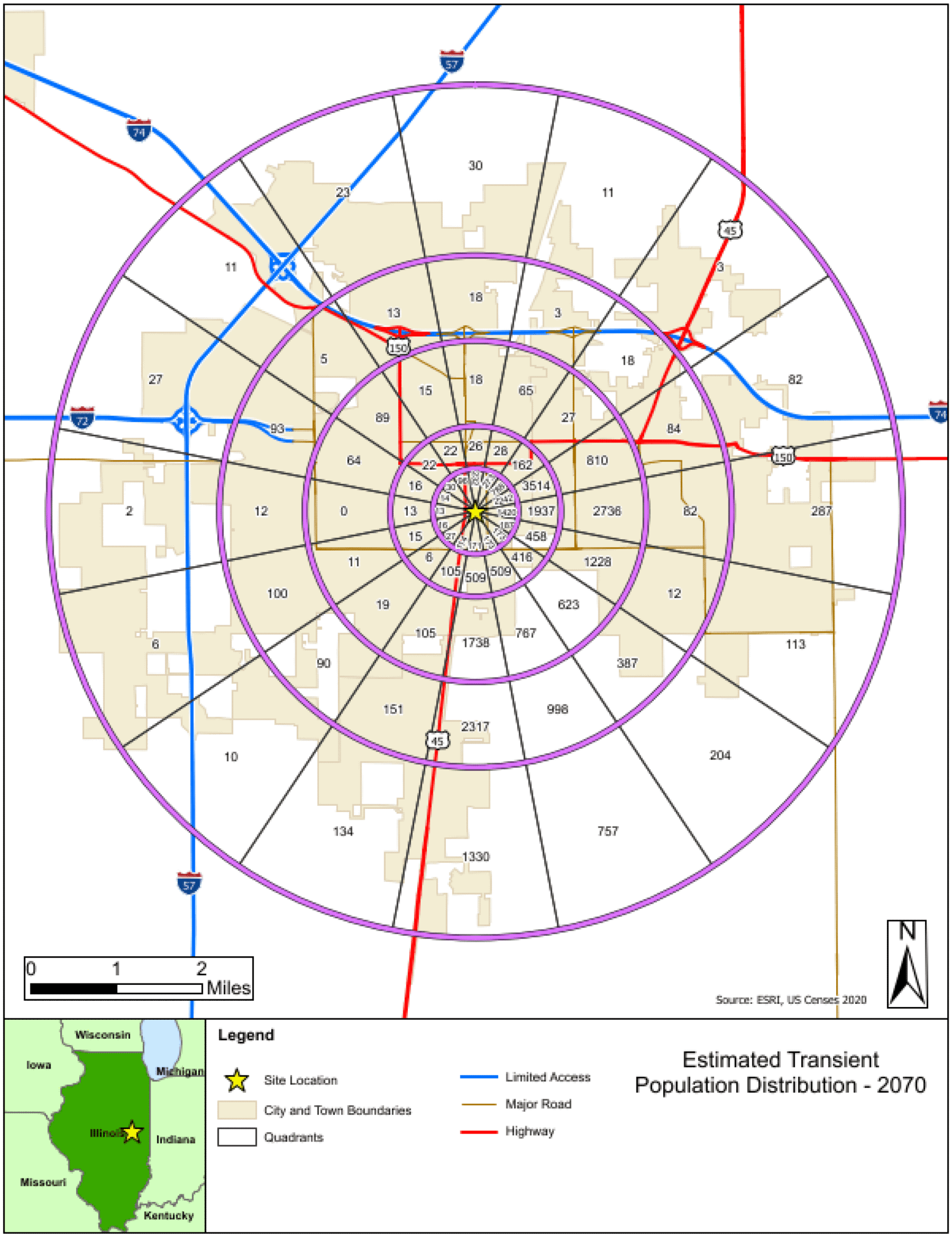
UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN

Figure 2-11 Estimated Transient Population Distribution – 2035



Sources: Reference 2-13, Reference 2-16

Figure 2-12 Estimated Transient Population Distribution – 2070



Sources: Reference 2-13, Reference 2-16

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY INSTALLATIONS

This section identifies and evaluates present and projected future federal, industrial, transportation, and residential facilities and operations in the vicinity of U. of I. research reactor facility. Consistent with the guidance outlined in NUREG-1537, the evaluation focuses on facilities and activities within a 5-mile (8-km) radius of the project site. While facilities and activities beyond this range were also reviewed, they were not analyzed in detail due to their negligible relevance to potential impact on the research facility.

2.2.1 Locations and Routes

Figure 2-13 shows the location of nearby facilities, including industrial and transportation facilities within 5 miles (8 km) of the site. Figure 2-14 illustrates the airports, jet routes, and airway routes identified within 10 miles (16 km) of the site.

The evaluation of transportation routes within the 5-mile (8-km) vicinity of the site identified four United States and state highways, three interstate highways, and three rail lines shown in Figure 2-13. The identified transportation features include:

- US Highway 45 (US-45)
- US Highway 150 (US-150)
- Illinois State Highway 10 (IL-10)
- Illinois State Highway 130 (IL-130)
- Interstate Highway 57 (I-57)
- Interstate Highway 72 (I-72)
- Interstate Highway 74 (I-74)
- Canadian National Railway
    - North-South Line
    - East-West Line
- Norfolk Southern Railway
    - East-West Line

There are three wastewater treatment plants, one chemical manufacturing facility, one natural gas power plant, multiple gasoline stations, and multiple educational and research laboratories within 5 miles (8 km) of the site. The nearest gasoline station and research laboratory are shown on Figure 2-13. The numerous university research laboratories and support facilities around the U. of I. campus that possess and use small quantities of hazardous and potential hazardous substances are not considered as credible sources of accidents that could have an impact at the project site.

There are natural gas pipelines and hazardous liquid pipelines within 5 miles (8 km) of the site. Other existing features that require consideration with respect to possible adverse effects on the reactor are identified in Figure 2-13 and Figure 2-14. Table 2-4 provides a description of these features and facilities, including their primary functions and industrial hazards and/or hazardous materials stored on site. In addition, an analysis of potential impacts to the U. of I. research reactor facility due to use and storage of hazardous materials both on and off the site is presented in Section 2.2.3.

2.2.1.1 Description of Pipelines

U. of I. is served by two natural gas distribution networks, one operated by Ameren, IL and the other by the university itself. In general, Ameren serves university buildings situated north of Florida Avenue/West Kirby Avenue, mostly for laboratory purposes. The university's system generally serves buildings south of Florida Avenue/West Kirby Avenue, but also the Abbott Power Plant (Reference 2-1).

The Abbott Power Plant has a nameplate capacity of 84 megawatts of electricity (Reference 2-17) and consumed over 3.1 million cubic feet (87,800 cubic meters) of natural gas in 2024 (Reference 2-18). The plant contains six boilers, two heat recovery steam generators (HRSGs), two gas turbines, and nine steam turbine generators. The three natural gas fired boilers are each capable of producing up to 175,000 pounds per hour (lb/hr) (79,400 kilograms per hour [kg/hr]). The three coal boilers collectively can produce up to 350,000 lb/hr (159,000 kg/hr). The two HRSGs work in conjunction with the plant's two natural gas turbines and generate approximately 40,000 lb/hr (18,100 kg/hr) from their flue gases. The ductwork between the gas turbine exhaust and the HRSG also is equipped with natural gas-fired duct burners. When these burners are used, the output from the HRSGs increases from 40,000 lb/hr up to 110,000 lb/hr (18,100 kg/hr to 49,900 kg/hr). The nine steam turbines have two exhaust modes: condensing and extraction. In extraction mode, steam is exhausted from the turbine and diverted to the campus system for heating (Reference 2-17).

The university's natural gas system is supplied by a transmission pipeline, which starts at Kinder Morgan's Natural Gas Pipeline Co. of America LLC pipeline meter station, north of Monticello, Illinois (Reference 2-17) and is routed to a regulator station at Curtis Road and the Canadian National Railroad tracks in Champaign (Reference 2-1) (see intersection of east-west and north-south gas pipelines near US-45 on Figure 2-13). The transmission pipeline operates at 700 pounds per square inch gauge (psig) (858 psig maximum operating pressure) and the regulator station at Curtis Road (2.5 miles [4.0 km] south-southwest of the reactor site) reduces the pressure to 400 psig (Reference 2-17). The 400-psig gas is routed to the Abbott Power Plant (Reference 2-1).

The 700-psig pipeline segment is operated by U. of I., is 8 inches (20.3 centimeters [cm]) in diameter, and is approximately 19 miles in length (Reference 2-17). The 400-psig segment of the pipeline starts at the Curtis Road pressure reduction station and runs to a gas regulator yard at Abbott Power Plant, where a portion of the gas is sent to the existing gas turbine generators (Reference 2-17). This U. of I. operated segment is 10 inches (25.4 cm) in diameter, approximately 2.7 miles (4.3 km) long, and has a maximum operating pressure of 618 psig (Reference 2-17). Another portion of the gas delivered to the Abbott Power Plant is heated, odorized, and reduced to 40 psig for distribution to the boilers and campus buildings (Reference 2-19). There are other smaller (4-inch [10.2 cm] and below) low-pressure natural gas lines west (east side of S. Oak Street), north (north side of Armory Avenue), east (east of the Personnel Services building, Building 154), and south (along Gregory Drive) of the project site. There is 2-inch line that feeds low-pressure natural gas to the Geological Survey Laboratory. The 4-inch natural gas line along S. Oak Street distributes natural gas to the residential area north of the project site.

As stated in the previous paragraph, excluding the campus-wide low-pressure pipeline network, there are two primary natural gas pipelines within 5 miles (8 km) of the U. of I. research reactor site. The 8-inch (20.3-cm) diameter 700 psig pipeline runs east-west and connects with the north-south 10-inch (24.4-cm) diameter pipeline 2.5 miles (4.0 km) south of the research reactor. The 8-inch (23.4 cm) 700 psig pipeline is closest to the reactor site at this junction. The 10-inch (25.4-cm) 400 psig pipeline enters the Abbott Power Plant on the south side of the building approximately 200 feet (61 meters) from the reactor facility's Licensed Area. Figure 2-13 illustrates the natural gas pipelines located within 5 miles (8 km) of the site, and Table 2-5 summarizes the description and operating parameters of the pipelines. These pipelines are evaluated further as hazards in Section 2.2.3.

U. of I. is currently studying and designing a Compressed Natural Gas Energy Storage system that will utilize the existing 700 psig and 400 psig pipelines (Reference 2-17). As the implementation of this project will not increase the pressure in either pipeline (Reference 2-17), it is not expected to have an impact on the analyses presented in Section 2.2.3.

Marathon Pipeline operates a 12-inch (30.5-cm) petroleum products pipeline located approximately 4.5 miles (7.2 km) west of the research reactor site project. Due to its distance from the site and the proximity of the natural gas pipelines, this pipeline was not evaluated as a potential explosion or other hazard. Additionally, fuel oil is piped to the Abbott Power Plant from large above-ground storage tanks located north of Kirby Avenue through a 6-inch fuel oil pipeline located in the steam tunnel along South Oak Street (see Section 2.2.1.5). This 6-inch (15.2 cm) line was also not evaluated as an explosion or other hazard.

2.2.1.2 Description of Waterways

Three rivers and/or creeks were identified within 5 miles (8 km) of the site; none are considered to be major waterways. Therefore, there is no movement of hazardous materials via waterborne cargo (e.g., chemicals and related products, petroleum, ordnance) that could pose a threat to operations at the site. As such, waterborne shipping is not evaluated further with respect to accidents and impacts on waterways and does not warrant further consideration in determining bounding accident scenarios involving transport of hazardous materials near the site.

2.2.1.3 Description of Highways

A network of roadways traverses Champaign County, making it a statewide transportation hub. There are three major interstates surrounding the U. of I. campus that include I-57, I-72, and I-74. The latter two interstates primarily run east and west just north of Champaign, while I-57 runs north and south and connects I-74 and I-72 on the west side of the city. I-74 is accessible from the research reactor facility via South Neil Street to North Neil Street. I-72 and I-57 are accessible using South Neil Street and West Church Street.

In relation to the research reactor site:

- I-72 is located 3.5 miles (5.6 km) to the west-northwest,
- I-74 is located 2.1 miles (3.4 km) to the north, and
- I-57 is located 3.4 miles (5.5 km) to the west.

Several major highways within 5 miles (8 km) of the site on which hazardous chemicals may be transported include US-45, US-150, IL-130, and IL-10. US-45 and IL-130 split from US-150 and both feed into I-74; it was assumed that any chemicals and quantities transported on I-74 are consistent with those that could be transported on US-45, US-150, IL-130 and IL-10.

- US-45 runs roughly north-south to the west of the U. of I. research reactor site; at its closest points it is approximately 0.1 miles (0.16 km) from the center of the facility (east lane to west side of the site).
- US-150 (East/West University Avenue) runs east-west to the north; at its closest points it is approximately 0.8 miles (1.3 km) from the center of the site.
- IL-130 (South High Cross Road) runs roughly north-south to the east of the site; at its closest points it is approximately 4.1 miles (6.6 km) from the center of the facility.
- IL-10 (East/West Springfield Avenue) runs roughly east-west to the north of the reactor site; at its closest points it is approximately 0.6 miles (0.97 km) from the center of the facility.

Traffic volumes, obtained from the Illinois Department of Transportation, are for 2024 and are listed below (Reference 2-20). Figure 2-15 provides a visual representation of traffic volumes near the project site for the major roadways listed below. Figure 2-16 provides a visual representation of traffic volumes on minor roadways near the project site.

- I-57 – 25,800 vehicles per day, west of the site
- I-74 – 48,900 vehicles per day, north of the site
- I-72 – 10,450 vehicles per day, northwest of the site
- US-45 – 19,600 vehicles per day, west of the site
- US-150 – 12,000 vehicles per day, north of the site
- IL-10 – 12,100 vehicles per day, northwest of the site
- IL-130 – 7,750 vehicles per day, east of the site

A list of chemicals and other hazardous materials that are likely transported along regional roadways and railways is provided in Table 2-6.

2.2.1.4 Description of Railroads

The railways through Champaign County are all Class 1 railroads. According to the Surface Transportation Board, a federal economic regulator of surface transportation, a Class 1 railroad is defined as having annual carrier operating revenues well above $400 million. The nearest railroad to the site is the Canadian National Railway, which is approximately 0.1 miles (0.16 km) west of the research reactor site (just east of IL-45) and is orientated in the north-to-south direction. There are also two other railways located near the site. The Canadian National Railway, which is oriented west-to-east, is located approximately 1.6 miles (2.6 km) northwest of the site. The Norfolk Southern Railway is located 1.2 miles (1.9 km) north of the site and is orientated east-to-west. These railways transport a variety of materials and goods, including various hazardous materials. The Illinois Terminal is located 0.8 miles (1.3 km) from the site and serves Amtrak passenger rail services utilizing the tracks of the Canadian National Railway (Reference 2-21).

2.2.1.5 Description of Other Nearby Facilities

The U. of I. research reactor facility is located in an urban area on the U. of I. campus. There are numerous commercial, industrial, educational, and health care facilities within a 5-mile (8 km) radius of the site that could possess quantities of hazardous materials. Table 2-4 lists those facilities that may provide a hazard to the U. of I. research reactor site operations. Table 2-4 lists three water treatment plants, the U. of I. Physical Plant Services building, APL Engineered Materials, Inc., research laboratories, and the nearest gasoline station.

The nearest research laboratory is the Nuclear Physics Laboratory; located approximately 0.2 miles (0.32 km) south-southwest of the research reactor site. The nearest gasoline station, Circle K, is also situated about 0.2 miles (0.32 km) southwest of the facility. Neither of these facilities, or others like them, are considered threats to the safety of the research reactor facility. While the Nuclear Physics Laboratory may contain various small quantities of laboratory chemicals and/or radioactive materials, like the other educational and research laboratories located throughout the U. of I. campus, accidents involving these materials would not have an effect on the safe operations of the reactor facility and laboratories are not analyzed further.

The Abbott Power Plant is located immediately west of the research reactor site across South Oak Street. Seven aboveground storage tanks (ASTs) ranging from 150 gallons (570 liters) to 1,300 gallons (4,900 liters) were identified within 1 mile (0.62 km) of the site. Each of these most likely contains, or contained, fuel (gasoline, diesel, or fuel oil) for construction equipment and/or backup generators and was located on U. of I. property.

Located approximately 0.6 miles (0.97 km) south-southwest of the research reactor site and north of St. Mary's Road is a solid waste transfer station. Located immediately north of the transfer station are two Fuel Oil No. 2 storage tanks, which were constructed in 1971; each tank is capable of holding 937,020 gallons (3,547,000 liters) (Reference 2-19). At the time the 2015 U. of I. Utility Master Plan was prepared, the east tank was slightly over half full while the west tank was empty and undergoing inspection and repairs (Reference 2-19). During normal operation, the oil is assumed divided evenly between the two tanks (Reference 2-19). The fuel oil is piped to the Abbott Power Plant through a 6-inch (15.4 cm) pipeline from Kirby Street in a steam tunnel along Oak Street.

The U. of I. Physical Plant Services building is situated on South Oak Street and is 0.3 miles (0.48 km) south-southwest of the research reactor site. This building houses the university's Facilities and Services Department, which provides all physical plant, operational, and essential services for the university and is considered the largest administrative unit on the U. of I. campus. Some of these essential services include pest control, stores and receiving, snow and ice removal and transportation and automotive services. The transportation and automotive services include carpool, garage, and transportation management. As such, there is a regular vehicle fueling station at their Garage and Carpool (1701 S. Oak Street) location that offers both unleaded and diesel fuel for university fleet and departmental vehicles.

APL Engineered Materials, Inc. is a specialty chemical manufacturing company located at 2401 N. Willow Road in Urbana, Illinois approximately 3 miles (4.8 km) from the research reactor site. The company manufactures high-purity, moisture-free, customized, performance chemicals (Reference 2-22). APL manufactures custom products for several conventional high intensity discharge lamp segments including metal-halide, ceramic metal-halide, high-pressure sodium, and compact fluorescent lamps (Reference 2-22). APL also manufactures and supplies high-purity, low moisture powders and beads as feedstock for single-crystal growth of scintillators as well as anhydrous lead iodide, cesium iodide and other halides for perovskite-based photovoltaics (Reference 2-22). However, no chemicals manufactured or stored at APL would have any effect on the safe operations of the research reactor facility and it is not analyzed further.

There are three water treatment plants listed in Table 2-4 that are within 5 miles (8 km) of the research reactor site. At these plants, chemical disinfectants, such as chlorine, chloramine, or chlorine dioxide, are typically added to the filtered water to eliminate remaining parasites, bacteria, or viruses. Chlorine and other chemicals are shipped to the plants by truck, and these trucks could take a route that approaches the research reactor site. However, no trucks would pass near the reactor site if traveling directly to the treatment plants from one of the three nearest interstate highways and transportation via alternative routes that would approach the research reactor facility, while possible, would be very inefficient and unlikely. Therefore, these water treatment plants and chemical shipments to these plants would not have an effect on the safe operations of the research facility and are not analyzed further.

A list of chemicals and other hazardous materials that are likely used and stored at these facilities is provided in Table 2-6. The list excludes metals and other chemicals that may only be present at APL Engineered Materials due to the small size of the facility and its distance from the reactor site (approximately 3 miles [4.8 km]).

2.2.2 Air Traffic

While an aircraft impact assessment per 10 CFR 50.150 is not applicable to a research reactor facility utilizing a Class 104(c) licensing pathway, this section is provided to supplement the discussion of offsite hazards.

2.2.2.1 Identification of Air Traffic Near the Site

There are two general aviation airports that are located within 5 miles (8 km) of the site: Willard Airport (CMI based on Federal Aviation Administration [FAA] identifier) and Frasca Field (C16 [FAA identifier]). Willard Airport is a commercial airport located southwest of the site and Frasca Field, a regional airport smaller than Willard, is located northeast of the site.

While some military air training takes place at Willard Airport, no military airports are located within 10 miles (16 km) of the site and, therefore, military airports were eliminated from further evaluation consistent with NUREG-1537. However, military aircraft were evaluated for training flights near Willard Airport and nearby airways.

2.2.2.1.1 Willard Airport

The Airport Data and Information Portal (ADIP) via the FAA provided the annual operations at Willard Airport (Reference 2-23) as 64,826 total operations in the calendar year 2025. This was subdivided by each category of aircraft: commercial aviation air carrier, commercial aviation air taxi, local general aviation, itinerant general aviation, and military. Combining local and itinerant general aviation together as "general aviation," in the calendar year 2025, Willard Airport had 3,042 commercial aviation air carrier operations, 14,009 commercial aviation air taxi operations, 47,515 general aviation operations, and 260 military operations. The type of military operations is not provided, however for conservatism in the evaluation of airway hazards, small military aircraft were assumed.

Willard Airport has four runways, three of which are currently active. The active runways include Runway-14L/32R and Runway-14R/32L, both running northwest and southeast, and Runway-4/22 which runs southwest and northeast. The runway that is not active is Runway-18/36, which runs north and south.

It was assumed that the three active runways had an equal percentage of usage (33.3 percent) as all available runways. The total number of annual operations per runway results in approximately 1,014 commercial aviation air carrier operations per runway, 4,669 commercial aviation air taxi operations per runway, 15,837 general aviation operations per runway, and 87 small military aircraft operations per runway. These operations assumed 50 percent takeoff operations and 50 percent landing operations (Reference 2-23, Reference 2-24)

2.2.2.1.2 Frasca Field (C16)

The ADIP via the FAA provided the annual operations for Frasca Field as 15,000 total operations over a one-year period from May 1, 2019 through April 30, 2020. This was subdivided into local general aviation, itinerant general aviation, and commercial aviation air taxi. Combining local general aviation and itinerant general aviation as “general aviation;” there were 13,500 general aviation operations and 1,500 commercial aviation air taxi operations that occurred over a one-year period (Reference 2-25).

Frasca Field has two runways: Runway-18/36 which runs north and south, and Runway-09/27 which runs west and east. It was assumed that the runways had an unequal percentage of usage due to Runway-18/36 having a turf surface when compared to Runway-09/27 having a concrete surface. Therefore, operation frequencies of 25 percent and 75 percent were applied to Runway-18/36 and Runway-09/27, respectively. The estimated total number of operations for Runway-18/36 are 3,375 total general aviation operations and 375 total commercial aviation air taxi operations. The estimated total amount of operations for Runway-9/27 are 10,125 total general aviation operations and 1,125 total commercial aviation air taxi operations. Commercial aviation and military aviation were assumed zero based on no recorded operations from the FAA ADIP data for Frasca Field (Reference 2-25).

2.2.2.1.3 Federal Airways

There are seven federal airways located within 10 miles (16 km) of the site (distance from the center of the research reactor site to the nearest edge of the airway). These airways are identified in Figure 2-14 and are further described in Table 2-7. Five of these routes are victor (V) or low-altitude airways and, in terms of airway centerline from site, include: V-429 (1.1 miles), V-251 (3.6 miles), V-434 (5.0 miles), V-192 (5.2 miles), and V-191 (14.2 miles). Airway J-71 is a jet route or high-altitude airway, and its airway centerline is located 7.4 miles away from the site. Airway Q-42 is an Area Navigation (RNAV) route, and its airway centerline is located 12.5 miles (20.1 km) away from the site. Most airways are 8 nautical miles (approximately 9.2 statute miles [14.8 km]) wide (Reference 2-26), and distances were measured from the airway centerline for mapping purposes.

NUREG-1537 states that “Factors such as frequency and type of aircraft movement, flight patterns, local meteorology, and topography should be considered…” However, the document does not provide a screening criterion for the distance of the airways from the facility. Therefore, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 3.5.1.6 was considered for guidance in evaluating airways near the facility. For airways where the outer edge of the airway is greater than 2 statute miles (3.2 km) from the facility, NUREG-0800 Section 3.5.1.6 allows the airway to be screened out with no further evaluation. Therefore, only four federal airways originally identified within 10 miles (16 km) of the site were identified as having an edge of the airway within 2 statute miles (3.2 km) of the site. They include V-429, V-251, V-434, and V-192. These airways were further evaluated in Section 2.2.2.2.

2.2.2.2 Evaluation of Airway Hazards

Four federal, low-altitude airways edges are within 2 miles (3.2 km) of the site. Given the approximate location of the airports and airways, crash frequencies are to be evaluated quantitatively.

The U.S. Department of Energy (DOE) provides a method for estimating the probability per year of an aircraft crashing into the facility. The methodology is outlined in DOE Standard DOE-STD-3014-2006 (Reference 2-27). Aircraft crash frequencies are estimated using a "four-factor formula," shown in Equation 2-1, which considers (1) the number of operations, (2) the probability that an aircraft will crash, (3) given a crash, the probability that the aircraft crashes into a 1-square-mile (2.6 square km) area where the facility is located, and (4) the size of the facility.

In DOE-STD-3014-2006 (Reference 2-27), the four-factor formula is implemented for near-airport activities, which consist of takeoffs and landings, through a combination of site-specific information and data obtained by the user of this standard, and a set of tables provided in Appendix B of this standard.

$$F = \sum_{ijk} N_{ijk} \cdot P_{ijk} \cdot f_{ijk}(x,y) \cdot A_{ij} \qquad \text{Equation 2-1}$$

Where:

F	=	estimated annual aircraft crash impact frequency (no./y);
N_{ijk}	=	estimated annual number of site-specific aircraft operations (takeoffs and landings) for each applicable summation parameter (no./y);
P_{ijk}	=	aircraft crash rate (per takeoff or landing for near-airport phases) for each applicable summation parameter;
$f_{ijk}(x,y)$	=	aircraft crash location conditional probability (per square mile) given a crash evaluated at the facility location for each applicable summation parameter;
A_{ij}	=	the site-specific effective area for the facility of interest that includes skid and fly-in effective areas (square miles) for each applicable summation parameter, aircraft category or subcategory, and flight phase for military aviation;
i	=	(index for flight phases): i=1 and 3 (takeoff and landing, respectively);
j	=	(index for aircraft category or subcategory): j=1, 2,..., 11;
k	=	(index for flight source): k=1, 2,..., K (there could be multiple runways, and nonairport operations);
$\sum$	=	$\sum k \sum j \sum i$;
ijk	=	site-specific summation over flight phase, i; aircraft category or subcategory, j; and flight source, k.

The effective facility area (A_j) for the safety-related structures of the site depends on the length, width, and height of the safety-related structure, as well as the aircraft's wingspan, skid distance, and impact angle as explained below:

$$A_j = A_f + A_s \qquad \text{Equation 2-2}$$

Where:

$$A_f = (WS + R) \cdot H \cdot \cot\Phi + (2 \cdot L \cdot W \cdot WS)/R + L \cdot W \qquad \text{Equation 2-3}$$

And:

$$A_s = (WS + R) \cdot S \qquad \text{Equation 2-4}$$

Where:

A_f	=	effective fly-in area
A_s	=	effective skid area
WS	=	aircraft wingspan
R	=	length of the diagonal of the facility = (L2+W2)0.5
H	=	facility height, facility-specific
cotΦ	=	mean of the cotangent of the aircraft impact angle
L	=	length of facility, facility-specific
W	=	width of facility, facility-specific
S	=	aircraft skid distance (mean value)

The following information is summarized in Table 2-7 through Table 2-10.

Tables B-14 and B-15 of DOE-STD-3014-2006 (Reference 2-27) provide NjPjFj(x,y) values for general aviation aircraft, air carriers, air taxis, and small military aircraft applicable for specific DOE sites. Tables B-14 and B-15 of DOE-STD-3014-2006 also provide crash probabilities for unspecified locations in the continental United States (CONUS). Due to conservatism, Argonne National Laboratory NjPjfj(x,y) values are used for the project site and are provided in Table 2-8.

Table B-16 in DOE-STD-3014-2006 (Reference 2-27) provides representative wingspans (WS) for commercial, general aviation, and military aircraft. For general aviation, the wingspan value for turboprop (73 feet [22.2 meters]) was selected over piston engine (50 feet [15.4 meters]) and turbo jet (50 feet [15.4 meters]) for conservatism. For commercial aviation, the wingspan for the air carrier is 98 feet (29.9 meters) and 59 feet (18.0 meters) for an air taxi. For large military aircraft, the wingspan is 223 feet (68.0 meters). For small military aircraft, the wingspan for a low performance small aircraft (110 feet [33.5 meters]) was selected over a high-performance small aircraft (78 feet [23.8 meters]) for conservatism. The WS inputs are summarized in Table 2-9.

Table B-17 in DOE-STD-3014-2006 (Reference 2-27) provides mean of the cotangent of the aircraft impact angle (cotΦ) for general aviation (8.2), commercial aviation (10.2), helicopters (0.58), large military aircraft (7.4), and small military aircraft (8.4). The aircraft impact angle inputs are summarized in Table 2-9.

Table B-18 in DOE-STD-3014-2006 (Reference 2-27) provides mean skid distances (S) for general aviation (60 feet [18.3 meters]), commercial aviation (1,440 feet [439 meters]), helicopters (0 feet), large military aircraft (780 feet [238 meters]), and small military aircraft (246 feet [75.0 meters]). The skid distances inputs are summarized in Table 2-8.

The total effective area (Aj) for the safety-related structures of the facility was calculated using Equation 2-2 through Equation 2-4. Effective area inputs are provided in Table 2-9 and Table 2-10. Dimensions of the facility used in the analysis include a width of 175 feet (53.3 meters), a length of 185 feet (56.4 meters), and assumes a height of 45 feet (13.7 meters). The calculated effective areas for the six aircraft types are provided in Table 2-11.

2.2.2.3 Evaluation of Airport Hazards and Helicopter Operations

2.2.2.3.1 Aircraft Hazards

Public resources suggest that Frasca Field supports approximately 15,000 operations per year, and Willard Airport supports approximately 65,000 operations per year. Given the approximate location of the two airports and associated operations described in Section 2.2.2.1, a calculation was conducted to estimate the crash impact frequency from nearby airport operations. This estimate was calculated in accordance with Equation 2-1 using methodologies outlined in DOE Standard DOE-STD-3014-2006 for airport operations (Reference 2-27).

Tables B-2 through B-13 of DOE-STD-3014-2006 provide f(x,y) aircraft crash location conditional probabilities for takeoff and landing for all applicable aircraft based on orthogonal x and y distances from the site to the midpoint of each runway. Because all runways are utilized for both takeoff and landing operations, the x and y distances were evaluated as both takeoff and landing operations, and the most conservative f(x,y) values were obtained based on the takeoff or landing direction using Tables B-2 through B-13 of DOE-STD-3014-2006. Tables B-2 and B-3 of DOE-STD-3014-2006 provide crash location probability takeoffs and landings values for near-airport operations of commercial aircraft (including air taxi and air carrier). Tables B-4 and B-5 of DOE-STD-3014-2006 provide crash probability takeoff and landing values for near-airport operations of general aviation. Tables B-10 through B-13 provide crash location probability takeoff and landing values for near airport operations for small military aircraft. No large military aircraft were evaluated as they resulted in less conservative crash probabilities. The f(x,y) values selected for small military aircraft used the higher of right vs left pattern side to direction of flight. Runway distances to the site are shown in Figure 2-17. Airport calculation inputs are summarized in Table 2-12.

Table B-1 of DOE-STD-3014-2006 provided aircraft crash rate (P_j) values for takeoff and landing. The crash rates for takeoff operations are as follows: 1.10E-5 for general aviation, 1.90E-7 for commercial aviation air carrier, 1.00E-6 for commercial aviation air taxi, and 5.70E-7 for military aviation large aircraft. The crash rates for landing operations are as follows: 2.00E-5 for general aviation, 2.80E-7 for commercial aviation air carrier, 2.30E-6 commercial aviation air taxi, and 1.60E-6 for military aviation large aircraft.

The effective area of the facility (A_j) was calculated as described above in Equation 2-2 through Equation 2-4. All applicable airport inputs are provided in Table 2-12. The crash impact frequency for airport operations is provided in Table 2-13.

2.2.2.3.2 Helicopter Operations

Based on an analysis of historical helicopter crash data, DOE-STD-3014-2006 states "the contribution to impact frequencies associated with nonlocal helicopter overflights is insignificant and need not be considered in the impact frequency calculations. However, it is necessary to consider local overflights, either planned overflights associated with the facility operations, e.g., security flights, or flights associated with area operations, e.g., spraying flights. Thus, the calculation of in-flight helicopter impact frequencies is a site-specific calculation. Using DOE-STD-3014-2006, the helicopter impact frequency evaluation is determined using the following formula:

$$F_H = N_H \cdot P_H \cdot A_H \cdot (2/L_H) \qquad \text{Equation 2-5}$$

Where:

F_H	=	helicopter impact frequency
N_H	=	expected number of helicopter local overflights per year
P_H	=	helicopter crash per flight (per takeoff or landing)
L_H	=	average length (in miles) of the flight
H	=	helicopter
A_H	=	effective area for helicopter in-flight crashes

The number of helicopter operations (N_H) was made using conservative assumptions based on ADIP data supplying the number of helicopters at each airport. According to the most recent ADIP data, no number was provided for the number of helicopters based at Willard Airport (Reference 2-28). According to previous ADIP data to use as a bounding case, Willard Airport had three total helicopters out of 150 total aircraft (2% of aircraft), and Frasca Field Airport has one helicopter out of 130 total aircraft (0.8% of aircraft) (Reference 2-24, Reference 2-25). By dividing the annual operations for Frasca Field Airport (Reference 2-25) and Willard Airport (Reference 2-28) by 365 days per year, the average daily operations of 41 per day and 178 per day were obtained for Frasca Field Airport and Willard Airport. The product (rounded up) of helicopter fraction at a given airport and the average number of operations per day gives a conservative estimate for the number of helicopter operations per day. This results in one helicopter operation per day for Frasca Field and four helicopter operations per day for Willard Airport. Therefore, a total of five helicopter operations per day or 1,825 helicopter operations per year are estimated to occur in proximity to the site.

Table B-1 in DOE-STD-3014-2006 provides aircraft crash rate values (P_H), and the helicopter effective area (A_H) was calculated in the same manner as Equation 2-2 through Equation 2-4. The average trip length of 33.69 miles (54.2 km) was selected based on an 8-year average of rotary wing flight lengths from Table 3.23 in (Reference 2-29). All applicable inputs are provided in Table 2-7 through Table 2-10. The crash impact frequency for helicopter operations is provided in Table 2-14.

2.2.2.4 Summary of Risks from Air Traffic

NUREG-1537 does not provide acceptance criteria to evaluate the aircraft accident probability posed by nearby airports and airways. However, NUREG-1537 states that the radiological risk from external incidents from manmade facilities (i.e., airports) are analyzed in or are shown to be bounded by accidents considered in Chapter 13 of the Preliminary Safety Analysis Report (PSAR). DOE-STD-3014-2006 provides a screening value of 1.00E-06 per year, where the risk of an aircraft accident is considered acceptable if the frequency of occurrence is less than 1.00E-06 per year (Reference 2-27).

The total crash frequency for airway operations is 2.08E-05 per year, exceeding the DOE acceptance criterion of 1.00E-06 per year. This exceedance is driven almost entirely by general-aviation traffic, which contributes approximately 99.4 percent of the total.

In the base-case calculation, the $N_jP_jF_j(x,y)$ parameter for general aviation was taken as 3.00E-03, the highest site-specific value listed in Table B-14 of DOE-STD-3014-2006. To test sensitivity, the calculation was repeated with the average CONUS $N_jP_jF_j(x,y)$ value of 2.00E-04. Even with this lower parameter, the resulting crash frequency is approximately 1.26E-06 per year, which still marginally exceeds the 1.00E-06 per year criterion.

The total crash frequency for Frasca Field airport operations is 1.37E-06, and the total crash frequency for all Willard Airport operations is 4.28E-06. Both airports exceed the criterion primarily due to general aviation. Additionally, the local helicopter operations crash frequency of 5.16E-06 exceeds the 1.00E-06 screening criterion.

The total calculated crash frequency for all airway, airport, and helicopter operations is 3.05E-05 per year. This frequency exceeds the established crash criterion primarily due to the presence of small, non-military aircraft from general aviation and helicopter operations. It is important to note that the risk associated with commercial aviation and military operations does not surpass the screening criterion.

The U. of I. research reactor is utilizing a Class 104(c) licensing pathway, therefore an Aircraft Impact Assessment for compliance with 10 CFR 50.150 is not required as part of the licensing basis for this facility. The safety-related structures of the research reactor structures will be designed to withstand the impact of a small, non-military general aviation aircraft and helicopters as described in Section 2.2.2.2.

The maximum crash frequency for all aircraft operations is provided in Table 2-15.

2.2.3 Analysis of Potential Accidents at Facilities

Each of the nearby facilities and projects listed in Table 2-4 was initially considered with respect to possible effects on the reactor facility that could precipitate an onsite event. Typical chemicals and other hazardous materials stored and used at these nearby facilities and transported in regional roadways and railways is provided in Table 2-6. The potential effects of pipeline and transportation-related accidents on the nuclear plant in terms of design parameters and physical phenomena are evaluated, using the following NRC guidance:

- Regulatory Guide (RG) 1.78, Revision (Rev.) 2: Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release
- RG 1.91, Rev. 3: Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants
- RG 4.7, Rev. 3: General Site Suitability Criteria for Nuclear Power Stations

Design-basis external events for commercial power reactors are defined in NUREG-0800, "Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants." According to this SRP, an event is retained in the design basis if its annual probability of causing a radiological release to the public is on the order of 1.00E-07 per year or greater, and if its potential consequences are significant enough that the dose guidelines in 10 CFR 100.11(a) could be exceeded.

While 10 CFR Part 100 is applicable to stationary power and testing reactors, it does not apply to research reactors licensed under 10 CFR 50.21(c). Furthermore, NUREG-1537, which serves as the primary guidance document for non-power utilization facilities, does not provide specific guidance on external events for research reactors. Therefore, in the absence of explicit regulatory direction in NUREG-1537, this analysis utilizes NUREG-0800 as a reference for external-event screening criteria for U. of I. research reactor.

This approach allows us to benchmark the external-event assessment against established NRC practices, ensuring that the safety analysis for this research reactor remains conservative and transparent. The following categories are considered in selecting design-basis events:

- explosions,
- flammable vapor clouds (delayed ignition),
- toxic chemical releases, and
- off-site fires.

The postulated events with the potential to result in a chemical release are analyzed at the following locations:

- Nearby natural gas pipelines,
- Nearby road and rail transportation routes, and
- Nearby chemical and fuel storage facilities.

2.2.3.1 Explosions

Accidents involving detonations of high explosives, munitions, chemicals or liquid and gaseous fuels are considered for facilities and activities in the vicinity of the site or onsite where such materials are processed, stored, used, or transported in quantity. The effects of explosions are considered based on structural response to blast pressures. The effects of blast pressure from explosions from nearby railways, highways, or facilities to critical plant structures are evaluated to determine if the explosion could have an adverse effect on plant operation or could prevent a safe shutdown.

NUREG-1537 does not provide specific guidance for explosions; therefore, the guidance in RG 1.91, Rev. 3 was considered in determining allowable (i.e., standoff) and actual distances of explosive chemicals transported or stored.

RG 1.91, Rev. 3 provides a series of equations to calculate the standoff distance for explosions resulting in less than 1 psi of peak incident pressure. The required standoff distance for a potential explosion to result in less than 1 psi of peak incident pressure can be determined based on the trinitrotoluene (TNT)-equivalent methodology using Equation B-1 from RG 1.91, Rev. 3. Table 1 in RG 1.91, Rev. 3 provides acceptable assumptions for determining the mass of TNT based on material type.

RG 1.91, Rev 3 cites 1 pound per square inch (psi) (6.9 kilopascal [kPa]) as a conservative value of peak positive incident overpressure, below which no significant damage would be expected. RG 1.91, Rev. 3 defines this standoff distance, shown in Equation 2-6, by the relationship:

$$R \geq KW^{1/3} \qquad \text{Equation 2-6}$$

Where:

R	=	distance in feet
W	=	equivalent pounds of trinitrotoluene (TNT)
K	=	scaled distance constant at a given overpressure at 1 psi (45 feet per lb).

The blast wave energy can be determined using Equation B-3 in RG 1.91, Rev. 3 shown below in Equation 2-7:

$$E = \alpha \cdot \Delta H_c \cdot m_F \qquad \text{Equation 2-7}$$

Where:

E	=	blast wave energy (British thermal units [BTU])
α	=	yield (i.e., the fraction of available combustion energy participating in blast wave generation)
ΔH_c	=	theoretical net heat of combustion (BTU/per pound mass [lbm])
m_F	=	mass of flammable vapor released (lbm)

Yields (α) for various explosive materials are provided in Table 1 in RG 1.91, Rev. 3. Tables in NUREG-1805, "Fire Dynamics Tools" provides values for the theoretical net heat of combustion. NUREG-1805 is for power reactors and is used as guidance in the absence of guidance in NUREG-1537. The blast wave energy requires a conversion to a corresponding TNT equivalent mass. The conversion is calculated using Equation B-4 from RG 1.91, Rev. 3 in Equation 2-8 below:

$$W_{TNT} = \frac{E}{1900BTU/lbm}$$ Equation 2-8

Where:

W_{TNT} = TNT equivalent mass (lbm)

E = blast wave energy (BTU)

For those chemicals where the standoff distance using the NUREG-1805 methods is greater than the actual distance from the chemical to the safety-related portion of the U. of I. research reactor, a probabilistic analysis may be used to show that "the rate of exposure to a peak positive incident overpressure in excess of 1 pound per square inch differential pressure (6.9 kPa) is less than 1.00E-06 per year, when based on conservative assumptions, or 1.00E-07 per year when based on realistic assumptions."

Conservative assumptions are used to determine a standoff distance, or minimum separation distance, required for an explosion to have less than 1 psi (6.9 kPa) peak incident pressure. In each of the explosion scenario analyses, an explosion yield factor of 100 percent is applied to account for an in-vessel confined explosion (except for pipeline releases to unconfined environment [yield of 15 percent]). The yield factor is an estimation of the available combustion energy released during the explosion, as well as a measure of the explosion confinement. This is a conservative assumption and represents an upper limit because a 100 percent yield factor is not achievable.

For some atmospheric liquids (e.g., diesel), the storage vessel was assumed to contain fuel vapors at the upper explosive limit (UEL). This is conservative because the UEL produces the maximum explosive mass given that it is the vapor, not the liquid fuel, that explodes. These assumptions are consistent with those used in Chapter 15 of NUREG-1805.

For compressed or liquefied gases (e.g., propane, hydrogen), it was conservatively assumed that the entire content of the storage vessel is between the UEL and lower explosive limit (LEL), given that the instantaneous depressurization of the vessel would result in vapor concentrations throughout the explosive range at varying pressures and temperatures that could not be assumed. Therefore, the entire content of the storage vessel was considered as the explosive mass.

An additional type of stationary explosion is a boiling liquid expansion vapor explosion (BLEVE). In a BLEVE, a tank of liquefied and (typically) refrigerated gas is released to the environment. The chemical flashes from liquid to vapor, causing a pressure wave. RG 1.91, Table 1 assumes a yield factor of 15 percent. Because all explosion calculations account for a conservative yield of 100 percent (except for pipelines), the BLEVEs impacts are a fraction of the explosion calculation results for liquefied gases and were therefore not specifically analyzed.

In some cases, chemicals are screened as being bounded by other chemicals. Three properties of the chemical hazard are used to determine if one of the hazards is bounded by another. First, chemicals that are gases at standard conditions would be more volatile and have a larger explosive mass per storage mass than chemicals that are liquids at standard conditions. Second, chemicals with a smaller LEL and a greater UEL would be more explosive. A larger flammable or explosive range would make an explosion more likely and increase the explosive mass per storage mass. Third, chemicals with a greater heat of combustion would have a larger amount of energy release in an explosion. In addition, the mass of the chemical and the distance from the chemical to the site are screening factors. Chemicals that are closer to the site and in larger tanks are chosen as bounding over chemicals that are farther or smaller. As the site is close to railroad tracks and a highway (US-45), transportation vessels are considered for these bounding scenarios.

The representative chemicals of interest transported and stored within 5 miles (8 km) of the site, indicated in Table 2-6, are evaluated to ascertain which hazardous materials have the potential to explode, thereby requiring further analysis. A summary of explosive findings is provided in Table 2-16.

2.2.3.1.1 Pipelines

A natural gas pipeline explosion occurring in the vicinity of the release point along either pipeline within the 5-mile (8-km) vicinity of the U. of I. research reactor site would be unconfined and, according to the NRC Safety Evaluation Report for Robinson (former Hartsfield) Nuclear Power Plant (NUREG-0014), a damaging detonation from an unconfined natural gas release is not credible. However, ignition of a natural gas release near the release point could result in a deflagration explosion or jet fire, which would result in less damage than unconfined detonation and from explosions where the natural gas vapor cloud becomes confined either outside or by migration inside a building. Thus, the dominant hazards from exterior natural gas pipelines are from the heat effect of thermal radiation from a sustained jet fire.

Even though the immediate ignition of natural gas resulting in overpressure events resulting from a ruptured gas pipeline was considered an unlikely event, an evaluation was conservatively conducted to evaluate a potential explosion from the natural gas transmission pipeline. While nearly all of the natural gas transmission lines in the vicinity of the U. of I. research reactor site are underground, the 10-inch (114-cm) 400 psig (maximum 618 psig) line leaves the ground at the pressure regulator yard located adjacent to the south side of the Abbott Power Plant. In an accident scenario, it is assumed that the 10-inch pipes bursts open (i.e., double-ended guillotine break/shear failure), leaving the full cross-sectional area of the pipe completely exposed to the air. It is also assumed that the ignition source existed at the break point. The standoff distance to 1 psi overpressure was calculated by determining the mass of natural gas released, whereby the TNT mass equivalency methodology can then be employed as described previously.

The maximum gas discharge rate, in pound per second (lb/s), from the break in the pipeline was calculated using Equation 2-9 which represents the release from the pipeline as an orifice (Reference 2-30).

$$Q_{max} = C \times A \times P_0 \sqrt{\frac{\gamma \times g_c \times MW}{R \times T} \times \left(\frac{2}{\gamma + 1}\right)^{\left(\frac{\gamma + 1}{\gamma - 1}\right)}} \qquad \text{Equation 2-9}$$

Where:

Q_{max} is the maximum gas discharge rate (lb/s)

C is the discharge coefficient (1 for maximum case),

A is the area of the puncture in square feet,

g_c is the gravitational constant (32.17 feetlb$_m$/lb$_f$second^2),

MW is the natural gas molecular weight (19.5 lb$_m$/lb-mole), (Table 4-3, Reference 2-30)

R is the ideal gas constant (1545 feet · lb$_f$/lb-mol · Rankine [°R]),

P_0 is the pipeline pressure,

γ is the heat capacity ratio for natural gas (1.27), (Table 4-3 Reference 2-30), and

T is the initial pipeline temperature (°R)

Due to the nature of a high-pressure release through a pipeline, upon a complete pipeline rupture, the release rate of the gas (lb/s) will initially be very large, but within seconds the release rate will drop to a fraction of the initial release rate. Therefore, to estimate the amount of gas discharged for an instantaneous release, the maximum discharge rate was conservatively assumed to occur for a period of 5 seconds, in accordance with methodology performed in the Clinch River Nuclear Site Early Site Permit Application (ESPA) Site Safety Analysis Report (SSAR) (Reference 2-31). This duration maintained the intent of the instantaneous detonation as applied in the TNT analysis; any longer and atmospheric dispersion effects will predominate resulting in a traveling vapor cloud while maximizing the amount of gas released for the TNT analysis. This is also a conservative assumption given that the discharge rate will begin to decrease significantly immediately after the break occurs. The amount of gas released was then determined by:

$$Mass\ (lb) = Q_{max}\left(\frac{lb}{s}\right) \times Time(s) \qquad \text{Equation 2-10}$$

Using the flammable mass calculated by the above methodologies, the equivalent mass of TNT was calculated using Equation 2-9 and Equation 2-10. The results indicated that the standoff distance (the distance to where the peak incident pressure does not exceed 1 psi) of 0.32 miles (0.51 km) for the 10-inch (114-cm) diameter pipeline is greater than the minimum separation distance from the U. of I. research reactor site to the south side of the Abbot Power Plant (0.04 miles [64 meters]). Therefore, overpressures from an explosion due to a rupture in the 10-inch (114-cm) natural gas pipeline could potentially and adversely affect the safe operation or shutdown of the reactor, and such an event must be considered in the design of the U. of I. research reactor facilities.

2.2.3.1.2 Waterway Traffic

Analysis for potential impacts due to water transportation of hazardous materials is not necessary. The three rivers (and/or creeks) are not considered to be major waterways. Therefore, there are no transport of hazardous materials via shipping cargos.

2.2.3.1.3 Highways and Railways

I-72, I-74, I-57, US-45, US-150, IL-130, and IL-10 have been identified as roads within 5 miles (8 km) of the site where hazardous chemicals may be transported. In a commodity flow study conducted by the Champaign County Regional Planning Commission on interstate highways, it was shown that the predominantly transported hazards were Class 3: Flammable Liquids at 39 percent, followed by hazard Class 8: Corrosive Materials at 19.5 percent, and hazard Class 9: Miscellaneous Dangerous Goods at 18 percent (Reference 2-21).

Because of nearby transportation scenarios involving explosive materials (e.g., hydrogen, butane, and gasoline/crude oil), the specific analysis of determining allowable (i.e., standoff) and actual distances of explosive materials transported must be quantitatively examined. For this calculation, the methodology outlined in RG 1.91, Rev. 3 was utilized for estimating the standoff distance for the explosion to have less than 1 psi of peak incident pressure. In RG 1.91, Rev. 3, these formulas are implemented for the calculation of the minimum standoff distance at nearby transportation routes based on estimates of TNT-equivalent mass of potentially explosive materials. The safe explosion distances results were then compared to the minimum distance from each respective highway and railway to the U. of I. research reactor site. US-45, US-150, IL-10, and IL-130 were evaluated in explosion calculations involving tanker trucks, as these highways were the most limiting due to proximity to the site. The Clinch River Nuclear Site ESPA SSAR (Reference 2-31) was referenced for gasoline and butane, as the explosions hazard would be the same regardless of where representative tanker was transported.

The Clinch River Nuclear Site SSAR reported that the standoff distance for an 8,500-gallon gasoline tanker truck explosion with less than 1 psi peak incident pressure is 273 feet (83.2 meters). The U. of I. research reactor site is approximately 530 feet (161 meters) from US-45 with the railroad tracks (on a berm approximately 6 feet high [1.8 meters]) and the large Abbott Power Plant building in the direct line of the shortest distance to US-45. The closest distance from US-45 to the center of the U. of I. research reactor site unobstructed by a building is at the intersection of US-45 and Stadium Dr., about 1,000 feet (305 meters) from the site. At this location, the berm containing the railroad track is over 10 feet (3.1 meters) high. Therefore, a gasoline tanker explosion on US-45 would not adversely affect the safe operation or shutdown of the reactor at the U. of I. research reactor site.

The Clinch River SSAR also reported that the minimum standoff distance for an 11,500-gallon (43,500-liter) compressed gas tanker with butane is 3,700 feet (1,130 meters). This distance exceeds the minimum distances of US-45 and IL-10 of 0.1 mile (0.16 km) and 0.6 mile (1.0 km), respectively. Therefore, explosions related to butane could adversely affect the safe operation or shutdown of the reactor, and such an event must be considered in the design of the U. of I. research reactor facilities.

Regarding hydrogen, the Clinch River Nuclear Site SSAR only considered the transport of a single tube of compressed hydrogen gas. However, a conveyance typically consists of nine tubes (Reference 2-32). Explosion evaluations were performed in accordance with RG 1.91 to evaluate conveyances containing nine superjumbo hydrogen tubes and conveyances with cryogenic liquid hydrogen. Nine super jumbo tube volumes correspond to an equivalent volume of 135,297 cubic feet at standard conditions (Reference 2-32). The standoff distances for a compressed gas tanker carrying nine super jumbo hydrogen tubes resulting in an explosion with less than 1 psi peak incident pressure is 1,200 feet (366 meters). This standoff distance exceeds the minimum distances of US-45 of 0.1 miles (0.16 km). Therefore, explosions related to hydrogen gas cylinders could adversely affect the safe operation or shutdown of the reactor, and such an event must be considered in the design of the U. of I. research reactor facilities.

For the cryogenic liquid hydrogen calculation, it was assumed that a cryogenic cargo tanker (e.g., Department of Transportation [DOT] MC 338, or similar) carries 13,620 gallons (51,560 liters) of water (Reference 2-33). However, 49 CFR 173.318 requires a maximum filling density of 6.60 percent, where filling density is defined as the percent ratio of the weight of lading in the tank to the weight of water that the tank will hold at the design service temperature. Assuming a conservative density of 8.33 pounds/gallon for water, a mass of liquid hydrogen could be determined. The standoff distances for a cryogenic cargo tanker carrying liquid hydrogen resulting in an explosion with less than 1 psi peak incident pressure is 2,700 feet (823 meters). This standoff distance exceeds the minimum distances of US-45 of 0.1 miles (0.16 km). Therefore, explosions related to cryogenic liquid hydrogen transportation could adversely affect the safe operation or shutdown of the reactor, and such an event must be considered in the design of the U. of I. research reactor facility.

With regard to railways, in 2020 approximately 107 trains, often carrying hazardous materials, passed through Champaign County's 225 miles (78 km) of active rails lines each day. Freight traffic accounts for about 101 of these trains. These freighters transport a variety of materials and goods, including various hazardous materials, usually diesel fuel or crude oil. The Champaign County Emergency Management Agency and the Champaign County Regional Planning Commission have conducted commodity flow studies on the transportation of hazardous materials for both the Norfolk Southern and Canadian National Railroads. Every hazardous material is assigned to one of nine hazard classes as defined by the U.S. DOT. According to the study, using 2017 and 2018 data for Canadian National Railway and the Norfolk Southern Railway, the majority of the railcars carrying hazardous materials were of hazmat Class 3: Flammable Liquids, followed by hazmat Class 8: Corrosive Materials, and hazmat Class 2: Gases and hazmat Class 9: Miscellaneous Dangerous Goods. (Reference 2-21). The results of the study are replicated in Table 2-17, indicating the percentage of railcars carrying each of the nine classes of hazardous materials.

For the U. of I. research reactor site, the nearest rail line corresponding to Canadian National Railway (north-south) is approximately 450 feet (approximately 0.1 mile [0.16 km]) to the west edge of the site along South Oak Street. Additionally, two other rail lines are within proximity to the U. of I. research reactor site. They include the Canadian National Railway (east-west) and Norfolk Southern Railway (east-west). Therefore, railcars containing 34,500 gallons (130,600 liters) of gasoline (assumed crude oil as an analogue for gasoline), cryogenic liquid hydrogen, and butane were evaluated in accordance with 49 CFR 179.13 volume limits for railcars. Railcars were evaluated at volumes corresponding to one railcar volume and twenty railcar volumes. The twenty railcar volumes were selected as an upper bound to represent a very conservative case and corresponds to the 49 CFR 171.8 definition of high-hazard flammable train (HHFT).

The standoff distances for a single and twenty railcar volumes carrying crude oil (gasoline analogue) resulting in an explosion with less than 1 psi peak incident pressure is 560 feet (171 meters) and 1,500 feet (457 meters), respectively. The single and twenty railcar scenario standoff distances are within the minimum distance of the U. of I. research reactor site from the Canadian National Railway (north-south) at

0.1 miles (0.16 km). Therefore, explosions related to crude oil/gasoline rail transportation could adversely affect the safe operation or shutdown of the reactor, and such an event must be considered in the design of the U. of I. research reactor facilities.

The standoff distances for a single and twenty railcar volumes carrying cryogenic liquid hydrogen resulting in an explosion with less than 1 psi peak incident pressure is 3,700 feet (1,128 meters) and 10,000 feet (3,048 meters), respectively. The single and twenty railcar scenario standoff distances are within the minimum distance of the U. of I. research reactor site from the Canadian National Railway (north-south). Additionally, the twenty-railcar scenario standoff distances are within the minimum distance of the U. of I. research reactor site from the Canadian National Railway (east-west) and Norfolk Southern Railway (east-west) at 1.8 miles (2.9 km) and 1.2 miles (1.9 km), respectively. Therefore, explosions related to transport of liquid hydrogen by rail could adversely affect the safe operation or shutdown of the reactor, and such an event must be considered in the design of the U. of I. research reactor facilities.

The standoff distances for a single and twenty railcar volumes carrying butane resulting in an explosion with less than 1 psi peak incident pressure is 5,300 feet (1,615 meters) and 15,000 feet (4,572 meters), respectively. The single and twenty railcar scenario standoff distances are within the minimum distance of the U. of I. research reactor site from the Canadian National Railway (north-south). Additionally, the twenty-railcar scenario standoff distances are within the minimum distance of the U. of I. research reactor site from the Canadian National Railway (east-west) and Norfolk Southern Railway (east-west). Therefore, explosions related to transport of butane by rail could adversely affect the safe operation or shutdown of the reactor, and such an event must be considered in the design of the U. of I. research reactor facilities.

As shown in Table 2-16, the analysis indicates that certain highway and railway transport hazards yield calculated standoff distances that exceed the minimum separation distance from the reactor site. Consequently, explosions resulting from transportation operations could pose a risk to the safe operation or shutdown of the reactor, necessitating consideration of such events in the design of U. of I. research reactor and facility. However, it is important to note that, as outlined in RG 1.206, "Applications for Nuclear Power Plants," Rev. 1, a design-basis event is defined as one with a probability of occurrence on the order of 1E-07 per year or greater, with potential consequences significant enough to impact plant safety to the extent that the guidelines in 10 CFR 100 could be referenced for consistency check.

The bounding scenario for the design of the reactor building in regards to transportation of chemicals on highway and railway is from the twenty railcar volumes for butane. The individual railcar volume is provided at 34,500 gallons (130,600 liters) as regulated 49 CFR 179.13, resulting in a total volume of 690,000 gallons (2,612,000 liters). The resulting mass of the twenty railcar volumes is 3,300,000 lbm. The standoff distance to reach the 1-psi peak incident pressure from this analysis is 2.80 miles (4.5 km). The nearest rail line is the Canadian National Railway (north-south) and is located approximately 450 feet (137 meters) from the proposed facility.

2.2.3.1.4 Onsite Chemicals

The location and quantities of chemicals that would be stored at the site have not yet been determined. The effects on explosion events from onsite chemical storage would be evaluated in the Operating License Application and accounted for in the final design. Chemicals stored at the future reactor site would be maintained and stored in a manner that would be protective of onsite personnel, the U. of I. research reactor, and the public.

2.2.3.1.5 Nearby Facilities

Table 2-16 identifies and evaluates specific stored chemical explosion risks for nearby facilities and transportation routes, consistent with the guidance of RG 1.91, Rev. 3. As noted previously, university laboratories and facilities that may contain small quantities of hazardous materials are not considered viable hazards to the U. of I. research reactor facility.

Nine facilities near the proposed site were initially considered for evaluation:

- Abbott Power Plant natural gas pipeline located 0.05 miles (0.08 km) west of the proposed U. of I. research reactor site;
- U. of I. Nuclear Physics Laboratory, an experimental nuclear physics research facility, approximately 0.2 miles (0.3 km) south-southwest of the proposed U. of I. research reactor site;
- Circle K Gas Station, a convenience store and gas station, approximately 0.2 miles (0.3 km) south-southwest of the proposed U. of I. research reactor site;
- U. of I. F&S Physical Plant Services building, a hub for essential services offered to the university, approximately 0.3 miles (0.5 km) south-southwest of the proposed U. of I. research reactor site;
- U. of I. fuel oil storage tanks, two nearly 1,000,000-gallon (3,790,000 liter) ASTs that feed the Abbott Power Plant with fuel oil (Reference 2-19), approximately 0.6 miles (1.0 km) south-southwest of the proposed U. of I. research reactor site;
- Champaign Water Treatment Plant, a wastewater treatment plant, approximately 2.8 miles (4.5 km) northwest of the proposed U. of I. research reactor site;
- Urbana & Champaign Sanitary District Northeast Treatment Plant, a wastewater treatment plant, approximately 2.6 miles (4.1 km) east northeast of the proposed U. of I. research reactor site;
- Urbana & Champaign Sanitary District Southwest Treatment Plant, a wastewater treatment plant, approximately 5.0 miles (8.0 km) west southwest of the proposed U. of I. research reactor site;
- and APL Engineered Materials, Inc., a high-purity performance chemical manufacturing facility, approximately 3.1 miles (5.0 km) northeast of the proposed U. of I. research reactor site.

As mentioned previously, there are various U. of I. educational and research laboratories, but the quantities of hazardous materials at these locations are small.

For this calculation, the methodology outlined in RG 1.91, Rev. 3 was utilized for estimating the standoff distance for the explosion to have less than 1 psi of peak incident pressure. In RG 1.91, Rev. 3, these formulas are implemented for the calculation of the minimum standoff distance at nearby facilities based on estimates of TNT-equivalent mass of potentially explosive materials. The safe explosion standoff distances were then compared to the minimum distance from each respective facility.

Of the facilities listed above, only the two Abbott Power Plant ASTs (1,000,000 gallons [3,790,000 liters] each) were evaluated for explosion hazards. All other facilities are either bound by the explosion hazards associated with highways and railways or they posed no credible explosion hazard to the U. of I. research reactor site. Both sources of explosion evaluated for nearby facility explosions contain fuel oil no. 2 (a.k.a. #2 fuel oil, #2 heating oil, 2 oil, off-road diesel fuel). These vapors may be ignited rapidly when exposed to heat, spark, open flame, or other source of ignition. When mixed with air and exposed to an ignition source, flammable vapors can burn in the open or explode in confined spaces. Being heavier than air, vapors may travel long distances to an ignition source and flash back (Reference 2-34). For conservatism, the volumes of the storage tanks were combined (e.g., a single 2,000,000-gallon [7,570,000-liter] storage tank) and assumed at full capacity.

With respect to explosion potential, the calculated minimum standoff distances for representative materials are shown in Table 2-16. The two 2,000-gallon (7,570-liter) backup generator fuel-oil tanks were not evaluated due to the expectation of their removal prior to construction of the U. of I. research reactor. The standoff distance for the two above-ground 1,000,000-gallon (3,790,000-liter) storage tanks is 2,000 feet (610 meters), less than the separation distance of 0.6 miles (0.97 km).

As noted earlier, a design-basis event is retained whenever its annual probability of occurrence is on the order of 1E-07 per year or greater and its consequences are large enough to challenge plant safety. This probability threshold – drawn from RG 1.206 and the power-reactor SRP – invokes the numerical framework of 10 CFR 100 solely to demonstrate consistency with established NRC practice; 10 CFR 100 itself does not apply to the Class 104(c) licensing pathway of this research reactor. If a probabilistic risk assessment using realistic assumptions shows that the likelihood of an explosion is less than 1E-07 using realistic assumptions (or 1E-06 under conservative assumptions), the scenario would not be classified as a design-basis event. These events will be addressed in the Operating License Application, if necessary.

Of the nearby facilities, only the two above-ground storage tanks at the Abbott Power Plant were analyzed. Calculations demonstrate that the peak incident pressure result from an explosion of these tanks at the reactor building will be less than 1 psi. Therefore, these tanks are not of concern.

2.2.3.2 Flammable Vapor Clouds

Flammable materials, whether in liquid or gaseous state, are hypothetically considered to have the potential to form an unconfined vapor cloud that could drift toward the reactor site prior to ignition. When a flammable chemical is released into the atmosphere, it can create a vapor cloud that disperses as it travels downwind. Within this cloud, certain regions may contain concentrations of the flammable substance that fall within the defined flammable range, specifically between the lower flammability limit (LFL) and the upper flammability limit (UFL).

If the vapor cloud encounters an ignition source, the portions of the cloud that are within this flammable range could ignite and burn. The rate at which the flame front propagates through the cloud would determine whether the combustion event is classified as a deflagration or a detonation. Should combustion occur at a sufficiently rapid rate to generate a detonation, an explosive force may be produced, posing a significant hazard (Reference 2-35).

A comprehensive catalogue of hazardous substances was previously compiled for Clinch River Nuclear Site ESPA, Part 2, SSAR. This catalogue includes chemicals used or stored at Oak Ridge National Laboratory that are also representative of those found in university research and teaching laboratories within a 5-mile (8-km) radius of the research reactor site. These substances are listed in Table 2-6.

The Clinch River Nuclear Site ESPA, Part 2, SSAR identified only anhydrous ammonia – a non-fuel chemical whose vapors can explode – as warranting further evaluation for a potential design-basis flammable-vapor-cloud event. Using the Areal Locations of Hazardous Atmospheres (ALOHA) air dispersion model and site-specific meteorological data, that study determined a safe standoff distance of

342 feet (104 meters) to ensure that any resulting overpressure remains below 1 psi. Applying these conservative results to the research reactor shows that no research laboratory or other potential ammonia source lies within this distance; the nearest candidate facility, the Nuclear Physics Laboratory, is more than 800 feet (244 meters) from the reactor site. Consequently, accidents involving off-site non-fuel chemicals, including anhydrous ammonia, pose no credible hazard to reactor safety and do not require design-basis consideration.

Two delayed-ignition phenomena were nevertheless reviewed for completeness:

1. Flash fire (deflagration). For chemicals with defined lower and upper flammability limits (LFL/UFL), the extent of a potential vapor cloud was estimated to determine whether any portion could reach the reactor site at flammable concentrations before ignition.

2. Vapor-cloud explosion (detonation). The corresponding peak overpressure was evaluated to confirm that, at the project site, it would not exceed the 1 psi (6.9 kPa) threshold associated with significant structural damage.

Both evaluations confirm that, at the U. of I. research reactor site, potential releases of anhydrous ammonia from sources beyond 342 feet (104 meters) would not pose a credible threat to the safe operation or shutdown of the research reactor. The Clinch River analysis provides a conservative basis for this conclusion, but the results are specific to the research reactor and its surrounding conditions.

2.2.3.2.1 Pipelines

As described previously, there are two high-pressure (700 and 400 psig) natural gas pipelines within 5 miles (8 km) of the U. of I. research reactor site as well as a campus-wide network of low-pressure (40 psig) pipelines. The 400-psig pipeline is within 130 feet (39.6 meters) from the U. of I. research reactor Licensed Area boundary. The 700-psig pipeline is located at Curtis Road 2.5 miles (4.0 km) south-southwest of the U. of I. research reactor site.

A stationary explosion of a pipeline is bounded by the delayed ignition explosion of a pipeline. This is because the constant mass release rate from the pipe results in a much larger total explosive mass, and because the wind is conservatively assumed to blow the release toward the site. The distance from the point of the explosion to the site is therefore much smaller for flammable vapor clouds than for pipeline explosions at the release point.

The distance a vapor cloud could travel to reach the LFL boundary once a vapor cloud has formed from an accidental release of natural gas from the pipeline is determined using the ALOHA dispersion model (modeled as methane). The LFL boundary distances for the 400-psig pipeline located at the Abbott Power Plant was determined to be 2,154 feet (657 meters). The LFL boundary distances for the 700-psig pipeline located at Curtis Road was determined to be 603 feet (184 m). Furthermore, safe distances for vapor cloud exposure resulting in less than 1 psi of peak incident pressure were 1,690 feet (515 meters) and 1,584 feet (483 meters) for the 400-psig and 700-psig pipelines, respectively. The U. of I. research reactor License Area boundary is within the 1 psi of peak incident pressure distance for the 400-psig pipeline while the License Area boundary is outside the 1 psi of peak incident pressure for the 700-psig pipeline.

Lastly, a safe distance evaluating the heat flux of 5 kilowatts per square meters (kW/m^2) from a jet fire scenario concluded distances were 237 feet (72 meters) and 299 feet (91 meters) were required for the 400-psig and 700-psig pipelines, respectively. Therefore, a jet fire or a flammable vapor cloud ignition or explosion from a rupture in the 400 psig is expected to adversely affect the safe operation or shutdown of the reactor at the U. of I. research reactor site. A jet fire or a flammable vapor cloud ignition or explosion from a rupture in the 700 psig is not expected to adversely affect the safe operation or shutdown of the reactor.

As previously discussed, a design-basis event is defined as one with a probability of occurrence on the order of 1E-07 per year or greater and with consequences significant enough to challenge plant safety. While the guidelines in 10 CFR 100 are referenced for consistency, they are not directly applicable to the U. of I. research reactor. If a probabilistic risk assessment demonstrates that the likelihood of a pipeline explosion is less than 1E-07 per year under realistic assumptions, or less than 1E-06 per year for conservative assumptions, the event would not qualify as a design-basis event. These events will be addressed in the Operating License Application, if necessary.

The bounding scenario for the pipelines is the 400-psig pipeline. Both the vapor cloud distance and the heat flux from a jet fire scenario are located within the safe distance to the reactor facility location.

2.2.3.2.2 Waterway Traffic

Analysis for potential impacts due to water transportation of hazardous materials giving rise to flammable vapor clouds is not necessary. There are no waterways near the U. of I. research reactor site that transport hazardous materials that could impact safe operation or shutdown of the reactor at the site.

2.2.3.2.3 Highways and Railways

As indicated previously, there are multiple interstate highways, US, and state highways within 5 miles (8 km) of the U. of I. research reactor site. The closest major roadway is US-45, located about 530 feet (177 meters) west of the site.

The effects of various materials transported along major roadways were evaluated as part of the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 2-31) using the ALOHA dispersion model to determine the safe distance for each postulated flammable vapor cloud scenario. Of the materials evaluated, only butane and gasoline were deemed to be of potential significance for flammable vapor clouds. There is no reason to conclude that the results of the modeling conducted for the Clinch River Nuclear Site SSAR would be significantly different than modeling conducted for the U. of I. research reactor site. The Clinch River Nuclear Site SSAR model results indicated that the longest flammable plume for butane would be 1,827 feet (557 meters) and 132 feet (40.2 meters) for gasoline. The distance from US-45 to the U. of I. research reactor site is about 330 feet (101 meters), indicating that there is no concern for a flammable plume from an accident involving a gasoline tanker, but an accident involving a butane tanker must be considered in the design of the U. of I. research reactor facility.

A vapor cloud explosion analysis was completed for the Clinch River Nuclear Site SSAR to obtain safety distances (the minimum distances required for an explosion to have less than a 1 psi peak incident pressure). The safe distance for butane was determined to be 3,864 feet (1,178 meters) and 618 feet (188 meters) for gasoline. The distance from the U. of I. research reactor site to US-45 is less than these distances, and the U. of I. research reactor facility design will consider these hazards unless a probabilistic risk assessment can demonstrate the risk of the incident occurring is less than 1E-07 using realistic assumptions (or 1E-06 for conservative assumptions), the accident would not be considered a design-basis event. These events will be addressed in the Operating License Application, if necessary.

Where rail corridors exist within 5 miles (8 km), hazardous-material transport would be treated analogously, using the same Clinch River SSAR basis and the ALOHA-based plume and overpressure surrogates to conservatively bound potential impacts at the U. of I. research reactor.

2.2.3.2.4 Onsite Chemicals

The location and quantities of chemicals that would be stored at the site have not yet been determined. The effects of flammable vapor clouds and vapor cloud explosions from onsite chemical storage will be evaluated in the Operating License Application.

2.2.3.2.5 Nearby Facilities

The Clinch River Nuclear Site SSAR (Reference 2-31) considered the hazards from flammable vapor clouds for a comprehensive list of chemicals at nearby facilities. As noted previously, the facilities located near the U. of I. research reactor site and listed in Table 2-4 are similar to those discussed in the Clinch River Nuclear Site SSAR, and the chemicals and hazardous materials are listed in Table 2-6. While these chemicals were considered for potential impacts to the safe operation and shutdown of the U. of I. research reactor facility, no facilities were identified that would be expected to create a more limiting hazard than those described in Section 2.2.3.1 for highways and railroads. Therefore, none of these facilities are considered a threat to safe operation or shutdown of the U. of I. research reactor facility.

Representative chemicals either used or stored within 5 miles (8 km) of the U. of I. research reactor site are identified in Table 2-6. With the potential for formation of flammable/explosive vapor clouds at the U. of I. research reactor site, a subset of these representative chemicals was identified as receiving a flammability analysis in the “Disposition” column of Table 2-6. These chemicals were evaluated with respect to their potential for formation of flammable/explosive vapor clouds. Each material was then dispositioned based on its identified physical properties and whether a bounding analysis exists. The results of this evaluation are found in Table 2-18, and the table footnotes indicate the reason for the disposition for those flammable/explosive material not carried forward for further analysis.

2.2.3.3 Toxic Chemicals

Events involving the release of toxic or asphyxiating chemicals in the vicinity of the project site are considered for their potential toxicity and ability to affect personnel. The potential for an offsite toxic gas release is evaluated within 5 miles (8 km) of the U. of I. research reactor site.

Given the proximity of major pipeline, roadways and railways to the U. of I. research reactor site, the evaluation considered only mobile sources. The effects of a chemical release from a pipeline are considered bound by the delayed ignition explosion of the nearby natural gas pipeline.

Chemicals are evaluated based upon their properties, quantities, and distances relative to the U. of I. research reactor site without consideration of plant design factors, such as control room ventilation. At this time, the control room characteristics (e.g., the control room volume and outside air infiltration and circulation rates) of the are unknown. Therefore, chemicals that lead to concentrations above the Immediately Dangerous to Life or Health (IDLH) limit at the Licensed Area boundary will be identified and evaluated during development of the Operating License Application.

The ALOHA air dispersion model was used to predict the chemical concentrations within a toxic or asphyxiating vapor cloud as it disperses downwind from all facilities and sources. ALOHA is a diffusion model that permits temporal as well as spatial variations. In the case of a toxic vapor cloud, the maximum distance a cloud can travel before it disperses enough to fall below the IDLH or other determined toxicity limit concentration in the vapor cloud is determined using ALOHA. Asphyxiating chemicals were evaluated to determine if their release resulted in the displacement of a significant fraction of the control room air. The Occupational Safety and Health Administration (OSHA) provides guidance on what is considered an oxygen-deficient atmosphere.

The IDLH is defined by the National Institute of Occupational Safety and Health (NIOSH) as a situation that poses a threat of exposure to airborne contaminants when that exposure is likely to cause death or immediate or delayed permanent adverse health effects, or prevent escape from such an environment. The IDLHs are determined by NIOSH so that workers are able to escape such environments without suffering permanent health damage. Where an IDLH is unavailable for a toxic chemical, the time-weighted average (TWA) or threshold limit value, promulgated by OSHA or adopted by the American Conference of Governmental Hygienists, is used as the toxicity limit.

Each postulated toxicity or asphyxiation event was evaluated under a spectrum of meteorological conditions, in accordance with RG 1.206, Rev. 1, to determine the worst-case meteorological condition. The meteorological parameters selected for the meteorological sensitivity analysis was selected based on the defined Pasquill meteorological stability classes (see Table 2-20). The analysis included the most stable meteorological class, F, which is allowable within the ALOHA model. Generally, the dispersion model predicts higher concentrations in the formed cloud before reaching safety-related structures. Furthermore, as cited in RG 1.78, “Evaluating the Habitability of a Nuclear Power Plans Control Room During a Postulated Hazardous Chemical Release, “Rev. 2, the Pasquill Stability Category F represents the worst 5th-percentile meteorology observed at most nuclear power plant sites.

Other atmospheric inputs/assumptions for the ALOHA model included:

- Ground Roughness: The model selected “Urban or Forest” was selected for the ground roughness. The degree of atmospheric turbulence influences how quickly a pollutant cloud moving downwind will mix with the air around it and will be diluted. Friction between the ground and air passing over it is one cause of atmospheric turbulence. The rougher the ground surface, the greater the turbulence that develops. “Urban or Forest” assumes a chemical cloud is traveling over an area with many friction generating elements, such as trees or small buildings. The release locations for each of the postulated scenarios are only 0.1 miles (0.16 km) from the U. of I. research reactor site and require a formed vapor cloud to travel over few friction generating elements.

- Liquid State Chemicals: For each of the identified (representative) chemicals in the liquid state (i.e., under the atmospheric release conditions for that scenario, the physical state of the substance is expected to be a liquid), it was conservatively assumed that the entire contents of the vessel are released. Additionally, for this release, the contents formed an instantaneous 0.4-inch (1-cm) thick puddle. This provided a significant surface area from which to maximize evaporation and formation of a vapor cloud.
- Gaseous State Chemicals: For each of the identified (representative) chemicals in the gaseous state, or for those chemicals that are normally gases at ambient temperatures (i.e., under the atmospheric release conditions for that scenario, the physical state of the substance was expected to be a gas), it was assumed that the quantity released from the vessel/pipeline is released over a 10-min period into the atmosphere as a continuous direct source (40 CFR 68.25).

The evaluation of potential toxic and asphyxiating chemical releases around the project site indicates that the methodologies employed, including the ALOHA model and the established toxicity limits, provide a comprehensive assessment of hazards. The findings will inform the development of the Operating License Application, ensuring that all relevant safety considerations are addressed.

2.2.3.3.1 Pipelines

There is one bounding natural gas pipeline within 5 miles (8 km) of the site. Natural gas is predominantly methane. The toxicity hazard from methane is that of a simple asphyxiant, and there are no defined IDLH or Emergency Response Planning Guideline levels for methane. The distance to the asphyxiating limit for the Abbott Plant pipelines was evaluated under the worst-case meteorological conditions of 1,209 feet (369 meters) and 3,597 feet (1,096 meters), respectively. The 1,209 feet (369 meters) for the 400-psig pipeline is greater than the separation distance from the pipeline, which is less than 150 feet (45.7 meters) southwest of the perimeter of the U. of I. research reactor site. Therefore, a break in the pipeline could displace enough oxygen for the control room to become an oxygen-deficient atmosphere. A cloud of methane would reach potentially explosive concentrations before displacing enough oxygen to cause asphyxiation. Therefore, the bounding hazard from natural gas is a potential explosion or fire, which was addressed in Section 2.2.3.2 and determined to be potentially a threat to the reactor site.

2.2.3.3.2 Waterway Traffic

Analysis for potential impacts due to water transportation of hazardous materials is not necessary, as there are no transportation waterways within 5 miles (8 km) of the reactor site.

2.2.3.3.3 Highways and Railways

The U. of I. research reactor site area is located approximately 2.1 miles (3.4 km) north of I-74 (closest interstate highway), approximately 0.1 miles (0.16 km) from US-45, and less than 0.1 miles (160 m) from the Canadian National Railway. For this analysis, these distances were also used as the distance from the transportation route to the U. of I. research reactor site control room.

The hazardous chemicals evaluated are provided in Table 2-6. This is a comprehensive list taken from the Clinch River Nuclear Site ESPA, Part 2, SSAR and considered representative of the chemicals expected to be transported in the Urbana-Champaign area and applicable to the analysis of the U. of I. research reactor site given similar proximity to major roadways, local industry, and research laboratories. These same chemicals were considered for analysis in rail transportation due to the close proximity of a railway to the research reactor site.

An evaluation of hazardous materials potentially transported on US-45 was performed using the ALOHA dispersion model. The results indicated that including anhydrous ammonia and chlorine, the distances to the identified toxicity limit for any plausible toxic vapor cloud that could form following an accidental release at the closest approach from US 10 are greater than the minimum separation distances from the U. of I. research reactor site to US-45 (approximately 0.1 miles). A release of anhydrous ammonia would result in a distance of 2.6 miles (4.2 km) to the toxicity endpoint, and a release of chlorine results in a distance of 4.5 miles (7.2 km) to the toxicity endpoint, which are both greater than the distance separating the site and US-45 as well as I-74. Therefore, an incident involving chlorine, anhydrous ammonia, gasoline, nitric acid, sulfur hexafluoride and vinyl chloride on US-45, I-74 and other primary transportation routes could have an adverse impact on the U. of I. research reactor control room. As such, the U. of I. research reactor control room will be designed with the respective detectors in the ventilation system as discussed in PSAR Section 7.8.

2.2.3.3.4 Onsite Chemicals

The location and quantities of chemicals that would be stored at the site have not yet been determined. The effects of toxic chemicals or fires resulting from onsite chemical storage will be evaluated in the application for the operating license.

2.2.3.3.5 Nearby Facilities

Representative chemicals either used or stored within 5 miles (8 km) of the U. of I. research reactor site are identified in Table 2-6. With the potential for formation of toxic vapor clouds at the U. of I. research reactor site, a subset of these representative chemicals was identified as receiving a toxicity analysis in the "Disposition" column of Table 2-6. These chemicals were evaluated with respect to its potential for formation of toxic vapor clouds and analyzed using the ALOHA dispersion model.

However, no nearby facilities present a greater hazard from toxic chemical hazards than the hazards provided by transport of toxic chemicals on nearby highways and railways. These hazards are described in the previous subsection and conclude that an incident involving chlorine, anhydrous ammonia on US-45, I-74 and other primary transportation routes could have an adverse impact on the U. of I. research reactor control room. Therefore, the U. of I. research reactor control room will be built with chlorine and ammonia detectors in the ventilation system as discussed in PSAR Section 7.8.

2.2.3.4 Fires

As demonstrated in the previous sections, potential external accidents near the U. of I. research reactor site could lead to high heat fluxes. The analyses showed that natural gas in nearby pipelines and chemicals stored at nearby facilities and transported on roadways and railways could result in explosions and/or vapor clouds with a potential to affect the U. of I. research reactor site.

External accidents were considered in the vicinity of the U. of I. research reactor site that could lead to high heat fluxes or smoke, and nonflammable gas or chemical-bearing clouds from the release of materials as a consequence of fires. Fires from nearby facilities and fires from transportation accidents are evaluated as events that could lead to high heat fluxes or to the formation of such clouds.

The threat of fires initiated by offsite events exists in any urban environment such as the U. of I. campus and surrounding areas, but the risk of a large fire engulfing multiple structures is rare. In the 5-mile (8-km) radius of the U. of I. research reactor site, fires are responded to by the City of Champaign Fire Department and Urbana Fire Rescue Services, since the Urbana-Champaign campus does not operate its own independent municipal fire department.

Table 2-4 Nearby Facilities and Projects

Facility/Project Name	Summary of Facility/Project	Distance from Licensed Area	Status	Industrial Hazard/Hazardous Material(s)	Evaluated for Potential Effect on U. of I. Research Reactor Site Operations
Industrial Facilities					
Abbott Power Plant	Generators with two 2,000-gallon above ground storage tanks.	0.01-mile N	Operational	Fuel Oil No. 2	No
Abbott Power Plant	Natural gas power plant	0.05-mile W	Operational	Natural Gas	Yes
U. of I. Nuclear Physics Laboratory	Experimental nuclear physics research facility	0.2-mile SSW	Operational	Laboratory chemicals; radioactive materials	No
Circle K	Gasoline Station	0.2-mile SW	Operational	Unleaded gasoline and diesel fuel; 20-pound propane tanks	No. Transportation route accident bounds gasoline station accident scenario.
U. of I. Facilities and Services Physical Plant Services Building	Physical Plant Services	0.3-mile SSW	Operational	Pesticides; cleaning, disinfection and janitorial supplies; brine de-icing solution; unleaded and diesel fuel	No
Fuel Oil Storage Tanks	Two 1,000,000-gallon above ground storage tanks; feeds Abbott Power Plant	0.6-mile SSW	Operational	Fuel Oil No. 2	Yes
Champaign Water Treatment	Wastewater Treatment Plant	2.8-mile NW	Operational	Chlorine, chloramine, and/or chlorine dioxide	No. Primary transportation route for hazardous materials would avoid the U. of I. Research Reactor site.
Urbana & Champaign Sanitary District Northeast Plant	Wastewater Treatment Plant	2.6-mile ENE	Operational	Chlorine, chloramine, and/or chlorine dioxide	
Urbana & Champaign Sanitary District Southwest Treatment Plant	Wastewater Treatment Plant	5.0-mile WSW	Operational	Chlorine, chloramine, and/or chlorine dioxide	

Table 2-4 Nearby Facilities and Projects (Continued)

Facility/Project Name	Summary of Facility/Project	Distance from Licensed Area	Status	Industrial Hazard/Hazardous Material(s)	Evaluated for Potential Effect on U. of I. Research Reactor Site Operations
APL Engineered Materials, Inc.	High-purity performance chemical manufacturing facility	3.0 miles NE	Operational	High purity and ultra-dry metal halides, amalgams, metals, mercury https://aplmaterials.com	No. Location contains primarily solid materials. Uncontrolled accidental release of hazardous materials is unlikely.
U. of I. laboratories	Educational and research laboratories	Various	Operational	Small quantities of laboratory chemicals	No. Quantities of hazardous materials are too small to provide a credible hazard.
Transportation					
U.S. Highway 45	United States Highway Route 45	0.1 miles W	Operational	Gasoline, hydrogen, and butane	Yes
U.S. Highway 150	United States Highway Route 150	1.1 miles NE	Operational		Yes
Illinois Highway 10	Illinois Highway Route 10	0.6 miles N	Operational		Yes
Illinois Highway 130	Illinois Highway Route 130	3.2 miles ENE	Operational		Yes
Interstate Highway 57	United States Interstate Route 57	3.4 miles W	Operational		Yes
Interstate Highway 72	United States Interstate Route 72	2.3 miles WNW	Operational		Yes
Interstate Highway 74	United States Interstate Route 74	2.1 miles N	Operational		Yes
Canadian National Railway (north-south)	Canadian National north-south railroad	0.1 miles W	Operational	Gasoline, hydrogen, and butane	Yes
Canadian National Railway (west-east)	Canadian National west-east railroad	1.8 miles NW	Operational		Yes
Norfolk Southern Railway (east-west)	Norfolk Southern east-west railroad	1.2 miles N	Operational		Yes

Table 2-5 Pipeline Information Summary

Operator	Pipeline Age	Product	Pipeline Diameter	Operating Pressure	Depth of Burial	Distance Between Isolation Valves
University of Illinois Urbana-Champaign	2000/2001	Natural Gas	10-inches	400 psig (618 psig max.)	4 foot minimum. Deeper at crossings	2 miles
University of Illinois Urbana-Champaign	1987/1988	Natural Gas	8-inches	700 psig (858 psig max.)	4 foot minimum. Deeper at crossings	11 miles on segment 1 and 8 miles on segment 2

* - While the natural gas transmission lines in the vicinity of the U. of I. research reactor site are underground, the 10-inch 400-psig line leave the ground at the pressure regulator yard located adjacent to the south side of the Abbott Power Plant.

Table 2-6 Chemical Storage and Transportation – Disposition

Material/Chemical	Toxicity Limit (IDLH)	Flammability LEL-UEL	Explosion Hazard	Vapor Pressure	Disposition
Water Treatment Plants[a]					
Chlorine	10 ppm	Not flammable	None listed	7600 mmHg @ 86°F (6.8 atm)	Toxicity analysis[c]
Laboratory and Other Campus Facilities[b]					
Anhydrous Ammonia	300 ppm	16-25%	Vapor may explode	400 mmHg @ -49.72°F	Toxicity analysis[c]
Argon	Not available	Not flammable	None listed	1,044,630 Pa @ 117.3K	Toxicity analysis[c]
Carbon Dioxide	Not available	Not flammable	None listed	56.5 atm @ 68°F	Toxicity analysis[c]
Chloroform	Not available	Not flammable	None listed	400 mmHg @ -49.72°F	Toxicity analysis[c]
Chromic Chloride	25 mg/m^3	Not flammable	None listed	Not available	Toxicity analysis[c]
Diesel Fuel No. 2	Not available	1.3-6%	Vapor may explode	2.17 mmHg @ 7I	Toxicity analysis[c] Flammability analysis[d] Explosion analysis[e]
Ferric Sulfate	Not available	Not flammable	Vapor may explode	Not available	No further analysis required[f]
Fertilizer	Not available	Not flammable	None listed	Not available	No further analysis required[f]
Gasoline (unleaded)	300 as TWA 750 ppm (as n-Heptane)	1.4-7.6%	Vapor may explode	382.58 mmHg	Toxicity analysis[c] Flammability analysis[d] Explosion analysis[e]
Hydrogen Fluoride	30 ppm	Not flammable	None listed	783 mmHg	Toxicity analysis[c]
Lead	100 mg/m^3	Not flammable	None listed	1.77 mmHg	No further analysis required[f]
Limestone	Not available	Not flammable	None listed	0 mmHg	No further analysis required[f]
Lithium Hydride	0.5 mg/m^3	Flammable	None listed	0 mmHg	No further analysis required[f]
Mercury	10 mg/m^3	Not flammable	None listed	0.0012 mmHg	No further analysis required[f]
Nitric Acid	25 ppm	Not flammable	None listed	48 mmHg	Toxicity analysis[c]
Nitrogen	Asphyxiant	Not flammable	None listed	1.931 psi @ -344°F	Toxicity analysis[c]

Table 2-6 Chemical Storage and Transportation – Disposition (Continued)

Material/Chemical	Toxicity Limit (IDLH)	Flammability LEL-UEL	Explosion Hazard	Vapor Pressure	Disposition
Oils	2500 mg/m^3	Combustible; no flammable limits	None listed	2.17 mmHg @ 70°F 0.042 psi	No further analysis required(f)
Sodium Chloride	Not available	Not flammable	None listed	1 mmHg	No further analysis required(f)
Sodium Bisulfate Solution	Not available	Not flammable	None listed	Solid – in solution	No further analysis required(f)
Sodium Hydroxide Solution	10 mg/m^3	Not flammable	None listed	Solid – in solution	No further analysis required(f)
Sodium Metal	Not available	Flammable	Vapor may explode	Not available	No further analysis required(f)
Sulfuric Acid	15 mg/m^3	Not flammable	None listed	1 mmHg @ 294.8°F	No further analysis required(f)
Sulfur Hexafluoride	1000 ppm as TWA	Not flammable	None listed	21.5 atm	Toxicity analysis(c)
Transportation on Roadways and Railways (b)					
Anhydrous Ammonia	300 ppm	16-25%	Vapor may explode	400 mmHg @ -49.72°F	Toxicity analysis(c)
Argon	Not available	Not flammable	None listed	1,044,630 Pa @ 117.3K	No further analysis required(f)
Butane	Asphyxiant	1.6-8.4%	Vapor may explode	760 mmHg @ 31.1°F	Flammability analysis(d) Explosion analysis(e)
Carbon Dioxide	40,000 ppm	Not flammable	None listed	56.5 atm @ 68°F	No further analysis required(f)
Chlorine	10 ppm	Not flammable	None listed	7600 mmHg @ 86°F (6.8 atm)	Toxicity analysis(c)
Chloroform	500 ppm	Not flammable	None listed	160 mmHg	No further analysis required(f)
Chromic Chloride		Not flammable	None listed		No further analysis required(f)
Ethanol	3300 ppm 10% LEL	3.3-19%	Vapor may explode	40 mmHg @ 66°F	No further analysis required(f)

Table 2-6 Chemical Storage and Transportation – Disposition (Continued)

Material/Chemical	Toxicity Limit (IDLH)	Flammability LEL-UEL	Explosion Hazard	Vapor Pressure	Disposition
Gasoline	300 as TWA 750 ppm (as n-Heptane)	1.4-7.6%	Vapor may explode	382.58 mmHg	Toxicity analysis[(c)]Flammability analysis[(d)]Explosion analysis[(e)]
Hydrogen Gas	Not available	4-75%	Vapor may explode	1.231 psi @ -434°F	Explosion analysis[(e)]
Hydrogen Fluoride	30 ppm	Not flammable	None listed	783 mmHg	No further analysis required[(f)]
Nitric Acid	25 ppm	Not flammable	None listed	48 mmHg	Toxicity analysis[(c)]
Nitrogen	Asphyxiant	Not flammable	None listed	1.931 psi @ -344°F	No further analysis required[(f)]
Sodium Hypochlorite	10 ppm as chlorine	Not flammable	None listed	12.1 mmHg @ 68°F	No further analysis required[(f)]
Sulfur Hexafluoride	1000 ppm as TWA	Not flammable	None listed	21.5 atm	Toxicity analysis[(c)]
Vinyl Chloride	Not available	3.6-33.0%	Vapor may explode	3.3 atm 3877.5 mmHg	Toxicity analysis[(c)]Flammability analysis[(d)]Explosion analysis[(e)]

Notes:

[a] Only chlorine was considered present at waste water treatment plants in sufficient quantities to warrant further analysis, other waste water treatment plant chemicals previously identified (e.g., aluminum chloralhydrate, fluoride, sodium hydroxide, sulfuric acid, etc.) (Reference 2-31) do not require further analysis.

[b] List derived from the list of chemicals likely present at the Oak Ridge National Laboratory provided in the Clinch River Nuclear Site SSAR (Reference 2-31); excludes metals and other chemicals that may only be present at APL Engineered Materials due to the small size of the facility and its distance from the U. of I. research reactor site (approximately 3 miles).

[c] Toxicity analysis using the ALOHA air dispersion model.

[d] Flammability (i.e., flammable vapor cloud) analysis using ALOHA air dispersion model.

Table 2-6 Chemical Storage and Transportation – Disposition (Continued)

[e] Explosion analysis per guidance in RG 1.91, Rev. 3, Evaluation of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plant.

[f] Per Clinch River Nuclear Site SSAR (Reference 2-31)

atm – atmosphere

IDLH – Immediately Dangerous to Life and Health

K – Kelvin

LEL – lower explosive limit

mg/m^3 – milligrams per cubic meter

mmHg – milligrams of mercury

Pa – Pascal

ppm – parts per million

TWA – time-weighted average

UEL – upper explosive limit

Sources for Table 2-6 chemical information: Reference 2-36, Reference 2-37, Reference 2-38

Table 2-7 Federal Airways within 10 miles of the Site

Airway	Distance from Airway Centerline to Site (miles)[(a)]	Airway Width (miles)[(b)]	Distance from Airway Edge to Site center (miles)[(c)]
V429	1.08	9.2	-[(d)(e)]
V251	3.57	9.2	-[(d)(e)]
V434	5.04	9.2	0.4[(e)]
V192	5.17	9.2	0.6[(e)]
J71	7.39	9.2	2.8
Q42	12.45	9.2	7.9
V191	14.19	9.2	9.6

[a] Statute miles

[b] Federal Airways and Jetways are 8 nautical miles (approximately 9.2 statute miles) wide (Reference 2-26)

[c] To calculate the distance from an airway edge to the center of the site, the airway edge was assumed to extend one-half of a standard airway width in all directions from the airway centerline, including past the termination of an airway at a navigational aid.

[d] Airway width encompasses site

[e] Airway within the NUREG-0800 criteria of 2-statute miles from site

Table 2-8 Airway Inputs

	NjPjfj(x,y) Values[(a)] (1/mi^2)
Air Carrier	7.00E-07
Air Taxi	4.00E-06
General Aviation	3.00E-03
Small Military	8.00E-07
Large Military	9.00E-08
Helicopter	n/a

[a] (Reference 2-27) Tables B-14 and B-15 for Argonne National Laboratory

Table 2-9 Effective Area Inputs

	WS (feet)[(a)]	cot[(φ)(b)]	S (feet)[(c)]
Air Carrier	98	10.2	1440
Air Taxi	59	10.2	1440
General Aviation	73	8.2	60
Small Military	110	8.4	246
Large Military	223	7.4	780
Helicopter	50	0.58	0

[a] WS – wingspan; reference (Reference 2-27), Table B-16

[b] φ – aircraft impact angle; reference (Reference 2-27), Table B-17

[c] S – skid distance; reference (Reference 2-27), Table B-18, takeoff skid length for in-flight crashes, if available.

Table 2-10 Approximate U. of I. Research Reactor Site Dimensions

	Feet	Miles
Length (L)	185[a]	3.51E-02[a]
Width (W)	175[a]	3.32E-02[a]
Height (H)	45	8.52E-03

[a] Foundation Plan, Drawing # UIC2001-SK-100-01 Reference (Reference 2-40).

Table 2-11 Calculated Effective Areas of Safety-Related Structures (square miles) by Aircraft Type Used for the Evaluation of Airways and Airport

Aircraft Type	Effective Fly-in Area (Af) (m^2)	Effective Skid Area (As) (m^2)	Effective Facility Area (Aj) (m^2)
Air Carrier	7.87E-03	1.82E-02	2.61E-02
Air Taxi	6.87E-03	1.62E-02	2.31E-02
General Aviation	6.17E-03	7.06E-04	6.88E-03
Small Military	7.12E-03	3.22E-03	1.03E-02
Large Military	8.91E-03	1.34E-02	2.23E-02
Helicopter	1.91E-03	0.00E+00	1.91E-03

Source: Reference 2-27

Table 2-12 Airport Inputs

Airport: Runway	Ortho. X (miles)	Ortho. Y (miles)	Crash Probability Category	Annual Operations	Crash Probability Takeoff Maximum [f(x,y)]	Crash Probability Landing Maximum [f(x,y)]	Aircraft Crash Rate Takeoff (P_{ijk})	Aircraft Crash Rate Landing (P_{ijk})
Frasca Field (C-16): 9/27	2.14	2.77	General Aviation	5063	9.10E-04	1.10E-03	1.10E-05	2.00E-05
			Commercial Air Carrier	N/A	N/A	N/A	1.90E-07	2.80E-07
			Commercial Air Taxi	563	1.00E-03	Null	1.00E-06	2.30E-06
			Large Military Aircraft	N/A	N/A	N/A	5.70E-07	1.60E-06
			Small Military Aircraft	N/A	N/A	N/A	1.80E-06	3.30E-06
Frasca Field (C-16): 18/36	3.14	2.46	General Aviation	1688	5.20E-04	7.50E-04	1.10E-05	2.00E-05
			Commercial Air Carrier	0	N/A	N/A	N/A	N/A
			Commercial Air Taxi	188	9.20E-04	N/A	1.00E-06	2.30E-06
			Large Military Aircraft	0	N/A	N/A	N/A	N/A
			Small Military Aircraft	0	N/A	N/A	N/A	N/A
Willard Airport (CMI): 4/22	4.51	1.51	General Aviation	7918	3.20E-03	7.40E-04	1.10E-05	2.00E-05
			Commercial Air Carrier	507	1.50E-03	6.50E-04	1.90E-07	2.80E-07
			Commercial Air Taxi	2335	1.50E-03	6.50E-04	1.00E-06	2.30E-06
			Large Military Aircraft	0	N/A	N/A	N/A	N/A
			Small Military Aircraft	43	1.70E-02	9.50E-03	1.80E-06	3.30E-06
Willard Airport (CMI): 14R/32L	1.6	4.57	General Aviation	7918	1.50E-05	6.10E-04	1.10E-05	2.00E-05
			Commercial Air Carrier	507	6.80E-05	N/A	1.90E-07	2.80E-07
			Commercial Air Taxi	2335	6.80E-05	Null	1.00E-06	2.30E-06
			Large Military Aircraft	0	N/A	N/A	N/A	N/A
			Small Military Aircraft	43	1.90E-03	6.60E-04	1.80E-06	3.30E-06
Willard Airport (CMI): 14L/32R	1.63	4.41	General Aviation	7918	1.50E-05	6.10E-04	1.10E-05	2.00E-05
			Commercial Air Carrier	507	6.80E-05	N/A	1.90E-07	2.80E-07
			Commercial Air Taxi	2335	6.80E-05	N/A	1.00E-06	2.30E-06
			Large Military Aircraft	0	N/A	N/A	N/A	N/A
			Small Military Aircraft	43	1.90E-03	6.60E-04	1.80E-06	3.30E-06

Table 2-13 Airport-Only Crash Probabilities

Airport Operations	Frasca Field (C-16)		Willard Airport (CMI)		Total
	Takeoff	Landing	Takeoff	Landing	
Air Carrier	0.00E+00	0.00E+00	4.11E-09	2.41E-09	6.52E-09
Air Taxi	1.70E-08	0.00E+00	8.82E-08	8.06E-08	1.86E-07
General Aviation	4.15E-07	9.40E-07	1.93E-06	2.13E-06	5.42E-06
Small Military	0.00E+00	0.00E+00	1.68E-08	1.60E-08	3.28E-08
Large Military	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total	4.32E-07	9.40E-07	2.04E-06	2.23E-06	5.65E-06

Table 2-14 Helicopter Inputs

Expected helicopter overflights per year (NH)	1,825[a]
PH, helicopter crash per flight (PH)	2.50E-05[b]
Average helicopter flight length, miles (LH)	33.69[c]

[a] Estimated. Assumes four nearby helicopter operations per day using FAA ADIP data for both Willard Airport (Reference 2-24) and Frasca Field Airport (Reference 2-25).

[b] Reference (Reference 2-27), Table B-1

[c] Reference (Reference 2-29), Table 3.23

Table 2-15 Crash Probability Summary for All Operations

Aircraft Type	Airway Operations	Airport Takeoff Operations	Airport Landing Operations	Helicopter Operations	Total
Air Carrier	1.83E-08	4.11E-09	2.41E-09	N/A	2.48E-08
Air Taxi	9.24E-08	1.05E-07	8.06E-08	N/A	2.78E-07
General Aviation	2.06E-05	2.35E-06	3.07E-06	N/A	2.61E-05
Small Military	8.27E-09	1.68E-08	1.60E-08	N/A	4.10E-08
Large Military	2.01E-09	N/A	N/A	N/A	2.01E-09
Helicopter	N/A	N/A	N/A	5.16E-06	5.16E-06
Total	2.08E-05	2.48E-06	3.17E-06	5.16E-06	3.16E-05

Source: Reference 2-27

Table 2-16 Design-Basis Events – Explosions

Source	Chemical Evaluated	Quantity Analyzed	Heat of Combustion (Btu/lb)	Distance to Site (miles)	Safe Distance for Explosion to have less than 1 psi of Peak Incident Pressure (feet)
Nearby Offsite Facilities					
Fuel Oil Storage Tanks	Fuel Oil. No. 2	2,000,000 gal	17,100	0.6 miles SSW	0.38 mile
Nearby Transport Routes/Roadway					
U.S Highway 45	Gasoline	8,500 gal	18,720	0.1 miles W	0.05 mile [(a)]
	Hydrogen (compressed gas)	135,297 $feet^3$ [(b)]	56,200		0.23 mile
	Hydrogen (liquid)	13,620 gal tank[(c)]	56,200		0.52 mile
	Butane	11,500 gal	19,512		0.70 mile [(a)]
U.S Highway 150	Gasoline	8,500 gal	18,800	1.1 miles NE	0.05 mile [(a)]
	Hydrogen (compressed gas)	135,297 $feet^3$ [(b)]	56,200		0.23 mile
	Hydrogen (liquid)	13,620 gal tank [(c)]	56,200		0.52 mile
	Butane	11,500 gal	19,512		0.70 mile [(a)]
Illinois Highway 10	Gasoline	8,500 gal	18,720	0.6 miles N	0.05 mile [(a)]
	Hydrogen (compressed gas)	135,297 $feet^3$ [(b)]	56,200		0.23 mile
	Hydrogen (liquid)	13,620 gal tank [(c)]	56,200		0.52 mile
	Butane	11,500 gal	19,512		0.70 mile [(a)]
Illinois Highway 130	Gasoline	8,500 gal	18,720	3.2 miles ENE	0.05 mile [(a)]
	Hydrogen (compressed gas)	135,297 $feet^3$ [(b)]	56,200		0.23 mile
	Hydrogen (liquid)	13,620 gal tank [(c)]	56,200		0.52 mile
	Butane	11,500 gal	19,512		0.70 mile [(a)]
Nearby Transport Routes/Railway					
Canadian National Railway (north-south)	Gasoline [(d)]	690,000 gal [(e)]	18,400	0.1 miles W	0.28 mile (0.11 mile for single car)
	Hydrogen [(f)]	690,000 gal [(g)]	56,200		1.9 mile (0.7 mile for single car)
	Butane	690,000 gal [(h)]	19,700		2.8 mile (1.0 mile for single car)

Table 2-16 Design-Basis Events – Explosions (Continued)

Source	Chemical Evaluated	Quantity Analyzed	Heat of Combustion (Btu/lb)	Distance to Site (miles)	Safe Distance for Explosion to have less than 1 psi of Peak Incident Pressure (feet)
Canadian National Railway (west-east)	Gasoline [d]	690,000 gal [e]	18,400	1.8 miles NW	0.28 miles (0.11 miles for single car)
	Hydrogen [f]	690,000 gal [g]	56,200		**1.9 miles** (0.7 miles for single car)
	Butane	690,000 gal [h]	19,700		2.8 miles (1.0 miles for single car)
Norfolk Southern Railway (east-west)	Gasoline [d]	690,000 gal [e]	18,400	1.2 miles N	0.28 miles (0.11 miles for single car)
	Hydrogen [f]	690,000 gal [g]	56,200		1.9 miles (0.7 miles for single car)
	Butane	690,000 gal [h]	19,700		2.8 miles (1.0 mile for single car)
Nearby Transport Routes/Pipelines					
400 psig pipeline (north-south)	Natural Gas (as methane)	4,561 lbs [i]	21,496	0.04 miles SW	0.32 mi
700 psig pipeline (east-west)	Natural Gas (as methane)	4,026 lbs [i]	21,496	2.5 miles S	0.30 mi

[a] Obtained from the Clinch River Nuclear Site ESPA SSAR (Reference 2-31)

[b] Equivalent ambient volume of 9 total tubes (Reference 2-32)

[c] Nominal water volume. Assumes liquid hydrogen filling density of 6.6% in accordance with 49 CFR 173.318

[d] Assumes crude oil

[e] Assume a total tank volume of 20 DOT-111 or DOT-117 tank cars based off 49 CFR 171.8 definition for high-hazard flammable train (HHFT)

[f] Assume cryogenic liquid

[g] Assume a total tank volume of 20 DOT-113 tank cars based off 49 CFR 171.8 definition for HHFT

[h] Assume a total tank volume of 20 DOT-112 tank cars based off 49 CFR 171.8 definition for HHFT

[i] Quantity of natural gas released over 5 seconds afeeter a postulated pipeline rupture at maximum pipeline pressure

Table 2-17 Section on Explosions – 2017 and 2018 Champaign County Commodity Flow Study on Railcars

Hazmat Class	Canadian National Railroad	Norfolk Southern Railroad
Class 1, Explosives	0%	0%
Class 2, Gases	15.2%	13.8%
Class 3, Flammable Liquids	45.6%	47.1%
Class 4, Flammable Solids	0.9%	1.2%
Class 5, Oxidizing Substances	5.4%	5.3%
Class 6, Toxic Substances	1%	0.9%
Class 7, Radioactive Materials	0%	0%
Class 8, Corrosive Substances	17.9%	18.2%
Class 9, Miscellaneous Dangerous Good	14%	13.5%

Source: Reference 2-21

Table 2-18 Design-Basis Events – Flammable Vapor Clouds (Delayed Ignition) Deflagration and Detonation and Jet Fire

Source	Chemical Evaluated	Quantity Analyzed	LEL (ppm)	Distance to Site (feet)	Distance to LEL (feet)	Jet Fire – Distance to 5 kW/m^2 (feet)
Nearby Offsite Facilities						
Fuel Oil Storage Tanks	Fuel Oil. No. 2	2,000,000 gal	10,000	3,168	-	
Nearby Transport Routes/Roadway						
U.S Highway 45	Gasoline	8,500 gal	10,500	528	144[c]	
	Butane	11,500 gal	16,000		1,821[c]	
	Vinyl Chloride	34,500 gal	36,000		1,032[d]	
Nearby Transport Routes/Railway						
Canadian National Railway (north-south)	Gasoline	690,000 gal [e]	18,400	528	-	
	Butane	690,000 gal [h]	19,700		-	
	Vinyl Chloride	690,000 gal	36,000		-	
Nearby Transport Routes/Pipelines						
400 psig pipeline (north-south)	Natural Gas (as methane)	185,454 lbs [a]	50,000	211	2,154[c]	237[f]
700 psig pipeline (east-west)	Natural Gas (as methane)	1,207,136 lbs [b]	50,000	13,200	603[e]	399[g]

a Calculated in ALOHA using diameter (10 inches), length between shutoff valves (2.5 miles), pressure (414.7 psia), and a 1-hour release

b Calculated in ALOHA using diameter (8 inches), length between shutoff valves (conservatively assumed minimum length of 200 times the pipe diameter), pressure (714.7 psia), and a 1-hour release

c Worst-case scenario meteorological condition is F stability class at 1 m/s

d Worst-case scenario meteorological condition is F stability class at 2 m/s

e Worst-case scenario meteorological condition is D stability class at 5.5 m/s

f Worst-case scenario meteorological condition is 0% relative humidity

g Worst-case scenario meteorological condition is 0% relative humidity

Table 2-19 Design-Basis Events - Toxic Vapor Clouds

Source	Chemical Evaluated	Quantity Analyzed	IDLH	Distance to Site (feet)	Distance to IDLH (feet)
Nearby Offsite Facilities					
Water Treatment Plant	Chlorine	44,000 lb	10 ppm	13,728	29,040[(a)]
Laboratory and Other Campus Facilities	Anhydrous Ammonia	11,500 gal	300 ppm	4,330	17,424[(b)]
	Argon	9,999 lb	71,400 ppm (asphyxiant)		198[(c)]
	Carbon Dioxide	50,000 lb	40,000 ppm		549[(c)]
	Chloroform	50,000 lb	500 ppm		1,101[(d)]
	Chromic Chloride	50,000 lb	25 mg/m^3		-[(e)]
	Hydrogen Fluoride	50,000 lb	30 ppm		32,208[(b)(f)]
	Nitrogen	50,000 lb	71,400 ppm (asphyxiant)		1134[(c)]
Nearby Transport Routes/Roadway					
U.S Highway 45	Gasoline	8500 gal	750 ppm (as n-heptane)	528	636[(c)]
	Nitric Acid	6000 gal	25 ppm		5,211[(a)]
	Sulfur Hexafluoride	50,000 lb	1000 ppm (TWA)		3,684[(c)]
	Vinyl Chloride	34,500 gal	2100 ppm		7,920[(c)]

Table 2-19 Design-Basis Events - Toxic Vapor Clouds

Source	Chemical Evaluated	Quantity Analyzed	IDLH	Distance to Site (feet)	Distance to IDLH (feet)
Nearby Pipelines					
400 psig pipeline (north-south)	Natural Gas (as methane)	185,454 lbs	71,400 ppm (asphyxiant)	211	1,209[(c)]
700 psig pipeline (east-west)	Natural Gas (as methane)	1,207,136 lbs	71,400 ppm (asphyxiant)	13,200	3,597[(c)]

[a] Worst-case scenario meteorological condition is F stability class at 3 m/s

[b] Worst-case scenario meteorological condition occurs with F stability class at 1, 2, and 3 m/s

[c] Worst-case scenario meteorological condition is F stability class at 1 m/s

[d] Worst-case scenario meteorological condition is D stability class at 5.5 m/s

[e] Not assessed using ALOHA. Chromic chloride is an extremely low volatility species (normal boiling point of 1300°C).

[f] Concentration above IDLH was predicted beyond maximum distance allowed in ALOHA of 6 miles.

Table 2-20 ALOHA Meteorological Sensitivity Analysis Inputs

ALOHA Vapor Cloud Analyses (Toxic, Flammable and Explosion [Delayed Ignition])[(a)]				
Stability Class	**Surface Wind Speed (m/s)**	**Cloud Cover (percent)**		**Date/ Time**
A	1.5	0		June 1, 2025 / 12 PM
B	1.5	50		June 1, 2025 / 12 PM
C	3	50		June 1, 2025 / 12 PM
C	5.5	0		June 1, 2025 / 12 PM
D	3	50		June 1, 2025 / 5 AM
D	5.5	50		June 1, 2025 / 12 PM
E	1	50		June 1, 2025 / 5 AM
E	2	50		June 1, 2025 / 5 AM
F	1	0		June 1, 2025 / 5 AM
F	2	0		June 1, 2025 / 5 AM
F	3	0		June 1, 2025 / 5 AM
ALOHA Jet Fire Analysis				
Stability Class	**Surface Wind Speed (m/s)**	**Cloud Cover (percent)**	**Humidity (percent)**	**Date/ Time**
F	1	0	0	June 1, 2025 / 5 AM
F	1	0	25	June 1, 2025 / 5 AM
F	1	0	50	June 1, 2025 / 5 AM
F	1	0	75	June 1, 2025 / 5 AM
F	1	0	100	June 1, 2025 / 5 AM

a A day time temperature of 85.0°F is used for the day time meteorological sets (12 noon). This is the highest mean of the extreme maximum temperatures for Champaign, IL. A night time temperature of 64.4°F is used for the night time meteorological sets (5 AM). This is the highest mean daily minimum temperature for Champaign, IL., Reference 2-41

m/s meters per second

Figure 2-13 Nearby Transportation and Facilities of Interest

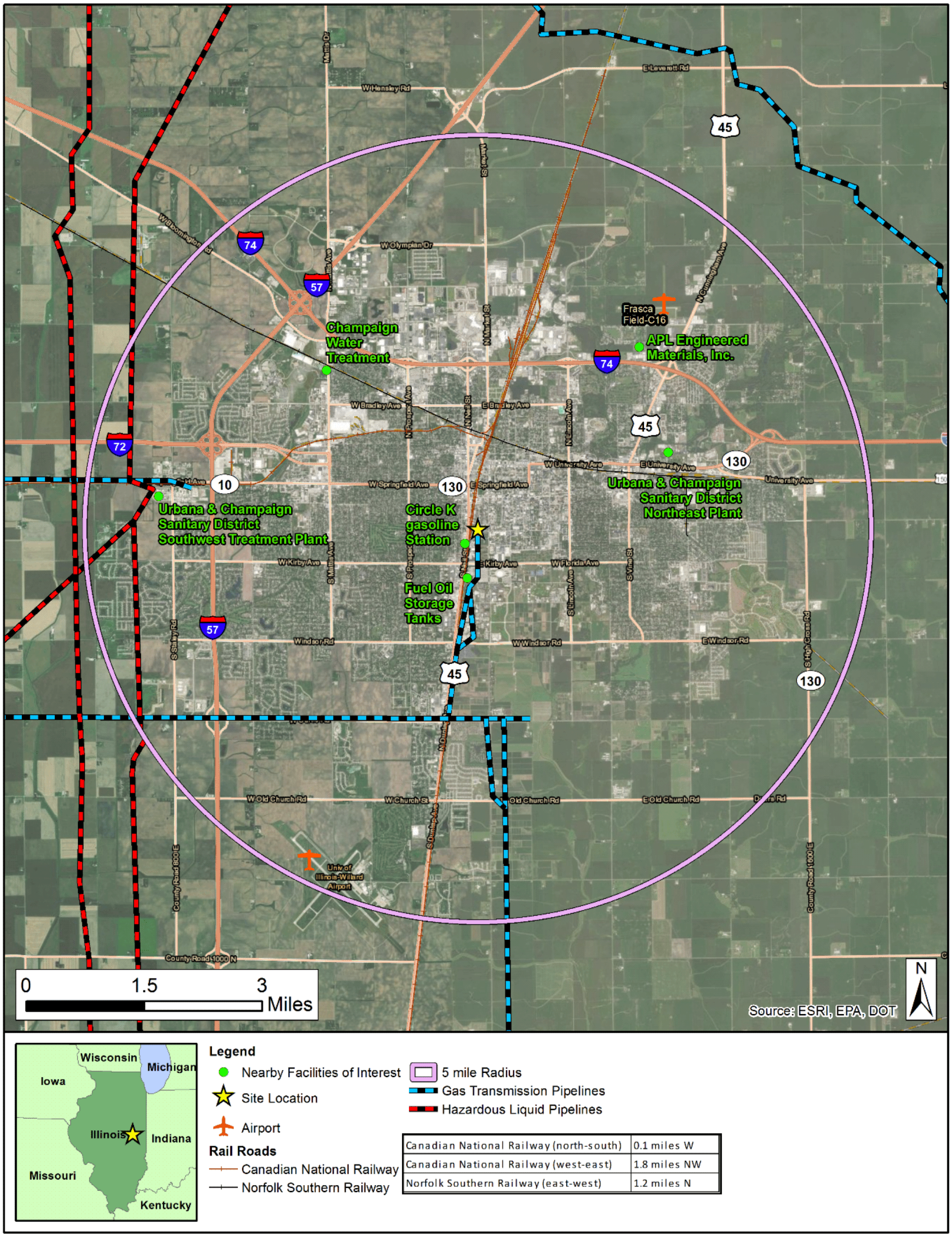


Figure 2-14 Air Traffic with a 10-miles Radius

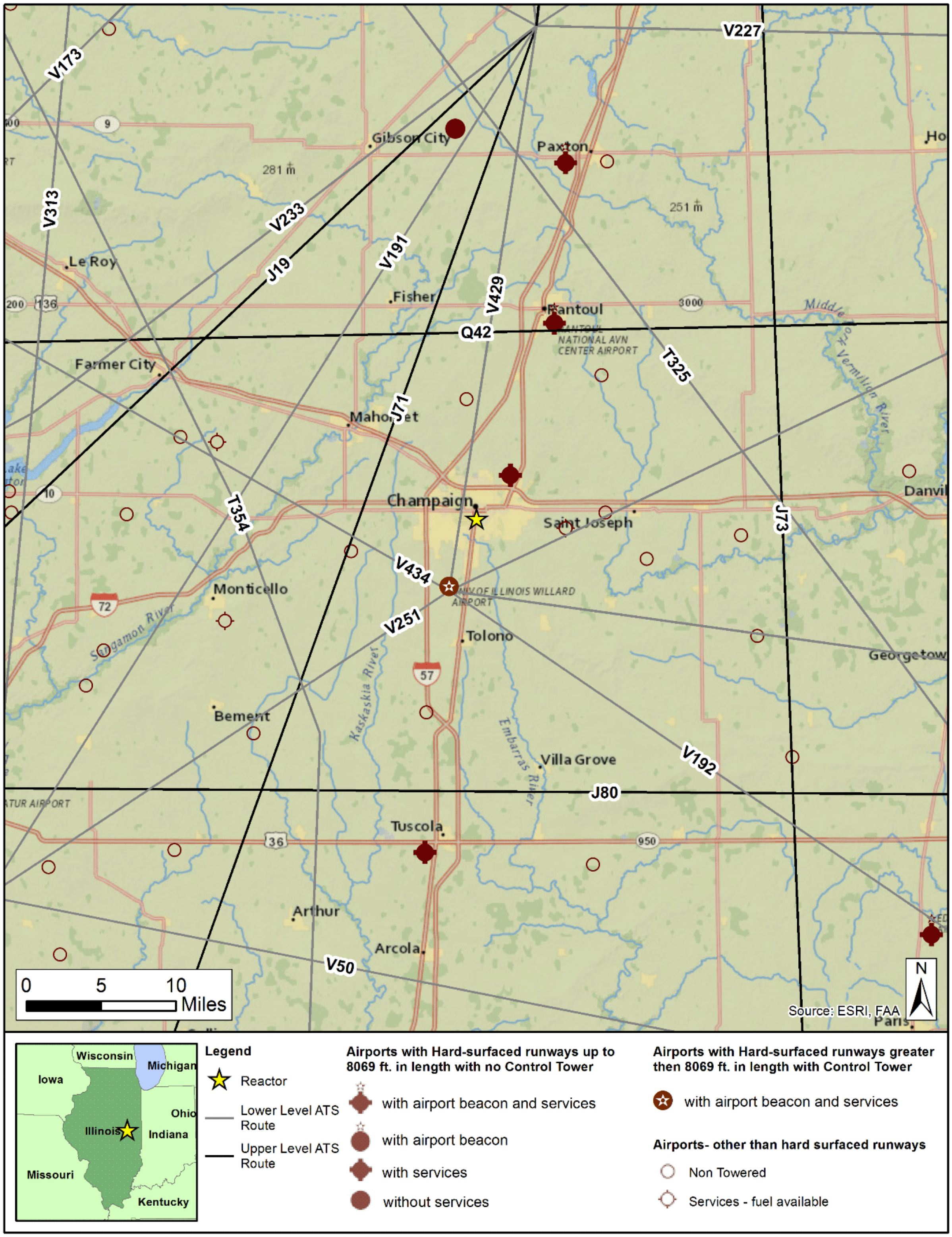


Figure 2-15 Traffic Volumes – Major Roadways

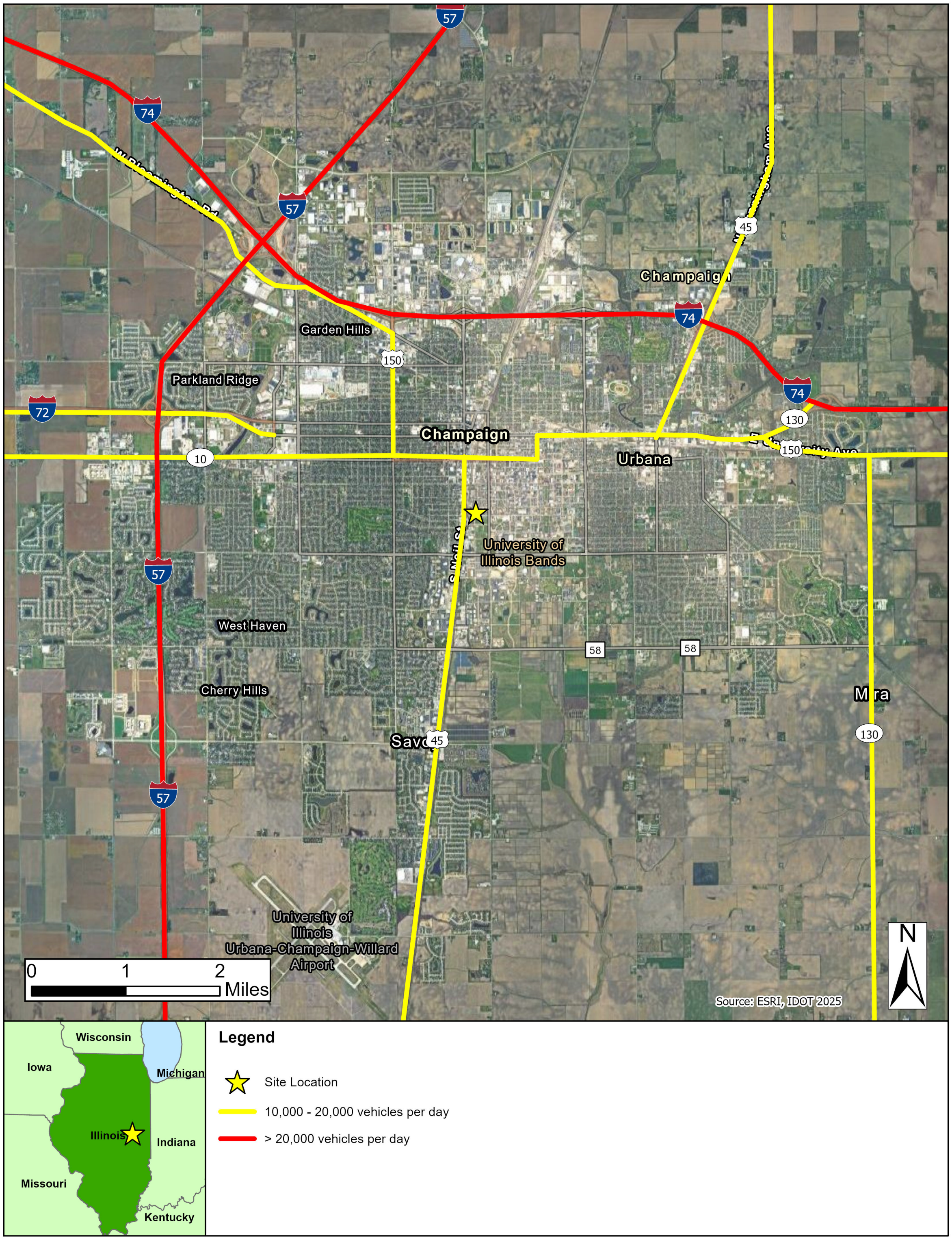


Figure 2-16 Traffic Volumes – Minor Roadways

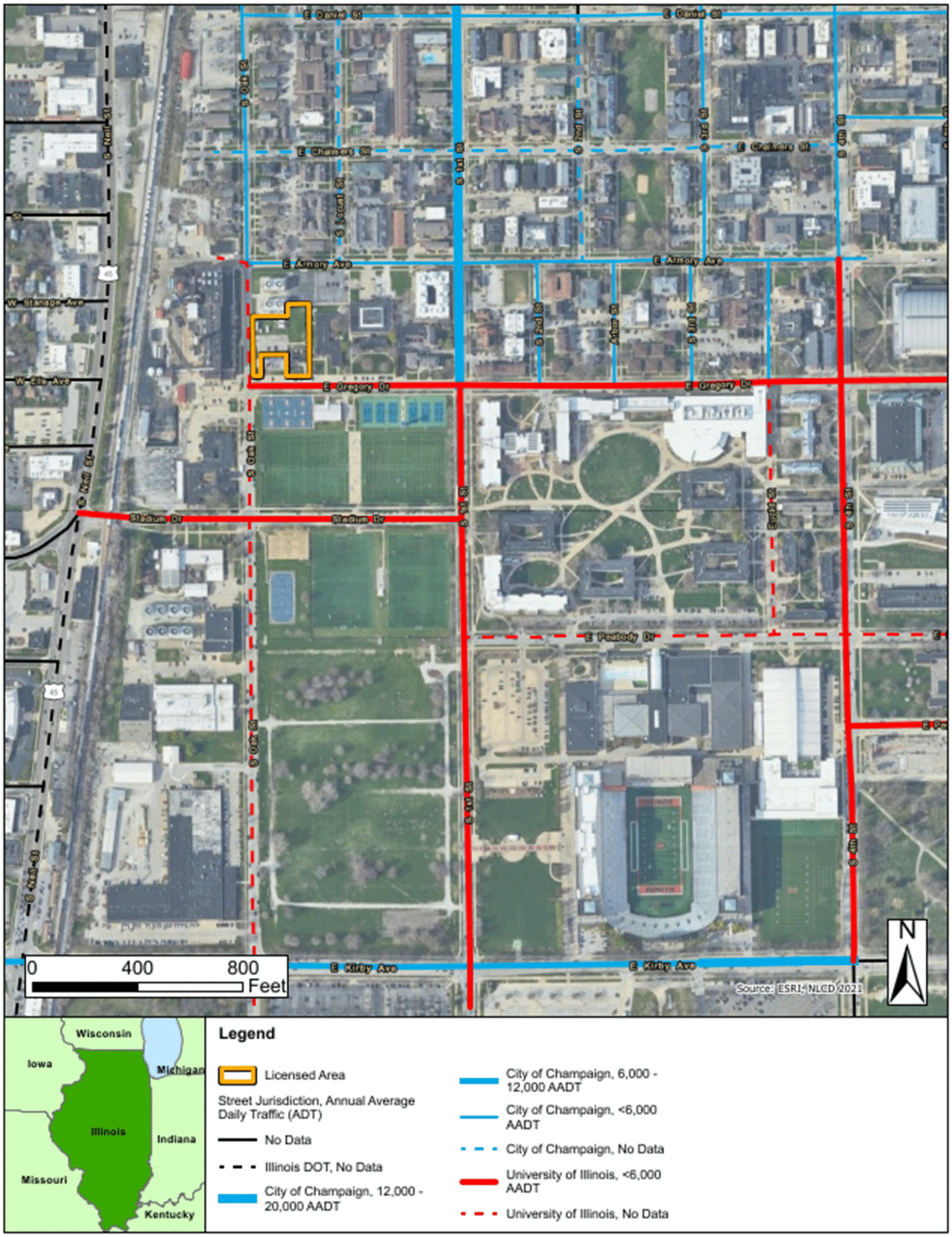
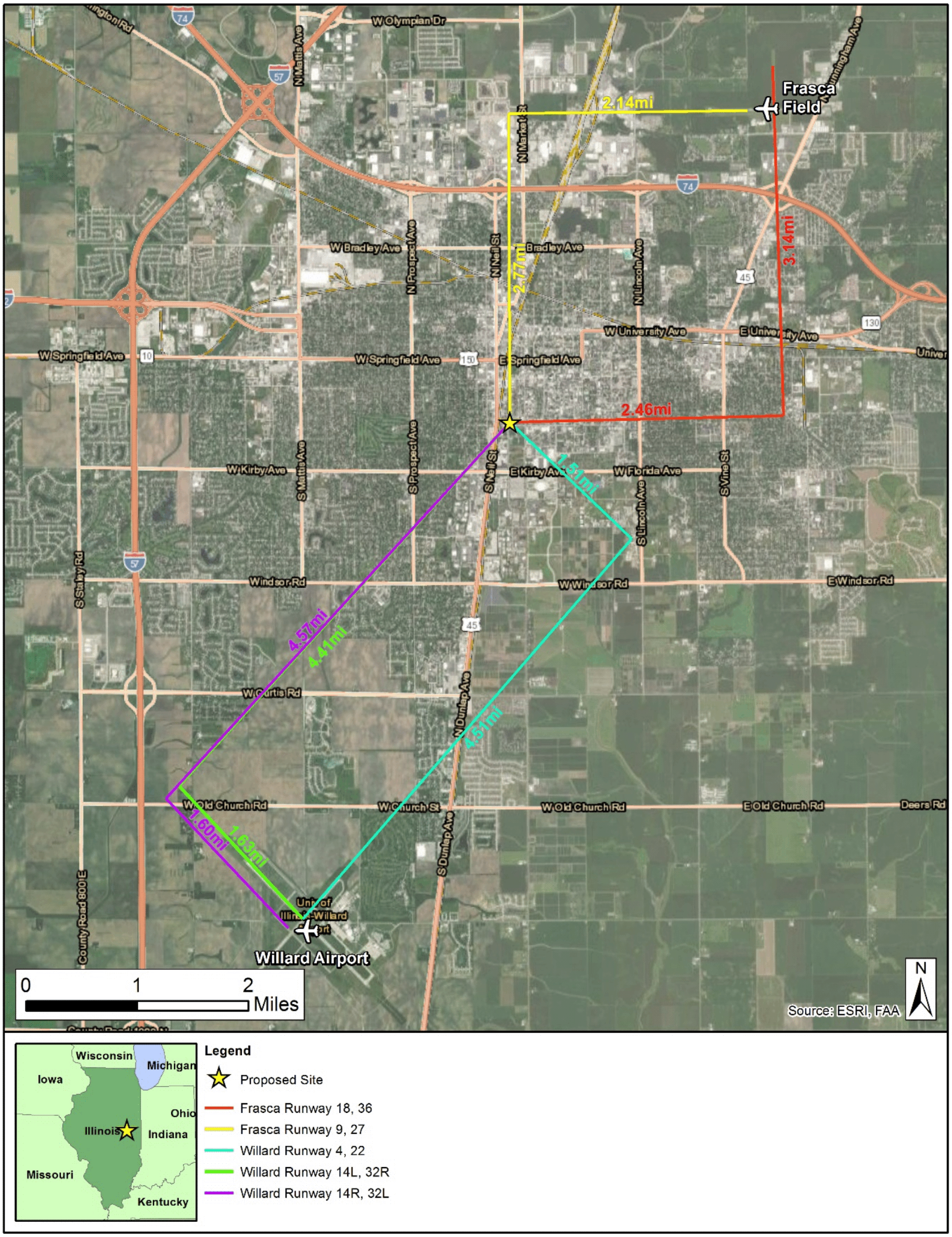


Figure 2-17 Distances to Runways



2.3 METEOROLOGY

Climate is defined as a statistical description of the weather conditions that occur over an extended period, typically several decades. In contrast, weather refers to short-term variations in atmospheric conditions, ranging from minutes to months. Data sources commonly used to analyze the climate at a specific site include weather maps, which depict regional weather phenomena at a given point in time; atlas maps that summarize long-term climate trends; records from specific monitoring stations at designated times; and long-term climatic statistics collected at these monitoring stations.

The purpose of analyzing regional climate is to gain an understanding of the local climate at the proposed reactor site within the broader context of the surrounding area's climate. Climate phenomena are examined at progressively smaller scales, focusing on more localized areas. As the analysis becomes more concentrated, some of the more distant monitoring stations initially considered in the broader analysis may be excluded. The result is a documented, systematic approach that defines the local climate within the context of the proposed reactor site, which includes the broader surrounding region.

2.3.1 Regional Climatology

The U. of I. research reactor site is located on the western edge of the U. of I. Urbana-Champaign campus in Champaign County, east-central Illinois. The campus lies approximately 80 miles (130 km) southeast of Peoria, 120 miles (190 km) west of Indianapolis and 130 miles (210 km) south of Chicago, as shown in Figure 2-18.

The region's climate is predominantly continental, characterized by cold winters, warm summers, and frequent short-term fluctuations in temperature, humidity, cloudiness, and wind direction (Reference 2-42). The flat topography of east-central Illinois allows weather systems to move across the area with minimal interference, contributing to the dynamic nature of the local climate.

The climate of Illinois is influenced by two primary factors: latitude and migratory weather systems. Latitude determines the seasonal solar input, while migratory weather systems involve interactions between polar and subtropical air masses, as well as cyclonic storms. The polar jet stream frequently resides over or near Illinois during the fall, winter, and spring, steering low-pressure systems that bring clouds, wind, and precipitation. High-pressure systems, which create periods of settled weather, typically last only a few days before being replaced by low-pressure systems (Reference 2-42). These dynamic weather patterns result in a wide variety of conditions that occur almost daily.

Urban development has a noticeable impact on the local climate. Built-up areas, such as cities, tend to retain heat due to large expanses of pavement, buildings, and industrial activities. For example, Chicago is, on average, 2 degrees Fahrenheit (°F) or 1.1 degrees Celsius, warmer than surrounding rural areas, particularly at night. Urbanizations also enhances summer precipitation downwind of developed areas and alters in humidity, cloudiness, and wind speeds and wind directions (Reference 2-42). While the research reactor is located in a university setting, the surrounding urban environment may influence localized meteorological conditions.

Area temperatures measured in Urbana-Champaign indicate warm summers and cold winters. In January, the normal daily maximum temperature is 33.5°F with a normal daily minimum temperature of about 17.9°F. In July, the normal daily maximum temperature is about 85.2°F, while the normal daily minimum temperature is about 65.2°F based on 30 years of data (1991–2020) from the National Climatic Data Center (NCDC) (Reference 2-41). Relative humidity in the region averaged 77 percent based on a 10-year period of record from the Willard Airport Local Climatological Data (September 2015 – August 2025) (Reference 2-43).

Based on the same 30 years of data (1991-2020) from the NCDC (Reference 2-41), precipitation averages about 41 inches (104 cm) annually. Late spring to early summer (May – July) is usually the wettest season, with more than 13 inches (33.0 cm), while late winter (January – March) is the driest season, with less than 10 inches (25.4 cm). Snowfall in the Urbana-Champaign area usually occurs from November through March, with an annual average snowfall of about 21 inches (53.3 cm) (Reference 2-41).

Drought is a common feature of the climate of Illinois. However, since 1965, extreme drought has only occurred twice (1988–1989 and 2012) (Reference 2-44).

The regional meteorological conditions that are relevant to the design and operating bases for the research reactor site are discussed below.

2.3.1.1 Severe Weather

Severe weather phenomena may require careful consideration in the design of safety-related structures, systems and components, and the impact of the severe weather on the plant safety needs to be evaluated. Statistics on severe weather phenomena are obtained from historical data. Most data are taken from the NCDC Storm Events Database that covers the 75-year period of 1950–2025, but even longer data periods are used for some phenomena to better capture the occurrence of rare events.

2.3.1.2 Thunderstorms

Thunderstorms account for 50-60 percent of annual precipitation and are quite common in Illinois, with an annual average of 60 storms (far northeast) to 80 storms (southwest). Nearly half of all thunderstorm days occur during the June – August period (Reference 2-42).

2.3.1.3 Hail

In Champaign County, severe hail (0.75 inches [1.9 cm] in diameter or larger) has been reported 219 times (over 134 days) between 1950–2025 (Reference 2-45). This corresponds to less than three severe hail events per year. State-wide annual average hail-days vary from 3.3 days (southwest) to less than 1.8 days (northeast) (Reference 2-42).

2.3.1.4 Lightning

The annual average number of cloud-to-ground lightning strikes per square mile ranges from 5 strikes (northeast) to more than 11 strikes (southwest) (Reference 2-42).

A nationwide study employed cloud-to-ground lightning flash data from the U.S. National Lightning Detection Network to examine temporal and spatial distributions of lightning from 1993–2018. Based on figures depicting the temporal predictions, the project site could expect a normal range of 45–54 days with thunderstorms, and averages 4 to 8 cloud-to-ground lightning flashes per square kilometer annually (Reference 2-46).

While 10-year data was not available from the NCDC, a nationwide study of 2021 data ranked Illinois 5th in the country in overall lightning counts, including both cloud-to-cloud and cloud-to-ground strikes. According to this study, Illinois had a total of 168 days in the year when lightning was detected in 2021. Champaign County was 74th overall, with 66 days when lightning was detected (Reference 2-47).

A nationwide study of 2024 lightning statistics ranked Illinois 10th in the nation for total lightning count and 13th in density, with 44.0 lightning events per square km (Reference 2-48).

2.3.1.5 Extreme Winds

Windstorms are relatively infrequent, but may occur several times a year, usually associated with thunderstorms. Moderate and occasionally strong winds sometimes accompany migrating cyclones and air mass fronts. The strong winds are usually associated with lines of thunderstorms along or ahead of cold fronts and are more probable in the late winter and spring than any other time of the year. Brief, strong gusts of wind result from downdrafts and outflow from individual thunderstorms can occur, but are generally limited to the large, intense thunderstorms that develop in the spring and summer. Estimated extreme winds are based on climatological data from Willard Airport (Reference 2-43).

Hourly averaged (scalar) wind speeds at the 10-meter level are taken every 10 seconds and averaged over the hour (Reference 2-49). Records from 1989 to 2024 are available from the Illinois Prairie Research Institute Water and Atmospheric Resources Monitoring Program. The maximum hourly averaged wind speed (gust) for the years of data analyzed (1989–2024) is 58.9 miles per hour (mph) (Reference 2-49).

For a 100-year return period, the fastest mile of wind in the project area is approximately 90 mph (Reference 2-50).

2.3.1.6 Precipitation Extremes

Historical precipitation data for the project area were obtained from several National Weather Service (NWS) sites (Reference 2-51, Reference 2-52, Reference 2-53, Reference 2-54, Reference 2-55), the State Climatologist Office for Illinois (Reference 2-43), and are summarized in Table 2-21. Based on the similarity of the maximum recorded 24-hour and monthly totals among these stations and the areal distribution of these stations around the site, the data suggests that these statistics are reasonably representative of precipitation extremes that might be expected to be observed at the research reactor site.

A drought is defined as an extended period of below-average precipitation that leads to some kind of impact (crops, water supplies, etc.). While droughts have been a familiar feature of the climate in Illinois, droughts were quite common in the first part of the 20th century and much rarer since 1965. Illinois experienced statewide extreme drought 8 times between 1895 and 1965 (Reference 2-44). Since 1965, extreme drought has occurred only twice (1988–1989 and 2012). These recent droughts were shorter and less intense than many of the droughts from the early 20th century (Reference 2-44).

Based on the most recent 26 years (2000–2025), the average annual precipitation for the Urbana-Champaign area is 40.49 inches (102.8 cm) of rain and 19.6 inches (49.8 cm) of snow (see Table 2-21). The maximum 24-hour rainfall in Illinois was 16.94 inches (43.0 cm) at Aurora on July 17–18, 1996 (Reference 2-42). The maximum monthly rainfall for the Urbana-Champaign area was about 9.16 inches (23.3 cm) (see Table 2-21).

Normal and extreme snowfall events are discussed in Section 2.3.1.11.

2.3.1.7 Tornadoes

Illinois is ranked high in terms of the number of tornados that occur throughout the state. Peak tornado frequency in Illinois is between April and June. The State averages 54 tornadoes per year based on 1991-2020 data. Notably, a record 142 tornadoes were documented in 2024 (Reference 2-56).

During the 75-year period of 1950-2025, 81 tornadoes were reported in Champaign County, of which 27 tornadoes were reported within 10 miles (16. 1 km) of the project site (Table 2-22) with intensities ranging from F0/EF0 to F3/EF3 (Reference 2-57).

Based on the tornado strike probability presented in NUREG/CR-4461, *Tornado Climatology of the Contiguous United States*, the number of tornado events from 1950 through August 2003 within a 2-degree box surrounding the research reactor site is 384. This gives an annual average of seven tornado events striking somewhere within the 2-degree box. Based on the Tornado Risk Assessment provided by the National Oceanic and Atmospheric Administration (NOAA), the number of tornado events from 1950 through 2019 within an 80 km radius (approximately the same area as the 2-degree box used in NUREG/CR-4461) of the Champaign zip code is 478. The annual average for tornado strikes for this area is also seven.

RG 1.76*, Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants*, lists the design-basis tornado parameters and the design-basis tornado-generated missile parameters required for a nuclear reactor in Illinois, which is located in Tornado Intensity Region I. Table 2-23 lists the maximum wind speed, translational speed, maximum rotational speed, radius of maximum rotational speed, pressure drop, and rate of pressure drop of a tornado that a Region I nuclear plant is required to withstand. Table 2-24 lists the maximum horizontal wind speeds that will turn a solid steel sphere 1 inch (2.54 cm) in diameter and 0.147 pounds (0.0669 kg), an automobile weighing approximately 4,000 pounds (1,810 kg), and a schedule 40 pipe approximately 6.625 inches (16.8 cm) in diameter, 15 feet (4.58 meters) long, and 287 lb (130 kg) into tornado-generated missiles.

2.3.1.8 Hurricanes

Tropical storm and hurricane winds are mainly a concern for coastal locations, as shown by the wind speed contours presented in RG 1.221, *Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants*, and NUREG/CR-7005, *Technical Basis for Regulatory Guidance on Design-Basis Hurricane Wind Speeds for Nuclear Power Plants*. Because of the rapid dissipation of hurricane winds as they move inland away from their oceanic energy source, hurricane winds should not be a concern for the research reactor site. The site is more than 600 miles (965 km) from the Atlantic Ocean and the Gulf of Mexico. The wind speed contours in RG 1.221 and NUREG/CR-7005 stop well short of the site location, with a wind speed contour of 130 mph. A review of the NCDC Storm Events Database for the period of January 1, 1950, through December 31, 2025, shows that there were no weather events that were attributed to hurricanes or tropical storms (Reference 2-58).

2.3.1.9 Winter Storm Events

The maximum reported snow depth in Urbana-Champaign-reported during the 30-year period of record was 19 inches (48.3 cm) in February 1982 (Reference 2-52). Snowfall records for stations around the research reactor site (Table 2-21) show a maximum 24-hour snowfall of 18.6 inches (47.2 cm) (January 1999) at Chicago and 13 inches (33.0 cm) (February 1914) at Lincoln (Reference 2-53, Reference 2-54).

Frost penetration depth is important for protection of water lines and other buried structural features that are subject to freeze damage. Frost depth in Champaign County is generally around 20 inches (50.1 cm), with extreme depth at about 30 inches (76.2 cm) (Reference 2-59, Reference 2-60).

2.3.1.10 Ice Storms

Estimations of regional glaze probabilities have been made by Tattleman and Gringorton (Reference 2-61). For Region II, which contains Illinois, storms with ice greater than or equal to 1 inch (2.54 cm) of ice occurred 15 times in 50 years, and storms with ice greater than or equal to 2 inches (5.1 cm) of ice occurred 3 times in 50 years. For ice storms with wind gusts greater than or equal to 44.7 mph, the estimated ice thickness is less than 1 inch (2.54 cm) for 25- and 50-year return periods, and 2.8 inches (7.1 cm) for a 100-year return period (Reference 2-61). Earlier studies (1900-1960) indicate 102 severe winter storms that produced 6 inches (15.2 cm) or more of snow in 48 hours, or glaze that covers 5,000 square miles or more of the state (Reference 2-62).

Based on the data provided in American Society of Civil Engineers (ASCE) Standard No. 7-22 (Reference 2-63), Figure 10.4-2B, the 500-year mean recurrence interval of uniform ice thickness resulting from freezing rain for Champaign County is 1.0 inch (2.54 cm), with a concurrent 3-second wind gust of 40 mph.

For glaze ice, the point probabilities for ice thicknesses greater than or equal to 0.25 inches (0.63 cm), and greater than or equal to 0.5 inches (1.3 cm), are about 0.343 and 0.28, respectively in any given year (Reference 2-61). Glaze ice thicknesses less than or equal to 0.5 inches (1.3 cm) generally results in little structural damage. However, storms which produce lesser ice thickness can present a hazard to travel in the affected areas, and when combined with strong winds, can damage above-ground utility wires.

2.3.1.11 Normal and Extreme Winter Precipitation Events

Snowpack, as used in this section, is defined as a layer of snow and/or ice on the ground surface, and is usually reported daily, in inches, by the NWS at all first-order weather stations. Historical snowpack and snowfall were developed by reviewing data from first-order NWS stations and the cooperative network.

From Figure 7.2-1A of ASCE No. 7-22 (Reference 2-63), the ground snow loads for Risk Category I for the Urbana-Champaign area are determined to be approximately 29 pounds per square foot (psf). Risk Category IV ground snow loads for Urbana-Champaign area are 56 psf.

The maximum reported snow depth for Urbana-Champaign was used to estimate the weight of the maximum historic snowpack at the research reactor site. The greatest snow depth reported since 1903 was 19 inches (48 cm) in February 1982 (Reference 2-42). The NRC's *Interim Staff Guidance (ISG) on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures (ISG-7)*, provides an algorithm (below) for converting historical maximum snowpack depth to a ground snow load:

$$L = 0.279D^{1.3} \qquad \text{Equation 2-11}$$

Where:

D is the snowpack depth in inches and L is the resulting snow load in psf.

Using the 19-inch (48-cm) snow depth for Urbana-Champaign gives a snow load of 15.3 psf for the maximum historical snowpack.

The 100-year return period snowfall event is given in data provided by the NCDC. Based on this data, the 48-hour 100-year return snowfall event for Urbana-Champaign is 15 inches (38 cm) during a January 1999 snowstorm (Reference 2-42), and 23 inches (58 cm) for Chicago during a January 1967 snowstorm (Reference 2-51). The equation below from ISG-7 was used to determine the snow load due to the 48-hour 100-year return period snowfall event and the historical maximum snowfall event.

$$L = 0.15 \times S \times 5.2 \qquad \text{Equation 2-12}$$

Where:

L is the snow load in pounds per square foot (psf)

S is the snowfall depth in inches.

Using the maximum 100-year return snowfall event of 15 inches (38 cm) results in a snow load of 11.7 psf. Using a 23-inch (58-cm) historical maximum snowfall event for a 48-hour period results in a snow load of 17.9 psf.

The Normal Winter Precipitation Event, defined as the maximum ground-level weight (in psf) of the (1) 100-year snowpack (snow cover), (2) historical snowpack (snow cover), (3) 100-year return 2-day snowfall event, or (4) historical maximum 2-day snowfall event, is determined to be 17.9 psf. The Extreme Frozen Winter Precipitation Event, defined as the maximum of the (1) 100-year return 2-day snowfall event or (2) historical maximum 2-day snowfall event, is also determined to be 17.9 psf.

From Hydrometeorological Report HMR-53 (Reference 2-64) the 48-hour Probable Maximum Winter Precipitation (PMWP) (January through March) for a 10 square mile area is estimated to be 23.5 inches (59.7 cm) by logarithmic interpolation. The March PMWP was utilized because the historically deepest snowpack occurred in March 1993. The 48-hour PMWP is equivalent to the Extreme Liquid Winter Precipitation Event.

2.3.1.12 Design Basis Dry- and Wet-Bulb Temperatures

This section provides ambient temperature and humidity statistics to establish heat loads for the design of normal plant heat sink systems, post-accident containment heat removal systems, and plant heating, ventilating, and air conditioning systems. The following parameters have been calculated:

- Maximum dry-bulb temperatures at 0.4 percent, 1 percent, and 2 percent annual exceedance levels
- Mean coincident wet-bulb temperatures at 0.4 percent, 1 percent, and 2 percent annual exceedance levels
- Maximum non-coincident wet-bulb temperature at 0.4 percent annual exceedance levels
- Minimum dry-bulb temperature at 0.4 percent, 1 percent, and 2 percent annual exceedance levels
- 50-year return maximum dry-bulb, mean coincident wet-bulb, maximum non-coincident wet-bulb, and minimum dry-bulb temperatures

Annual exceedance and 50-year maximum values for dry-bulb and wet-bulb temperatures of 0.4 percent, 1 percent, and 2 percent will be used in the design basis for safety-related and non-safety-related ventilation and heat removal system design for the U. of I. research reactor site.

Climate for 2025 were found in the American Society of Heating, Refrigerating, and Air Conditioning Engineers Handbook - Fundamentals (Reference 2-65) for Champaign Willard Airport. This data set contains measurements of dry-bulb and dewpoint temperature records, amongst several other meteorological variables. This data was used to calculate the various exceedance temperatures. Results of the ambient design temperature analysis are presented in Table 2-25, Table 2-26, and Table 2-27. The monthly design dry bulb temperatures with mean coincident wet bulb temperatures and the monthly design wet bulb temperatures are presented in Table 2-28 and Table 2-29, for annual exceedances listed above.

2.3.1.13 Meteorological Data for Evaluating Ultimate Heat Sink

The U. of I. research reactor does not rely on an external water source as its ultimate heat sink but rather uses direct to air heat rejection. Therefore, considerations of evaporation and drift loss of water, minimum water cooling, and the potential for water freezing in a ultimate heat sink water storage facility are not applicable.

2.3.1.14 Climate Change

While climatic conditions change over time, such changes are cyclical in nature on various times and spatial scales. The timing, magnitude, relative contributions to, and implications of these changes are generally more speculative, even for specific areas or locations. Furthermore, the most extreme projected changes are for time scales much longer than the design life of the U. of I. research reactor.

Projected changes are generally small compared to natural variation. General predictions of global or United States climatic changes expected during the period of reactor operation are uncertain and are only applicable on a macroclimatic scale. Because the maximum data span available was used in the severe weather analysis, accurate severe weather phenomena have been provided based on best-available historical data. Projections of future severe weather conditions at the research reactor site are highly uncertain at best, based on current understanding and modeling of global climate change. Predictions provided by the U.S. Geological Survey (Reference 2-66) are somewhat consistent for 2025 through 2049, showing a maximum summer month (July) maximum temperature increase from 88°F to 92°F with a standard deviation of less than 1°F for most months. However, the models get less consistent for later timeframes. For example, one model (the BNU-ESM model) gives a July maximum temperature increases from approximately 87°F to 94°F, with a standard deviation of approximately 3°F over the period of 2050 through 2074.

Using monitoring data from the NOAA Centers for Environmental Information over the past 120 years, The Illinois State Climatologist Office, along with the Prairie Research Institute, prepared a factsheet on climate change. Their research indicates the average daily temperature in Illinois has increased by 1 to 2°F, overnight minimum temperatures have increased more than daytime maximum temperatures, and there has been seasonal variation in warming. Winter and spring temperatures in Illinois have increased by 2 to 3°F, while summer temperatures have increased very little since the beginning of 20th century (Reference 2-67). Additionally, precipitation has increased by 5 inches (12.7 cm) over the last 120 years, which equates to a 12-15 percent increase in total annual precipitation. Precipitation has also gotten more intense as the number of 2-inch (5.1-cm) rain days has increased by 40 percent.

By the end of the 21st century, average daily temperatures are projected to increase between 4 and 9°F under the lower emissions scenario, and between 8 and 14°F under the higher emissions scenario. These increases are expected to coincide with increased risk of extreme high temperatures in Illinois, and reduced risk of extreme cold temperatures (Reference 2-67).

Although long-term climate trends from multiple sources indicate an increase in average temperatures and a heightened possibility of extreme precipitation events through the end of the 21st century, the anticipated operational period for the U. of I. research reactor is only a fraction of this projection period. Considering the reactor's design life of 40 years, along with an additional decommissioning period of 10 years, the total facility lifetime is projected to be approximately 50 years. Therefore, planning based on recent meteorological statistics discussed elsewhere in this section accounts for only slightly higher temperatures (1-2°F on average) and a somewhat increased likelihood of extreme weather events.

2.3.2 Local Meteorology

2.3.2.1 Local Meteorological Data Overview

The U. of I. research reactor project site is located on the western edge of the U. of I. campus in Champaign County in east-central Illinois. The Illinois State Water Survey (ISWS) is a division of the Prairie Research Institute of Illinois, headquartered in Champaign on the campus of U. of I. It is the primary agency in Illinois concerned with the measurement and evaluation of water resources. A cornerstone in helping the ISWS achieve its mission is a statewide network of data-gathering sites called the Water and Atmospheric Resources Monitoring Network (WARM). The WARM Network routinely collects and compiles data to facilitate both comprehensive assessments of Illinois water and atmospheric resources and provide timely information on these resources to users both in Illinois and across the United States. Network data collection began in the 1950s. Newer automated data-gathering platforms began operation in the late 1980s. Evolution of the network continues, as increased automation of data collection and more real-time retrieval and dissemination of data and products are either being achieved or are proposed (Reference 2-68). The WARM monitoring location in Champaign County is the U. of I. meteorological tower. U. of I. maintains this meteorological monitoring station about 1.4 mi. south of the project site. The relevant instrumentation and operation of the U. of I. station are discussed in Subsection 2.3.3.

Wind speed and direction data are collected at the 33-foot (10-meter) level. Temperature, relative humidity, and precipitation are measured at 6.6 feet (2 meters). Barometric pressure and solar radiation are also measured at the tower. Instrument calibrations are managed by the ISWS (Reference 2-69).

The Champaign station is in a fenced area south of Building 6 at the ISWS Research Center on the southwest edge of the University of Illinois campus (Figure 2-19). The station is surrounded by buildings and deciduous trees. To the north and east, mature deciduous trees are located within 131 feet (40 meters) of the tower. The ISWS buildings are north of the tower and extend to the west for 410 feet (125 meters). Beyond the buildings, the city of Champaign extends for 6.2 miles (10 km) to the north and west. The city of Urbana extends 4.4 to 5.0 miles (7 to 8 km) to the northeast and east. From the north in a clockwise direction to the east, there are several three-story apartment buildings within 328 feet (100 meters) of the tower. Mature evergreen trees are located within 164 feet (50 meters) east of the tower. Behind the evergreens, several deciduous trees greater than 33 feet (10 meters) tall extend from 260 to 390 feet (80 to 120 meters) to the east of the tower. Beyond the deciduous trees is open farmland. From the southeast to the southwest, the nearest obstruction is a row of deciduous trees approximately 10 meters apart and 164 feet (50 meters) from the tower. Beyond the row of trees is open land. A building approximately 33 feet (10 meters) tall is located 1,640 feet (500 meters) south of the tower, and another row of mature deciduous and evergreen trees is situated approximately 165 meters to the west of the tower (Reference 2-70).

The topography of an area can influence the local climate. Illinois is generally flat, lying wholly within the Central Plains, with gentle sloping hills and shallow river valleys as shown by Figure 2-20 for areas near the research reactor site, and in Figure 2-21 out to 50 miles (80 km) from the site. The till plains were carved and leveled by glaciers, allowing a similar climate to prevail over the entire area. The climate is typically continental with frequent short fluctuations in temperature, humidity, cloudiness, and wind direction. Weather systems create a wide variety of weather conditions because of varying air masses and passing storm system. The site is located at an elevation of approximately 765 feet (233 meters) above mean sea-level. Figure 2-22 through Figure 2-29 show the elevation profiles within 50 miles (80 km) of the site in each of eight compass directions (at 45-degree intervals).

Meteorological measurements from the U. of I. station were reviewed. The tower is located approximately 1.44 miles (2.3 km) south of the research reactor site (Figure 2-19). Figure 2-30 through Figure 2-33 provide data from the U. of I. station. All wind roses have a similar pattern of winds, with the prevailing winds coming from the west. Further south, where there is less urban influence, the prevailing winds at Willard Airport are from the south (Figure 2-34).

2.3.2.2 Normal and Extreme Values of Meteorological Parameters

Long-term temperature and wind data from regional stations were reviewed in Subsection 2.3.1 to determine if data collected locally near the site are consistent with regional conditions, both spatially and over time. These data show meteorological characteristics of the site have not changed significantly over time and are not expected to change over the life span of the project.

Comparing data from the U. of I. meteorological tower helps to determine if the site is consistent with regional conditions. Data were examined for the 5-year period of calendar years 2021 to 2025. The local tower's average wind speeds are generally lower than those at the regional airport due to the nearby tree canopy obstructions. These comparisons indicate that, for these variables, data from the site is consistent with overall meteorological conditions in the Urbana-Champaign area.

2.3.2.3 Winds

Joint frequency distributions of lower tower levels for wind direction (vector), average (scalar) wind speed, and for stability class are presented in Subsection 2.3.5. The tower data for the project area are presented as wind roses in Figure 2-30 through Figure 2-33. These are based on five years of data (2021–2025). A wind rose for Willard Airport, based on approximately 29 years of data (1996–2025), is presented in Figure 2-34.

Wind speeds at the U. of I. meteorological tower near the research reactor site (Table 2-30) during 2021–2025 were generally light, with an average 33-foot (10-meter) speed of 3.99 mph (Table 2-31). The maximum hourly average (scalar) wind speed was 23.68 mph. The combination of high pressure associated with the Alberta-Low anticyclonic circulation generally results in light wind speeds, with average surface wind speeds for the site being less than or equal to 5 mph. The site is surrounded by generally flat terrain. The local wind patterns are influenced by the generally flat terrain and local obstacles and hence better represented by the U. of I. meteorological tower than those at the unobstructed regional airport.

Generally, the longer the winds blow in the same direction, the lower the dilution potential because effluent is not dispersing significantly from the persistent wind sector. Wind direction persistence is an indicator of the duration of atmospheric transport from a single sector (same sector, 22.5 degrees wide), three adjoining sectors (± 1 sector, 67.5 degrees), and five adjoining sectors (± 2 sectors, 112.5 degrees). For the site (Table 2-32), the maximum persistence at 32.8 feet (10 meters) for the 2021-2025 time period is 55 hours from N and NNE for the same sector, 112 hours from N-NE for ± 1 sector, and 210 hours from NNW-ENE for ± 2 sectors.

The wind data show a consistent pattern of wind directions with predominant winds from the W-SW, with a second maximum of persistent winds from the SE and NE. There is seasonal variation in this pattern (Figure 2-31). There is also a diurnal pattern with the winds. During the day (Figure 2-32) the winds show the W-SW patterns of predominant winds. During the night (Figure 2-33) the winds show the patterns of W-SW and NE patterns of predominant winds.

2.3.2.4 Air Temperature

Temperature data for Urbana Champaign are presented in Table 2-33. Normal dry bulb temperatures have ranged from the upper 20s (°F) in the winter to the low 70s in the summer. Normal daily maximum temperatures ranged from about 33°F in mid-winter to about 85°F in mid-summer. The normal daily minimum temperatures ranged from about 18°F in mid-winter to about 65°F in mid-summer. The extreme daily maxima recorded was 109°F (July 1954), while the extreme daily minima was -25°F and recorded on four occasions (February 1899 and 1905, January 1994 and 1999).

Temperatures measured by the ISWS for 2021–2025 at the U. of I. WARM meteorological tower are presented in Table 2-34. This data shows a greater variance in winter daily averages. The average monthly maximum temperatures ranged from about 50°F in mid-winter to about 95°F in mid-summer. The average monthly minimum temperatures ranged from about -1°F in mid-winter to about 58°F in mid-summer. A maximum temperature of 98.6°F and a minimum temperature of -13.5°F were recorded over the 5-year period.

2.3.2.5 Atmospheric Moisture

Long-term relative humidity and absolute humidity data collected at the U. of I. WARM meteorological tower are presented in Table 2-35. Short-term humidity data based on measurements at the U. of I. meteorological tower are summarized in Table 2-36. Site data are comparable to the long-term data. The humidity is a little higher for the 2021–2025 period collected at the U. of I. tower than for the longer-term data periods.

2.3.2.6 Precipitation

2.3.2.6.1 Rain

Hourly precipitation observations are available from the Urbana NWS station (less than 1 mile [0.61 km] northwest of the site). The long-term observations from the precipitation data from Urbana (Reference 2-72) are presented in Table 2-37. Precipitation falls an average of about 74 days per year, and the normal annual precipitation is nearly 38 inches (97 cm). The maximum monthly rainfall has ranged from about 6 inches (15 cm) to almost 14 inches (36 cm). The minimum monthly amount was a trace in November 1904. The maximum in 24 hours was 5.32 inches (13.5 cm) in August 1993. Precipitation is uniformly distributed throughout the year. May to August are normally the wettest months of the year.

Precipitation data from the U. of I. WARM meteorological tower (Table 2-38) indicate more than normal precipitation from in 2021 and 2024 and below normal precipitation in 2022, 2023 and 2025. Maximum rainfall, estimated by statistical analysis of regional precipitation data, is given in Table 2-39 for return periods of 1 to 100 years, and for rainfall durations from 5 minutes to 10 days. These data were taken from the NOAA Atlas 14 website (Reference 2-73).

Probable Maximum Precipitation (PMP), sometimes called maximum possible precipitation, for a given area and duration is the depth that is expected to possibly be reached, but not exceeded, based on historical meteorological observations. For the research reactor site area, using a 100-year return period, the PMP for 6, 12, 24, and 48 hours is 5.16, 5.87, 6.89, and 7.72 inches (13.1, 14.9, 17.5, and 19.8 cm) respectively (see Table 2-39). Approximately 47 thunderstorms occur in a typical year (Reference 2-74). Thunderstorm activity is most predominant in the spring and summer seasons, and the maximum frequency of thunderstorm days is normally in June (Table 2-37).

2.3.2.6.2 Snow

Daily snowfall observations are available from the Urbana NWS station. Snowfall data are summarized in Table 2-40. Normal annual snowfall has ranged from about 0.9 to 6.5 inches (2.3 to 16.5 cm) during winter months. The 24-hour maximum snowfall was 14 inches (35.6 cm) in 1904.

2.3.2.6.3 Precipitation Wind Roses

Figure 2-35 shows composite 2021–2025 precipitation and wind directions (vector) data from the U. of I. tower. Precipitation is most often associated with wind directions from the north-northeast, corresponding to the predominant wind flow direction sectors. There is a secondary maximum with wind directions from the south.

2.3.2.7 Fog

Fog data for Willard Champaign Urbana Airport are presented in Table 2-41. These data indicate that heavy fog (visibility less than or equal to 0.25 miles [400 meters]) occurs about 30 days per year, with the winter normally the foggiest season.

2.3.2.8 Atmospheric Stability

The frequency of occurrence of Pasquill (classes A-G) atmospheric stability classes is based on the Turner Method which requires wind speed, solar altitude, total cloud cover, and ceiling height. The calculation of atmospheric stability for the research reactor site is based on five years (2020–2024) of observational data from Willard Airport.

RG 1.23 specifies that the preferred method for estimating P-G stability class is the temperature gradient method (ΔT). However, RG 1.23 also suggests *"Alternative methods may be used to classify atmospheric stability for licensing purposes if appropriate justification is provided."* Using the ΔT method requires precise measurements of temperature differential between two levels on the same meteorological towers spaced approximately 50 meters apart. The ΔT measurement is typically only available for a customized site-specific measurement program. For small nuclear reactors and/or research reactors, the site-specific measurement programs are not typically required creating a need to approximate Pasquill stability class using an alternative approach.

Turner's method for estimating Pasquill stability class does not require the precise ΔT measurement, rather uses routinely available meteorological measurements such as wind speed and net radiation index which can be quantified using solar altitude, cloud cover and ceiling height. All these parameters can be obtained or quantified for a variety of meteorological measurement sites across the United States.

The frequency of occurrence of each stability class calculated using data from the U. of I. WARM meteorological tower and the Turner Method is presented in Table 2-42. The atmosphere, for the 5 years analyzed, appears to be neutral stability (class D) about 52 percent of the time. The stable hours (classes E, F, and G) occur 32 percent of the time, while unstable hours (classes A, B, and C) occur approximately 16 percent of the time.

2.3.2.9 Inversion Persistence

An atmospheric in version is a measurement of temperature increase as a function of height from at least two or more thermometers placed at different levels on the measurement tower. The data available from the U. of I. Station only include one temperature measurement level. There is no ability to quantify both presence and, thus, persistence of a temperature inversion using the data from the U. of I. Station.

2.3.2.10 Mixing Heights

Holzworth (Reference 2-76) provides estimated monthly mean maximum heights for Peoria, IL (the NWS upper air site closest to the research reactor site). Seasonal and annual estimates of rural mixing heights for the site are as follows:

- Winter (December, January, February) – 392 meters (morning), 594 meters (afternoon)
- Spring (March, April, May) – 431 meters (morning), 1,443 meters (afternoon)
- Summer (June, July, August) – 305 meters (morning), 1,532 meters (afternoon)
- Autumn (September, October, November) – 321 meters (morning), 1,104 meters (afternoon)
- Annual – 362 meters (morning), 1,168 meters (afternoon)

Additional discussion of atmospheric stability is referenced in Subsection 2.3.2.8.

2.3.2.11 Potential Influence of the Plant and Its Facilities on Local Meteorology

The U. of I. research reactor systems have very limited potential to noticeably affect local meteorology. The research reactor utilizes air-cooling as the primary heat sink, which limits emission of water droplets or water vapor or aerosol. The decay heat removal system utilizes low-pressure evaporative cooling, which removes less than 0.03 percent of the total reactor power through saturated steam, so while there will be some steam plumes due to heat rejection exhaust, these will be hot exhaust streams that will rapidly evaporate when mixed with ambient air. There will be some minor air quality and visibility impacts on local air quality during construction, although the impacts will be very localized due to near-ground-level releases of non-radioactive particulate related to construction activities.

2.3.2.12 Local Meteorological Conditions for Design and Operating Bases

The meteorological conditions for the design and operational bases are provided in Subsection 2.3.1.

2.3.3 Meteorological Monitoring Program

2.3.3.1 U. of I. Meteorological Measurements

Meteorological monitoring and measurements are taken at the U. of I. meteorological tower. The tower is 32. 8 feet (10 meters) tall, about 1.44 miles (2.32 km) from the research reactor site with no intervening terrain. A photo of the tower is provided in Figure 2-36. A 5-year period of full calendar years from January 1, 2021–December 31, 2025, is selected for characterization of the wind patterns and for short-term modeling as the representative meteorological input data. Wind roses for the 32.8-foot (10-meter) tower for 2021–2025 are provided in Figure 2-30.

Instrumentation on the tower consists of the following, list with the height of the instruments above the ground:

- EE 181, air and relative humidity — 6.6 feet / 2 meters
- RM Young 05103, wind speed and direction — 33 feet / 10 meters
- Epply 8-48, solar radiation — 33 feet / 10 meters
- Ott Pluvio, Precipitation — 3.3 feet / 1 meter

The WARM/Illinois Climate Network program maintains these instruments and they are calibrated before being installed. Once installed, the instruments are verified periodically through daily validation checks and field visits. (Reference 2-77)

2.3.4 Short-Term Atmospheric Dispersion Modeling for Accidental Releases

Details regarding the source term determination for the dispersion modeling, the modeling inputs, and the interpretation of the modeling results will be provided in the Final Safety Analysis Report.

2.3.5 Long-Term Atmospheric Dispersion Estimates for Routine Releases

For routine airborne releases, the concentration of radioactive material in the surrounding region depends on the amount of effluent released, the height of the release, the momentum and buoyancy of the emitted plume, the wind speed, atmospheric stability, airflow patterns of the site, and various effluent removal mechanisms. Annual average relative concentration, X/Q, and annual average relative deposition, D/Q, for gaseous effluent routine releases were calculated for the U. of I. research reactor site. This section describes the development of long-term diffusion and deposition estimates.

2.3.5.1 Purpose and Background

As required by 10 CFR 50 and aligned with 10 CFR 100, estimates of atmospheric dispersion, expressed as X/Q, and relative deposition values, D/Q, were calculated for routine releases from the U. of I. research reactor site at long-term (annual) time intervals for the Licensed Area boundary, at points of maximum individual exposure, and at points within a radial grid of sixteen 22.5-degree sectors extending to a distance of 50 miles (80 km). The XOQDOQ-82 (XOQDOQ) modeling program, described in NUREG-2919, is the NRC-recommended dispersion model. XOQDOQ implements the assumptions outlined in RG 1.111, Rev. 1.

Using joint frequency distributions (JFD) of wind direction, wind speed, and atmospheric stability class, XOQDOQ provides annual average X/Q and D/Q calculations at the required distances and sectors. Radioactive decay and dry deposition are considered, and a straight-line Gaussian trajectory is modeled between the point of release and all distances for which X/Q values are calculated.

2.3.5.2 Calculation Methodology and Assumptions

RG 1.206, Rev. 1, states that the Applicant should provide meteorological data from at least two consecutive annual cycles for calculating the short-term and long-term atmospheric dispersion estimates. Inputs used in the XOQDOQ model are listed in Table 2-43.

Meteorological data from the U. of I. meteorological tower covering a 3-year period from January 1, 2021 through December 31, 2023 was used in the development of the JFD of wind direction, speed, and atmospheric stability class for long-term atmospheric dispersion analysis. This subset of meteorological data was chosen due to its representativeness relative to the meteorological data from nearby Willard Airport. The Willard Airport data is used to supplement missing required data in the development of meteorological inputs for other atmospheric dispersion models (e.g. AERMOD) considered in this assessment. Meteorological data needed for the X/Q and D/Q calculations in XOQDOQ included average (scalar) wind speed, wind direction, and atmospheric stability (Table 2-43 through Table 2-51). Seven wind speed categories (including calms) were defined in the JFDs.

Using the JFDs, XOQDOQ provides the X/Q values as function of direction for various time periods at points within a radial grid of sixteen 22.5-degree sectors extending to 50 miles (80 km) and at the site boundary. To produce conservative estimates of ground level concentrations and deposition, no building wake credit was used in the XOQDOQ model (e.g., cross-sectional area and building height were both set to zero). Emissions from the reactor were also considered as a ground-level release and no corrections for open terrain recirculation were applied given the flat terrain in the area.

Consistent with RG 1.111. Rev. 1, radioactive decay and deposition were considered. For conservative estimates of radioactive decay, an overall half-life of 2.26 days for short-lived noble gases, a half-life of 101 days for long-lived noble gases (effectively inert in the dispersion models), and a half-life of 8 days for all iodines are acceptable for releases to the atmosphere. Only the effects of dry deposition are considered from a plume depletion perspective. The calculations which consider no deposition are identified in the output as undepleted, while those that consider dry deposition are identified as depleted.

2.3.5.3 Advanced X/Q and D/Q Modeling

In addition to analysis with XOQDOQ, dispersion and deposition calculations were carried out with a more advanced atmospheric dispersion modeling system, AERMOD, and the corresponding suite of preprocessors for terrain, building effects, and meteorology. AERMOD is US EPA's recommended dispersion model for receptors within 31 miles (50 km) of a modeled source. AERMOD can account for the effect on plume dispersion associated with building-induced turbulence and is also capable of handling the source geometry, terrain, and dispersion environment associated with the reactor source.

It is expected that in the case of a small reactor with close proximity to surrounding buildings, AERMOD may produce higher concentrations relative to XOQDOQ. For this analysis, AERMOD was set up to calculate dispersion predictions analogous to those predicted by XOQDOQ. To do this, an analogous radial receptor grid (i.e. 22 distances at 16 different directions from the source) was established from the center of the proposed reactor facility, and four different model runs were performed to capture the dynamics of radioactive decay and dry deposition. AERMOD provides the option to impose exponential decay of the pollutant being modeled. The user can specify a half-life to define the decay for the three decay scenarios required in equivalent XOQDOQ modeling: inert, 2.26-day half-life, and 8-day half-life. Dry deposition of particles can also be considered by defining a mean particle diameter associated with emissions as well as a fraction of particles that can be described as “fine” (< 2.5 micrometers [µm] in diameter). For the 8-day half-life with depletion case, deposition parameters for radionuclide emissions were taken from the Argonne National Lab (ANL) report (Reference 2-78) which is the recommended source in the user's guide for the latest version of AERMOD (Reference 2-79). For radionuclides, the ANL report recommends a fine particle mass of 80% and mean particle diameter of 0.4 µm. To produce dispersion results analogous to those achieved through use of XOQDOQ, the emission rate from the source in AERMOD is set to 1 gram per second. This allows the dispersion impacts to be scaled with actual emissions to perform dose calculations.

Both the AERMOD and XOQDOQ models used a single ground-level point source. In AERMOD the stack height was conservatively set to 0 meters, the temperature of the release was set to ambient, the velocity was set to 0.001 meters per second to minimize the momentum of the release and produce conservative results.

2.3.5.4 Summary of Results and Conclusions

Consistent with RG 1.111, Rev. 1, the long-term, routine-release X/Q and D/Q values were evaluated with the XODOQ model at the site boundary and at receptor points within a radial grid of sixteen 22.5-degree sectors extending to a distance of 50 miles (80 km) from the site. A set of data points were located within each sector at increments of 0.25 miles (0.4 km) to a distance of 1 mile (1.61 km) from the site; at increments of 0.5 miles (0.8 km) from a distance of 1 mile (1.61 km) to 5 miles (8 km); at increments of 2.5 miles (4 km) from a distance of 5 miles (8 km) to 10 miles (16.1 km); and at increments of 5 miles (8 km) from a distance of 10 miles (16.1 km) to a distance of 50 miles (80 km). Estimates of X/Q (undecayed, decayed, and decayed with depletion) and D/Q (undecayed) are provided at each of these points. To obtain conservative results, X/Q values were calculated using AERMOD while D/Q values were calculated using XOQDOQ. This is based on the observation that, for a ground-level release, AERMOD produces higher X/Q values at nearly all distances and directions. The deposition module in XOQDOQ however produces higher D/Q values than those calculated with AERMOD. The results of the modeling analysis, based on three years of onsite meteorological data, are presented in (Table 2-52 through Table 2-59). These X/Q and D/Q results will serve as key inputs into the calculation of predicted radiation doses from operations at the U. of I. research reactor site.

Table 2-21 Regional Precipitation Extremes

Station	Period of Record (years)	Normal Annual Rainfall (inches)	Max 24-hour Rainfall (inches)	Max Monthly Rainfall (inches)	Normal Annual Snowfall (inches)	Max 24-hour Snowfall (inches)	Max Monthly Snowfall (inches)
Chicago NWS Station	139[a] (1886–2025)	-	9.35 (Aug 1987)	17.10 (Aug 1987)	-	18.6 (Jan 1999)	42.5 (Jan 1918)
	26[b] (2000–2025)	37.90	-	13.63 (Sep 2008)	37.8	-	33.7 (Jan 2014)
Lincoln, IL	120[c] (1905–2025)	-	5.22 (May 1914)	15.93 (July 1992)	-	13.0 (Feb 1914)	24.0 (Feb 1914)
	26 [d] (2000–2025)	38.88	-	11.0 (Jul 2008)	17.6	-	20.3 (Feb 2022)
Urbana-Champaign	29 [e] (1981-2010)	41.38	-	-	23.2	15.0 (Jan 1999)	-
	26 [f] (2000–2025)	40.49	-	9.16 (Jun 2015)	19.6	-	20.4 (Dec 2010)

[a] Source: Reference 2-51

[b] Source: Reference 2-52

[c] Source: Reference 2-53

[d] Source: Reference 2-54

[e] Source: Reference 2-42

[f] Source: Reference 2-55

Table 2-22 Tornados within 10 Miles of the U. of I. Research Reactor Site

Date	Counties Affected	Magnitude (WS range)	Length (miles)	Width (yards)	Closest Distance to the Site (miles)
05/13/2025	Champaign	EFU (Unknown)	0.05	10	2.45
03/14/2025	Champaign	EF1 (86-110 mph)	13.72	450	7.51
02/27/2023	Champaign	EFU (Unknown)	1.53	25	4.34
02/27/2023	Champaign	EF0 (65-85 mph)	1.14	30	5.92
05/26/2019	Champaign	EF1 (86-110 mph)	1.01	250	6.18
05/26/2019	Champaign	EF1 (86-110 mph)	0.85	250	0.32
05/26/2019	Champaign	EF1 (86-110 mph)	0.9	250	0.34
06/07/2015	Piatt and Champaign	EF0 (65-85 mph)	1.18	25	10.9
11/17/2013	Champaign	EF3 (136-165 mph)	14.67	880	10.5
05/25/2011	Champaign	EF0 (65-85 mph)	0.2	15	8.7
04/02/2006	Champaign	EF0 (65-85 mph)	0.1	30	1.0
05/31/2006	Champaign	EF0 (65-85 mph)	1.2	50	7.6
05/31/2006	Champaign	EF0 (65-85 mph)	0.4	50	8.0
04/20/2004	Champaign	EF0 (65-85 mph)	0.1	50	8.5
06/10/2004	Champaign	EF0 (65-85 mph)	0.7	10	8.0
06/10/2004	Champaign	EF0 (65-85 mph)	0.5	10	8.5
07/09/2003	Champaign	EF0 (65-85 mph)	0.1	10	1.0
08/18/2001	Champaign	EF0 (65-85 mph)	0.1	5	8.5
10/24/2001	Champaign	EF1 (86-110 mph)	1.0	100	1.0
06/04/1999	Champaign	EF0 (65-85 mph)	0.1	20	1.0
05/02/1998	Champaign	EF0 (65-85 mph)	0.1	10	8.0
06/29/1998	Champaign	EF0 (65-85 mph)	0.6	20	8.0
04/19/1996	Champaign	EF3 (136-165 mph)	4.0	220	3.0
05/28/1996	Champaign	EF0 (65-85 mph)	0.3	40	1.0
06/17/1992	Champaign	EF0 (65-85 mph)	0.1	10	1.0
05/09/1990	Champaign	EF1 (86-110 mph)	0.5	50	1.0
06/20/1990	Champaign	EF2 (111-135 mph)	5	100	1.0

Sources: Reference 2-56

Table 2-23 Design-Basis Tornado Characteristics for Region I

Maximum Wind Speed (m/s) (mph)	Translational Speed (m/s) (mph)	Maximum Rotational Speed (m/s) (mph)	Radius of Maximum Rotational Speed (m/s) (mph)	Pressure Drop (mb) (psi)	Rate of Pressure Drop (mb/s) (psi/s)
103 (230)	21 (46)	82 (184)	45.7 (150)	83 (1.2)	37 (0.5)

Source: Reference 2-78

m meters

mph miles per hour

mb millibars

psi pounds per square inch

s second(s)

Table 2-24 Design-Basis Tornado Missile Characteristics for Region I

Missile Type	Mass (kg)	Maximum Wind Speed (m/s) (mph)
Solid Steel Sphere	0.0669	8 (18)
Automobile	1,810	41 (92)
Schedule 40 Pipe	130	41 (92)

Source: Reference 2-78

kg kilogram(s)

m meters

mph miles per hour

s second(s)

Table 2-25 Maximum Dry Bulb and Mean Coincident Wet Bulb Temperatures

Annual Exceedance	Description	Temperature (°F)
50-Year Run Period	Dry Bulb Temperature	100.5
	Coincident Wet Bulb Temperature	86.1
0.4%	Dry Bulb Temperature	91.3
	Coincident Wet Bulb Temperature	74.8
1%	Dry Bulb Temperature	89.5
	Coincident Wet Bulb Temperature	74.5
2%	Dry Bulb Temperature	87.2
	Coincident Wet Bulb Temperature	73.3

Source: Reference 2-65

°F degrees Fahrenheit

Table 2-26 Maximum Wet Bulb Temperatures

Annual Exceedance	Temperature (°F)
50-Year Run Period	84.9
0.4%	79.4

Source: Reference 2-65

°F degrees Fahrenheit

Table 2-27 Minimum Dry Bulb Temperatures

Annual Exceedance	Temperature (°F)
50-Year Run Period	-24.4
0.4%	28.4
1.0%	28.3

Source: Reference 2-65

°F degrees Fahrenheit

Table 2-28 Monthly Design Dry Bulb and Mean Coincident Wet Bulb Temperatures

Monthly Exceedance		Jan	Feb	Mar	Apr	May	Jun
0.4%	Dry Bulb Temperature	60.5	64.1	77.0	83.8	91.4	94.0
	Mean Coincident Wet	56.8	55.7	62.7	64.1	69.9	73.8
2%	Dry Bulb Temperature	52.9	56.9	69.7	79.7	87.5	90.6
	Mean Coincident Wet	50.4	52.3	59.0	62.2	68.5	73.0
5%	Dry Bulb Temperature	47.3	51.8	63.7	74.9	84.0	88.2
	Mean Coincident Wet	44.2	47.3	55.1	60.9	67.1	72.4
Monthly Exceedance		**Jul**	**Aug**	**Sep**	**Oct**	**Nov**	**Dec**
0.4%	Dry Bulb Temperature	94.3	92.6	93.1	86.7	73.7	62.0
	Mean Coincident Wet	78.5	77.4	72.8	69.3	60.6	57.6
2%	Dry Bulb Temperature	90.5	89.6	89.3	81.4	68.1	56.4
	Mean Coincident Wet	77.9	76.7	71.5	65.7	57.9	53.1
5%	Dry Bulb Temperature	88.2	86.5	85.2	75.9	63.0	51.9
	Mean Coincident Wet	76.8	75.3	70.0	62.8	55.9	49.0

Source: Reference 2-65

Air Temperature (°F)

Table 2-29 Monthly Design Wet Bulb Temperatures

Monthly Exceedance	Jan	Feb	Mar	Apr	May	Jun
0.4%	57.4	58.7	64.4	68.4	74.3	79.4
Monthly Exceedance	**Jul**	**Aug**	**Sep**	**Oct**	**Nov**	**Dec**
0.4%	82.1	81.3	77.0	72.0	64.1	58.6

Source: Reference 2-65

Air Temperature (°F)

Table 2-30 Meteorological Tower at U. of I.

Meteorological Tower	Location	Data Collected	Data Collection Period
U. of I. Campus Meteorological Station	Latitude: 40.084004° N Longitude: 88.240427° W Elevation: 718.5 feet above mean sea level UTM: Zone 15 Northing: 4447.949 km Easting: 905.858 km	10-meter Wind 2-meter Temperature Relative Humidity Precipitation Solar Radiation Barometric Pressure	February 16, 1989-Present

Source: Reference 2-39

Table 2-31 Average (Scalar) Wind Speed Applied to the Research Reactor Site (2021–2025)

University of Illinois Urbana-Champaign Meteorological Tower			
Quarter	**Average (scalar) 10-m Wind Speed (mph)**	**Quarter**	**Average (scalar) 10-m Wind Speed (mph)**
2021		**2022**	
1st quarter	5.57	1st quarter	5.69
2nd quarter	3.68	2nd quarter	4.07
3rd quarter	2.42	3rd quarter	2.46
4th quarter	3.80	4th quarter	4.64
2023		**2024**	
1st quarter	5.74	1st quarter	5.54
2nd quarter	3.70	2nd quarter	3.78
3rd quarter	2.33	3rd quarter	2.54
4th quarter	3.65	4th quarter	4.08
2025			
1st quarter	5.69	Overall	3.99
2nd quarter	3.94		
3rd quarter	2.18		
4th quarter	4.19		

Notes:

Source: Reference 2-70

mph miles per hour

Table 2-32 Wind Direction Persistence Applied to the Research Reactor Site (2021–2025)

Wind Sector	Maximum Hours of Wind Direction Persistence at 10 meters for U. of I. Onsite Tower (mph)		
	Same Sector	+/-1 Sector	+/-2 Sector
N	**55**	104	192
NNE	**55**	**112**	**210**
NE	21	109	118
ENE	23	69	127
E	17	54	73
ESE	16	38	101
SE	15	49	59
SSE	26	59	117
S	25	82	123
SSW	15	64	91
SW	23	51	85
WSW	20	56	125
W	26	107	122
WNW	34	82	107
NW	20	77	97
NNW	25	84	113

Notes:

Source: Reference 2-70

mph miles per hour

Bold indicates the maximum values.

Gray fill indicates the sector range.

Data Period: January 1, 2021 – December 31, 2025.

Table 2-33 Air Temperatures for Urbana-Champaign, Illinois

	Normal Daily Maximum	Normal Dry Bulb	Normal Daily Minimum	Extreme Daily Maximum	Extreme Daily Minimum
Period of Record (years)	**30 [a]**	**10 [b]**	**30 [a]**	**120 [c]**	**120 [c]**
January	33.5	27.3	17.9	70	-25
February	38.4	32.4	21.2	76	-25
March	50.4	43.6	31.2	86	-9
April	63.1	52.6	41.6	91	14
May	73.8	65.0	52.7	97	26
June	82.7	74.5	62.1	103	37
July	85.2	75.5	65.2	109	41
August	84.0	73.7	63.6	102	39
September	78.8	69.7	55.6	101	29
October	65.8	56.7	43.9	93	13
November	50.7	42.7	32.2	81	-5
December	38.5	34.6	23.6	71	-20
Annual	62.1	54.0	42.6	--	--

[a] Reference 2-41 (1991–2020)

[b] Reference 2-42 (Sep 2015–Aug 2025)

[c] Reference 2-79 (1991–2020)

Notes: Air Temperature (°F)

Table 2-34 Air Temperatures Measured at the U. of I. Meteorological Station

Month	Average Monthly Maximum (°F)	Average Monthly Minimum (°F)
January	49.8	-0.9
February	62.9	1.2
March	74.5	16.5
April	82.1	27.6
May	89.4	41.1
June	94.8	50.0
July	93.9	58.4
August	94.9	54.7
September	92.3	45.2
October	85.8	31.3
November	72.7	15.2
December	61.1	3.0

Source: Reference 2-70 (2021–2025)

Notes:

Maximum temperature 98.6

Minimum temperature -13.5

°F degrees Fahrenheit

Table 2-35 Long-Term Humidity Values Measured at the U. of I. Meteorological Station

	Mean Temperature	Mean Dewpoint Temperature	Mean Relative Humidity (%)	Mean Absolute Humidity (g/m^3)
January	27.1	22.2	82.3	3.30
February	30.9	24.9	79.4	3.69
March	41.5	33.1	74.2	5.14
April	52.7	41.7	69.5	7.17
May	63.5	52.6	71.1	10.60
June	72.7	61.8	71.7	14.42
July	75.2	66.5	76.5	16.69
August	73.6	65.5	78.1	16.15
September	67.0	57.1	73.6	12.33
October	54.9	45.6	73.7	8.23
November	42.2	35.3	78.3	5.56
December	31.7	27.1	83.3	4.00

Source: Reference 2-49 (1989–2024)

% percent

g/m^3 gram(s) per cubic meter

Table 2-36 Most Recent 5-Year Humidity Values Measured at the U. of I. Meteorological Station

Month	Average Dry Bulb Temperature (°F)	Mean Relative Humidity (%)
January	26.7	84.4
February	30.6	78.9
March	44.2	72.1
April	52.9	70.0
May	64.4	70.8
June	74.3	69.3
July	75.8	80.4
August	74.0	80.3
September	68.7	74.0
October	57.1	73.2
November	42.4	77.3
December	34.3	84.7

Source: Reference 2-70 (2021–2025)

°F degrees Fahrenheit

% percent

Table 2-37 Historical Precipitation Data for Urbana, IL

	Normal Monthly (inches)	Maximum Monthly	Minimum Monthly	Maximum in 24 hours	Days with Precipitation (0.01 inch)	Days with Thunderstorms [d]
Period of Record (yrs)	**136[a]**	**136 [a]**	**136 [a]**	**136 [b]**	**30[c]**	**20**
January	2.12	7.62	0.06	2.47	5.1	1.8
February	2.00	6.12	0.15	3.18	4.6	2.3
March	3.10	8.35	0.38	2.93	5.8	3.5
April	3.75	9.55	0.50	3.09	7.7	5.3
May	4.16	11.20	0.22	4.50	8.6	6.8
June	4.12	11.58	0.32	3.89	7.2	8.6
July	3.85	13.82	0.47	4.43	6.5	7.3
August	3.51	10.02	0.06	5.32	5.6	6.6
September	3.20	9.76	0.25	3.91	5.3	3.7
October	2.83	9.01	0.16	3.72	6.1	2.7
November	2.78	10.08	Trace	4.07	6.4	1.5
December	2.39	7.47	0.05	2.97	5.1	1.4
Annual	37.72	58.54	23.95	--	74.0	46.65

Notes: Station ID: CHAMPAIGN 3S

Sources:

[a] Reference 2-72 (1890–2025)

[b] Reference 2-73 (1890–2025)

[c] Reference 2-41 (1991–2020)

[d] Reference 2-74 (2006–2025)

Table 2-38 Precipitation Data for U. of I. Meteorological Station

Month	2021 Monthly Precipitation Totals at U. of I. Meteorological Station (inches)	2022 Monthly Precipitation Totals at U. of I. Meteorological Station (inches)	2023 Monthly Precipitation Totals at U. of I. Meteorological Station (inches)	2024 Monthly Precipitation Totals at U. of I. Meteorological Station (inches)	2025 Monthly Precipitation Totals at U. of I. Meteorological Station (inches)
January	2.68	0.64	2.12	5.32	0.97
February	2.21	3.45	2.29	0.69	0.51
March	4.09	4.57	4.00	1.98	2.62
April	2.07	3.16	1.47	6.62	3.35
May	3.35	3.22	1.92	6.16	2.61
June	7.32	0.78	1.77	2.43	3.23
July	4.54	2.35	2.93	7.64	3.55
August	4.11	4.86	3.77	5.06	1.15
September	3.04	4.64	3.12	2.46	1.85
October	7.29	2.33	4.12	0.82	1.51
November	1.26	1.83	0.79	6.70	1.74
December	2.97	2.78	3.68	3.31	1.68
Annual Sum	44.93	34.59	31.97	49.18	24.76

Source: Reference 2-70 (2018–2022)

Table 2-39 Point Precipitation (Inches) by Recurrence Interval for Region

	Recurrence Intervals (Years)						
Duration	**1**	**2**	**5**	**10**	**25**	**50**	**100**
5 minutes	0.41	0.49	0.58	0.65	0.740	0.81	0.88
10 minutes	0.64	0.76	0.89	0.99	1.13	1.23	1.33
15 minutes	0.78	0.923	1.10	1.23	1.40	1.53	1.65
30 minutes	1.03	1.24	1.50	1.71	1.98	2.18	2.39
1 hour	1.26	1.52	1.89	2.17	2.56	2.87	3.19
2 hours	1.50	1.81	2.23	2.57	3.08	3.53	4.02
3 hours	1.61	1.94	2.39	2.76	3.33	3.83	4.40
6 hours	1.91	2.29	2.80	3.25	3.92	4.50	5.16
12 hours	2.22	2.67	3.25	3.75	4.49	5.14	5.87
24 hours	2.57	3.07	3.73	4.32	5.19	5.99	6.89
2 days	2.99	3.56	4.29	4.93	5.88	6.75	7.72
4 days	3.42	4.06	4.85	5.52	6.51	7.36	8.31
7 days	4.03	4.76	5.59	6.27	7.28	8.13	9.07
10 days	4.59	5.43	6.34	7.09	8.17	9.07	10.0

Notes:

Data is from NOAA Atlas 14 (Reference 2-73).

Data is for the Urbana Station (ID 11-8740).

Table 2-40 Historical Snowfall at Urbana NWS Station

	Normal Monthly (inches)	Normal Monthly (inches)	Maximum Monthly (inches)	Maximum in 24 hours (inches)	Maximum Snow Depth (inches)	Normal Number of Days with Snowfall ≥0.01 inch
Period of Record (yrs)	**30**[a]	**123**[b]	**123**[b]	**123**[c]	**123**[d]	**30**[a]
January	6.50	6.0	28.3	11.4	18	5.4
February	5.80	5.7	20.0	11.0	19	4.4
March	2.50	3.7	32.0	14.0	15	2.1
April	0.30	0.5	8.0	6.0	4	0.3
May	0.00	0.0	2.5	2.5	0	0.0
May - October	0.00	0.0	0.0	0.0	0	0.0
October	0.00	0.1	3.3	2.8	1	0.0
November	0.90	1.5	11.2	9.1	9	1.1
December	4.80	4.8	20.4	9.5	13	4.1
Annual	20.8	22.2			19[(d)]	17.4

Sources:

[a] Reference 2-41

[b] Reference 2-80

[c] Reference 2-81

[d] Reference 2-82 (1982)

Table 2-41 Fog Occurrence at Willard Airport

Period of Record	20 years
Month	**Number of Days with Heavy Fog (visibility ≤ 1/4 mile)**
January	4.2
February	4.2
March	3.3
April	2.0
May	2.2
June	2.4
July	3.0
August	2.7
September	2.8
October	2.1
November	2.1
December	4.1
Annual	29.6

Source: Reference 2-74

Table 2-42 Calculated Pasquil Atmospheric Stability at the U. of I. Meteorological Station

Pasquill Atmospheric Stability Class	Percent Occurrence
A	1.10
B	5.17
C	9.85
D	51.74
E	13.07
F	14.36
G	4.71
Unstable	16.12
Neutral	51.74
Stable	32.14

Table 2-43 List of Inputs Used in XOQDOQ Modeling

XOQDOQ Input Variable	Value
Wind Sensor Height	10 m
Type of Release	Ground
Vent Average Velocity	0 m/s
Vent Inside Diameter	0 m
Vent Release Height	10 m
Containment Building Height	0 m
Building Cross-Sectional Area	0 m^2

m = meters

m/s = meters per second

Notes: No building wake credit was used in the modeling. For a ground-level release, the exit velocity and diameter are set to zero, while the wind height is set to 10 m, consistent with NUREG-2919.

Table 2-44 Joint Frequency Distribution (Hours) of Wind Speed and Direction by Atmospheric Stability Class - Stability Class A

Wind Direction	Wind Speed (m/s)					
	≤ 0.5	0.5 < WS ≤ 2	4 < WS ≤ 4	4 < WS ≤ 6	6 < WS ≤ 9	> 9
N	2	16	0	0	0	0
NNE	0	29	0	0	0	0
NE	0	26	0	0	0	0
ENE	2	15	0	0	0	0
E	1	16	0	0	0	0
ESE	1	16	0	0	0	0
SE	0	19	0	0	0	0
SSE	4	11	0	0	0	0
S	0	14	0	0	0	0
SSW	1	11	0	0	0	0
SW	0	12	1	0	0	0
WSW	0	31	1	0	0	0
W	0	26	1	0	0	0
WNW	0	7	1	0	0	0
NW	0	10	0	0	0	0
NNW	0	8	0	0	0	0

Table 2-45 Joint Frequency Distribution (Hours) of Wind Speed and Direction by Atmospheric Stability Class - Stability Class B

Wind Direction	Wind Speed (m/s)					
	≤ 0.5	0.5 < WS ≤ 2	4 < WS ≤ 4	4 < WS ≤ 6	6 < WS ≤ 9	> 9
N	30	60	11	0	0	0
NNE	0	91	27	0	0	0
NE	2	92	22	0	0	0
ENE	2	70	4	0	0	0
E	4	45	0	0	0	0
ESE	8	57	1	0	0	0
SE	12	75	1	0	0	0
SSE	8	76	1	0	0	0
S	2	58	0	0	0	0
SSW	2	55	0	0	0	0
SW	2	101	7	0	0	0
WSW	3	93	31	0	0	0
W	6	63	33	0	0	0
WNW	4	57	12	0	0	0
NW	2	38	5	0	0	0
NNW	3	49	4	0	0	0

Table 2-46 Joint Frequency Distribution (Hours) of Wind Speed and Direction by Atmospheric Stability Class - Stability Class C

Wind Direction	Wind Speed (m/s)					
	≤ 0.5	0.5 < WS ≤ 2	4 < WS ≤ 4	4 < WS ≤ 6	6 < WS ≤ 9	> 9
N	72	81	45	0	0	0
NNE	4	73	94	1	0	0
NE	5	110	69	0	0	0
ENE	4	101	22	0	0	0
E	8	57	9	0	0	0
ESE	12	79	13	0	0	0
SE	15	112	18	2	0	0
SSE	11	131	21	1	0	0
S	13	106	7	3	0	0
SSW	6	145	20	0	0	0
SW	4	184	85	0	0	0
WSW	2	91	143	3	0	0

Table 2-46 Joint Frequency Distribution (Hours) of Wind Speed and Direction by Atmospheric Stability Class - Stability Class C (Continued)

Wind Direction	Wind Speed (m/s)					
	≤ 0.5	0.5 < WS ≤ 2	4 < WS ≤ 4	4 < WS ≤ 6	6 < WS ≤ 9	> 9
W	5	89	152	6	0	0
WNW	1	70	60	1	0	0
NW	2	94	20	0	0	0
NNW	0	76	26	1	0	0

Table 2-47 Joint Frequency Distribution (Hours) of Wind Speed and Direction by Atmospheric Stability Class - Stability Class D

Wind Direction	Wind Speed (m/s)					
	≤ 0.5	0.5 < WS ≤ 2	4 < WS ≤ 4	4 < WS ≤ 6	6 < WS ≤ 9	> 9
N	249	256	386	112	3	0
NNE	14	231	449	116	19	3
NE	20	321	229	23	8	0
ENE	21	284	283	19	0	0
E	28	211	151	14	1	0
ESE	32	212	139	19	0	0
SE	47	297	236	26	1	0
SSE	49	524	292	34	0	0
S	34	563	553	85	1	0
SSW	25	493	482	32	1	0
SW	23	379	530	75	4	0
WSW	15	219	398	118	15	0
W	12	201	519	355	122	1
WNW	14	268	611	265	28	0
NW	17	334	409	26	0	0
NNW	7	330	479	52	0	0

Table 2-48 Joint Frequency Distribution (Hours) of Wind Speed and Direction by Atmospheric Stability Class - Stability Class E

Wind Direction	Wind Speed (m/s)					
	≤ 0.5	0.5 < WS ≤ 2	4 < WS ≤ 4	4 < WS ≤ 6	6 < WS ≤ 9	> 9
N	251	60	86	0	0	0
NNE	19	156	170	0	0	0
NE	24	132	22	0	0	0
ENE	32	119	34	0	0	0
E	28	91	13	0	0	0
ESE	45	75	5	0	0	0
SE	55	186	15	0	0	0
SSE	37	204	4	0	0	0
S	36	161	8	0	0	0
SSW	25	182	13	0	0	0
SW	10	242	71	0	0	0
WSW	4	128	62	0	0	0
W	6	114	81	1	0	0
WNW	6	129	60	0	0	0
NW	9	119	29	0	0	0
NNW	6	104	85	0	0	0

Table 2-49 Joint Frequency Distribution (Hours) of Wind Speed and Direction by Atmospheric Stability Class - Stability Class F

Wind Direction	Wind Speed (m/s)					
	≤ 0.5	0.5 < WS ≤ 2	4 < WS ≤ 4	4 < WS ≤ 6	6 < WS ≤ 9	> 9
N	1101	144	13	0	0	0
NNE	58	254	32	0	0	0
NE	53	132	1	0	0	0
ENE	58	63	1	0	0	0
E	41	26	1	0	0	0
ESE	39	22	0	0	0	0
SE	70	69	0	0	0	0
SSE	30	35	0	0	0	0
S	32	25	0	0	0	0
SSW	17	52	0	0	0	0
SW	24	164	0	0	0	0
WSW	19	145	5	0	0	0

Table 2-49 Joint Frequency Distribution (Hours) of Wind Speed and Direction by Atmospheric Stability Class - Stability Class F (Continued)

Wind Direction	Wind Speed (m/s)					
	≤ 0.5	0.5 < WS ≤ 2	4 < WS ≤ 4	4 < WS ≤ 6	6 < WS ≤ 9	> 9
W	48	276	30	0	0	0
WNW	44	216	18	0	0	0
NW	36	107	1	0	0	0
NNW	19	143	8	0	0	0

Table 2-50 Joint Frequency Distribution (Hours) of Wind Speed and Direction by Atmospheric Stability Class - Stability Class G

Wind Direction	Wind Speed (m/s)					
	≤ 0.5	0.5 < WS ≤ 2	4 < WS ≤ 4	4 < WS ≤ 6	6 < WS ≤ 9	> 9
N	735	27	0	0	0	0
NNE	13	50	1	0	0	0
NE	8	10	0	0	0	0
ENE	7	5	0	0	0	0
E	10	1	0	0	0	0
ESE	14	1	0	0	0	0
SE	8	4	0	0	0	0
SSE	6	3	0	0	0	0
S	7	1	0	0	0	0
SSW	4	2	0	0	0	0
SW	8	10	0	0	0	0
WSW	7	17	1	0	0	0
W	28	37	0	0	0	0
WNW	32	35	0	0	0	0
NW	24	19	0	0	0	0
NNW	27	45	0	0	0	0

Table 2-51 Long-Term Average X/Q Values (s/m^3) as the Licenses Area Boundary

Direction from Modeled Emission Point	Distance from Modeled Emission Point (miles)	Undepleted	2-Day Decay	8-Day Decay with Depletion
N	0.04	7.85E-04	7.85E-04	7.72E-04
NNE	0.04	1.13E-03	1.13E-03	1.11E-03
NE	0.05	4.97E-04	4.97E-04	4.86E-04
ENE	0.03	1.15E-03	1.15E-03	1.13E-03
E	0.03	1.04E-03	1.04E-03	1.03E-03
ESE	0.03	1.12E-03	1.12E-03	1.10E-03
SE	0.02	1.70E-03	1.70E-03	1.68E-03
SSE	0.02	1.54E-03	1.54E-03	1.53E-03
S	0.02	1.36E-03	1.36E-03	1.35E-03
SSW	0.02	3.00E-03	3.00E-03	2.97E-03
SW	0.06	2.09E-04	2.09E-04	2.04E-04
WSW	0.06	1.92E-04	1.92E-04	1.89E-04
W	0.06	2.16E-04	2.16E-04	2.13E-04
WNW	0.06	1.94E-04	1.94E-04	1.92E-04
NW	0.06	1.42E-04	1.42E-04	1.40E-04
NNW	0.06	2.60E-04	2.60E-04	2.54E-04

s/m^3 = second per cubic meter

Notes: X/Q values were calculated using AERMOD

Table 2-52 Long-Term Average D/Q Values at the Licensed Area Boundary

Direction from Modeled Emission Point	Distance from Modeled Emission Point (miles)	D/Q (m^{-2})
N	0.04	4.15E-07
NNE	0.04	4.73E-07
NE	0.05	4.96E-07
ENE	0.03	8.47E-07
E	0.03	1.04E-06
ESE	0.03	1.04E-06
SE	0.02	7.95E-07
SSE	0.02	9.26E-07
S	0.02	1.01E-06
SSW	0.02	1.18E-06
SW	0.06	2.53E-07
WSW	0.06	2.16E-07

Table 2-52 Long-Term Average D/Q Values at the Licensed Area Boundary (Continued)

Direction from Modeled Emission Point	Distance from Modeled Emission Point (miles)	D/Q (m^{-2})
W	0.06	1.35E-07
WNW	0.06	1.35E-07
NW	0.06	2.25E-07
NNW	0.06	2.83E-07

m^{-2} = per square meter

Notes: D/Q values were calculated using XOQDOQ

Table 2-53 Long-Term Average X/Q Values (s/m^3) 10 meters from the Source

Direction from Modeled Emission Point	Undepleted	2-Day Decay	8-Day Decay with Depletion
N	2.66E-02	2.66E-02	2.65E-02
NNE	3.12E-02	3.12E-02	3.11E-02
NE	3.00E-02	3.00E-02	2.99E-02
ENE	2.47E-02	2.47E-02	2.46E-02
E	2.49E-02	2.49E-02	2.48E-02
ESE	2.78E-02	2.78E-02	2.77E-02
SE	2.74E-02	2.74E-02	2.73E-02
SSE	3.00E-02	3.00E-02	2.99E-02
S	2.70E-02	2.70E-02	2.69E-02
SSW	2.72E-02	2.72E-02	2.71E-02
SW	3.06E-02	3.06E-02	3.05E-02
WSW	2.89E-02	2.89E-02	2.88E-02
W	2.68E-02	2.68E-02	2.67E-02
WNW	2.67E-02	2.67E-02	2.66E-02
NW	3.57E-02	3.56E-02	3.56E-02
NNW	3.50E-02	3.50E-02	3.49E-02

s/m^3 = second per cubic meter

Notes: X/Q values were calculated using AERMOD

Table 2-54 Long-Term Average X/Q Values (s/m^3) 20 meters from the Source

Direction from Modeled Emission Point	Undepleted	2-Day Decay	8-Day Decay with Depletion
N	7.62E-03	7.62E-03	7.57E-03
NNE	9.07E-03	9.06E-03	9.01E-03
NE	9.62E-03	9.62E-03	9.57E-03
ENE	7.42E-03	7.42E-03	7.37E-03
E	8.20E-03	8.20E-03	8.15E-03
ESE	8.97E-03	8.97E-03	8.92E-03
SE	8.17E-03	8.16E-03	8.11E-03
SSE	8.87E-03	8.87E-03	8.81E-03
S	7.85E-03	7.85E-03	7.80E-03
SSW	7.91E-03	7.90E-03	7.85E-03
SW	9.05E-03	9.05E-03	8.99E-03
WSW	8.68E-03	8.68E-03	8.62E-03
W	8.08E-03	8.08E-03	8.02E-03
WNW	7.91E-03	7.91E-03	7.86E-03
NW	1.07E-02	1.07E-02	1.06E-02
NNW	1.04E-02	1.04E-02	1.03E-02

s/m^3 = second per cubic meter

Notes: X/Q values were calculated using AERMOD

Table 2-55 Long-Term Average D/Q Values at 10 meters and 20 meters from Source

Direction from Modeled Emission Point	D/Q 10m (m^{-2})	D/Q 20m (m^{-2})
N	3.09E-06	1.56E-06
NNE	4.26E-06	2.16E-06
NE	2.84E-06	1.44E-06
ENE	1.51E-06	7.65E-07
E	1.52E-06	7.68E-07
ESE	2.52E-06	1.28E-06
SE	3.17E-06	1.61E-06
SSE	3.76E-06	1.91E-06
S	3.53E-06	1.79E-06
SSW	4.43E-06	2.24E-06
SW	3.56E-06	1.80E-06
WSW	5.00E-06	2.53E-06
W	4.36E-06	2.21E-06
WNW	4.36E-06	2.21E-06

Table 2-55 Long-Term Average D/Q Values at 10 meters and 20 meters from Source (Continued)

Direction from Modeled Emission Point	D/Q 10m (m^{-2})	D/Q 20m (m^{-2})
NW	2.88E-06	1.46E-06
NNW	3.35E-06	1.70E-06

m^{-2} = per square meter

Notes: D/Q values were calculated using XOQDOQ

Table 2-56 Annual Average X/Q (s/m^2) with No Decay or Depletion at Specified Distances for each Sector

Annual Average X/Q (s/m^2) for No Decay, Undepleted					
Sector	Distance (miles)				
	1	2	3	4	5
N	9.52E-06	3.32E-06	1.35E-06	7.88E-07	5.23E-07
NNE	1.37E-05	3.43E-06	1.14E-06	6.43E-07	4.42E-07
NE	9.11E-06	3.32E-06	1.44E-06	7.83E-07	4.88E-07
ENE	6.59E-06	2.27E-06	9.88E-07	5.70E-07	3.79E-07
E	7.19E-06	2.56E-06	1.12E-06	6.36E-07	3.91E-07
ESE	7.00E-06	2.75E-06	1.23E-06	6.67E-07	4.00E-07
SE	5.71E-06	2.22E-06	1.08E-06	6.18E-07	3.94E-07
SSE	8.38E-06	2.58E-06	9.94E-07	5.39E-07	3.68E-07
S	6.90E-06	2.01E-06	9.52E-07	5.44E-07	3.62E-07
SSW	1.19E-05	3.10E-06	1.19E-06	6.82E-07	4.51E-07
SW	5.95E-06	2.81E-06	1.37E-06	7.22E-07	4.37E-07
WSW	3.79E-06	2.29E-06	1.26E-06	7.01E-07	4.36E-07
W	3.59E-06	1.59E-06	1.08E-06	6.25E-07	4.04E-07
WNW	4.06E-06	1.28E-06	9.89E-07	6.24E-07	4.07E-07
NW	6.67E-06	2.44E-06	1.40E-06	6.96E-07	3.63E-07
NNW	9.87E-06	3.32E-06	1.42E-06	7.67E-07	4.98E-07
Annual Average X/Q (s/m^2) for No Decay, Undepleted					
Sector	Distance (miles)				
	10	20	30	40	50
N	2.37E-07	7.72E-08	3.33E-08	1.94E-08	1.26E-08
NNE	2.01E-07	6.60E-08	2.85E-08	1.67E-08	1.11E-08
NE	2.12E-07	6.86E-08	3.03E-08	1.77E-08	1.19E-08
ENE	1.64E-07	5.75E-08	2.66E-08	1.60E-08	1.13E-08
E	1.70E-07	5.75E-08	2.59E-08	1.55E-08	1.06E-08
ESE	1.76E-07	5.97E-08	2.69E-08	1.61E-08	1.05E-08
SE	1.72E-07	5.86E-08	2.62E-08	1.56E-08	1.08E-08

Table 2-56 Annual Average X/Q (s/m^2) with No Decay or Depletion at Specified Distances for each Sector (Continued)

SSE	1.68E-07	5.71E-08	2.60E-08	1.58E-08	1.10E-08
S	1.68E-07	5.69E-08	2.52E-08	1.56E-08	1.09E-08
SSW	2.02E-07	6.45E-08	2.79E-08	1.64E-08	1.13E-08
SW	1.90E-07	6.29E-08	2.73E-08	1.62E-08	1.11E-08
WSW	1.93E-07	6.38E-08	2.80E-08	1.64E-08	1.09E-08
W	1.85E-07	6.15E-08	2.77E-08	1.64E-08	1.07E-08
WNW	1.85E-07	6.14E-08	2.70E-08	1.52E-08	1.07E-08
NW	2.08E-07	7.40E-08	2.91E-08	1.52E-08	1.14E-08
NNW	2.24E-07	7.30E-08	3.02E-08	1.75E-08	1.18E-08

Notes: X/Q values between sector distance boundaries are aggregated using Equation (4) on pg 24 of the XOQDOQ user's guide (NUREG-2919)

Table 2-57 Annual Average X/Q (s/m^2) for 2.26-day Decay and No Depletion at Specified Distances for each Sector

Annual Average X/Q (s/m^2) for 2.26 Day Decay and No Depletion					
Sector	Distance (miles)				
	1	2	3	4	5
N	9.44E-06	3.26E-06	1.31E-06	7.54E-07	4.94E-07
NNE	1.36E-05	3.37E-06	1.10E-06	6.15E-07	4.16E-07
NE	9.04E-06	3.28E-06	1.40E-06	7.51E-07	4.61E-07
ENE	6.53E-06	2.23E-06	9.57E-07	5.44E-07	3.56E-07
E	7.13E-06	2.52E-06	1.09E-06	6.07E-07	3.68E-07
ESE	6.94E-06	2.70E-06	1.19E-06	6.38E-07	3.76E-07
SE	5.66E-06	2.19E-06	1.04E-06	5.89E-07	3.70E-07
SSE	8.33E-06	2.54E-06	9.62E-07	5.14E-07	3.46E-07
S	6.85E-06	1.97E-06	9.19E-07	5.18E-07	3.40E-07
SSW	1.18E-05	3.05E-06	1.15E-06	6.51E-07	4.24E-07
SW	5.91E-06	2.76E-06	1.33E-06	6.91E-07	4.10E-07
WSW	3.77E-06	2.25E-06	1.23E-06	6.70E-07	4.10E-07
W	3.56E-06	1.56E-06	1.05E-06	5.94E-07	3.79E-07
WNW	4.03E-06	1.26E-06	9.58E-07	5.95E-07	3.82E-07
NW	6.62E-06	2.41E-06	1.36E-06	6.72E-07	3.47E-07
NNW	9.80E-06	3.28E-06	1.38E-06	7.38E-07	4.74E-07
Annual Average X/Q (s/m^2) for 2.26 Day Decay and No Depletion					
Sector	Distance (miles)				
	10	20	30	40	50
N	2.18E-07	6.55E-08	2.56E-08	1.42E-08	7.84E-09

Table 2-57 Annual Average X/Q (s/m^2) for 2.26-day Decay and No Depletion at Specified Distances for each Sector (Continued)

NNE	1.82E-07	5.52E-08	2.18E-08	1.09E-08	6.36E-09
NE	1.94E-07	5.72E-08	2.26E-08	1.16E-08	6.96E-09
ENE	1.48E-07	4.75E-08	1.93E-08	1.03E-08	6.45E-09
E	1.54E-07	4.75E-08	1.88E-08	9.98E-09	6.12E-09
ESE	1.59E-07	4.92E-08	1.94E-08	1.03E-08	5.99E-09
SE	1.56E-07	4.81E-08	1.89E-08	9.98E-09	6.15E-09
SSE	1.53E-07	4.71E-08	1.88E-08	1.01E-08	6.35E-09
S	1.52E-07	4.69E-08	1.83E-08	9.96E-09	6.19E-09
SSW	1.84E-07	5.33E-08	2.03E-08	1.06E-08	6.47E-09
SW	1.71E-07	5.17E-08	1.96E-08	1.03E-08	6.31E-09
WSW	1.75E-07	5.26E-08	2.02E-08	1.05E-08	6.20E-09
W	1.67E-07	5.05E-08	1.99E-08	1.04E-08	6.05E-09
WNW	1.67E-07	5.02E-08	1.95E-08	1.03E-08	6.22E-09
NW	1.91E-07	6.22E-08	2.21E-08	1.10E-08	7.29E-09
NNW	2.06E-07	6.05E-08	2.27E-08	1.23E-08	6.89E-09

Notes: X/Q values between sector distance boundaries are aggregated using Equation (4) on pg 24 of the XOQDOQ user's guide (NUREG-2919)

Table 2-58 Annual Average X/Q (s/m^2) for 8-day Decay with Depletion at Specified Distances for each Sector

Annual Average X/Q (s/m^2) for 8 Day Decay with Depletion					
Sector	Distance (miles)				
	1	2	3	4	5
N	8.13E-06	2.54E-06	9.59E-07	5.10E-07	3.21E-07
NNE	1.19E-05	2.68E-06	7.72E-07	4.12E-07	2.59E-07
NE	7.72E-06	2.71E-06	1.08E-06	5.27E-07	3.02E-07
ENE	5.38E-06	1.74E-06	7.18E-07	3.79E-07	2.31E-07
E	5.94E-06	1.91E-06	7.85E-07	4.10E-07	2.48E-07
ESE	6.20E-06	2.24E-06	9.20E-07	4.59E-07	2.64E-07
SE	5.01E-06	1.76E-06	7.65E-07	3.94E-07	2.40E-07
SSE	7.64E-06	2.14E-06	7.22E-07	3.65E-07	2.35E-07
S	5.87E-06	1.51E-06	6.18E-07	3.37E-07	2.16E-07
SSW	1.03E-05	2.39E-06	8.50E-07	4.66E-07	2.90E-07
SW	5.39E-06	2.26E-06	9.78E-07	4.81E-07	2.68E-07
WSW	3.52E-06	1.78E-06	8.61E-07	4.39E-07	2.63E-07
W	3.23E-06	1.21E-06	6.78E-07	3.66E-07	2.36E-07
WNW	3.64E-06	1.07E-06	6.89E-07	3.95E-07	2.37E-07

Table 2-58 Annual Average X/Q (s/m^2) for 8-day Decay with Depletion at Specified Distances for each Sector (Continued)

NW	6.01E-06	2.07E-06	1.06E-06	4.99E-07	2.51E-07
NNW	8.89E-06	2.87E-06	1.08E-06	5.38E-07	3.40E-07
Annual Average X/Q (s/m^2) for 8 Day Decay with Depletion					
Sector	**Distance (miles)**				
	10	**20**	**30**	**40**	**50**
N	1.36E-07	3.79E-08	1.35E-08	7.35E-09	4.09E-09
NNE	1.07E-07	3.00E-08	1.13E-08	5.62E-09	3.42E-09
NE	1.22E-07	3.30E-08	1.15E-08	5.80E-09	3.47E-09
ENE	9.64E-08	2.75E-08	1.02E-08	5.36E-09	3.20E-09
E	1.02E-07	2.90E-08	1.09E-08	5.70E-09	3.47E-09
ESE	1.06E-07	2.98E-08	1.09E-08	5.59E-09	3.44E-09
SE	1.00E-07	2.81E-08	1.05E-08	5.45E-09	3.28E-09
SSE	9.98E-08	2.87E-08	1.08E-08	5.64E-09	3.39E-09
S	9.11E-08	2.66E-08	1.02E-08	5.33E-09	3.20E-09
SSW	1.20E-07	3.31E-08	1.19E-08	6.06E-09	3.57E-09
SW	1.07E-07	2.95E-08	1.06E-08	5.38E-09	3.19E-09
WSW	1.06E-07	2.90E-08	1.04E-08	5.30E-09	3.25E-09
W	9.76E-08	2.74E-08	9.74E-09	5.03E-09	3.18E-09
WNW	9.28E-08	2.49E-08	9.01E-09	4.58E-09	2.83E-09
NW	1.23E-07	3.21E-08	1.13E-08	5.86E-09	3.39E-09
NNW	1.37E-07	3.34E-08	1.23E-08	6.52E-09	3.45E-09

Notes: X/Q values between sector distance boundaries are aggregated using Equation (4) on pg 24 of the XOQDOQ user's guide (NUREG-2919)

Table 2-59 Annual Average D/Q (m^{-2}) at Specified Distances for each Sector

Annual Average D/Q (m^{-2})					
Sector	**Distance (miles)**				
	1	**2**	**3**	**4**	**5**
N	7.53E-09	2.33E-09	9.26E-10	5.06E-10	3.22E-10
NNE	7.07E-09	2.18E-09	8.69E-10	4.75E-10	3.02E-10
NE	8.86E-09	2.74E-09	1.09E-09	5.95E-10	3.78E-10
ENE	7.13E-09	2.20E-09	8.77E-10	4.79E-10	3.04E-10
E	1.00E-08	3.09E-09	1.23E-09	6.72E-10	4.27E-10
ESE	8.73E-09	2.70E-09	1.07E-09	5.87E-10	3.73E-10
SE	5.75E-09	1.78E-09	7.07E-10	3.86E-10	2.46E-10
SSE	6.70E-09	2.07E-09	8.23E-10	4.50E-10	2.86E-10
S	6.18E-09	1.91E-09	7.59E-10	4.15E-10	2.64E-10

Table 2-59 Annual Average D/Q (m^{-2}) at Specified Distances for each Sector (Continued)

SSW	8.53E-09	2.64E-09	1.05E-09	5.73E-10	3.64E-10
SW	5.69E-09	1.76E-09	6.99E-10	3.82E-10	2.43E-10
WSW	4.85E-09	1.50E-09	5.96E-10	3.26E-10	2.07E-10
W	3.02E-09	9.33E-10	3.71E-10	2.03E-10	1.29E-10
WNW	3.04E-09	9.38E-10	3.73E-10	2.04E-10	1.30E-10
NW	5.04E-09	1.56E-09	6.20E-10	3.39E-10	2.15E-10
NNW	6.35E-09	1.96E-09	7.81E-10	4.27E-10	2.71E-10
Annual Average D/Q (m^{-2})					
Sector	**Distance (miles)**				
	10	**20**	**30**	**40**	**50**
N	1.38E-10	4.28E-11	1.70E-11	9.06E-12	5.61E-12
NNE	1.30E-10	4.02E-11	1.59E-11	8.51E-12	5.27E-12
NE	1.63E-10	5.04E-11	2.00E-11	1.07E-11	6.60E-12
ENE	1.31E-10	4.06E-11	1.61E-11	8.58E-12	5.31E-12
E	1.84E-10	5.69E-11	2.26E-11	1.21E-11	7.46E-12
ESE	1.60E-10	4.97E-11	1.97E-11	1.05E-11	6.51E-12
SE	1.06E-10	3.27E-11	1.30E-11	6.92E-12	4.29E-12
SSE	1.23E-10	3.81E-11	1.51E-11	8.06E-12	4.99E-12
S	1.13E-10	3.51E-11	1.39E-11	7.43E-12	4.60E-12
SSW	1.57E-10	4.85E-11	1.92E-11	1.03E-11	6.36E-12
SW	1.04E-10	3.23E-11	1.28E-11	6.84E-12	4.24E-12
WSW	8.89E-11	2.76E-11	1.09E-11	5.83E-12	3.61E-12
W	5.54E-11	1.72E-11	6.81E-12	3.64E-12	2.25E-12
WNW	5.57E-11	1.73E-11	6.84E-12	3.65E-12	2.26E-12
NW	9.24E-11	2.87E-11	1.14E-11	6.07E-12	3.76E-12
NNW	1.17E-10	3.61E-11	1.43E-11	7.65E-12	4.73E-12

Notes: X/Q values between sector distance boundaries are aggregated using Equation (4) on pg 24 of the XOQDOQ user's guide (NUREG/CR-2919)

Figure 2-18 Regional Setting

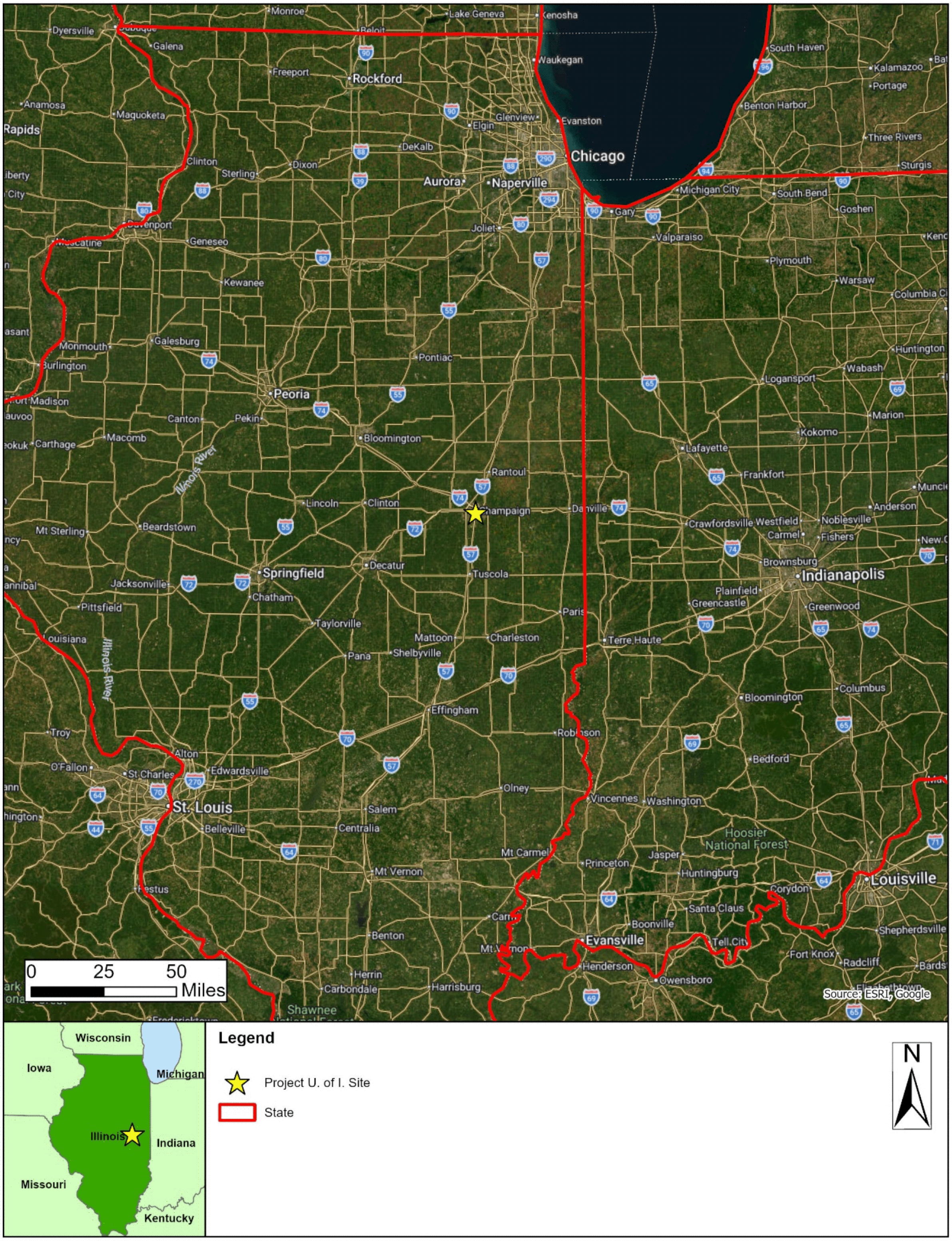


Figure 2-19 Meteorological Tower Location

Figure 2-20 Local Topography and Location of U. of I. Meteorological Tower

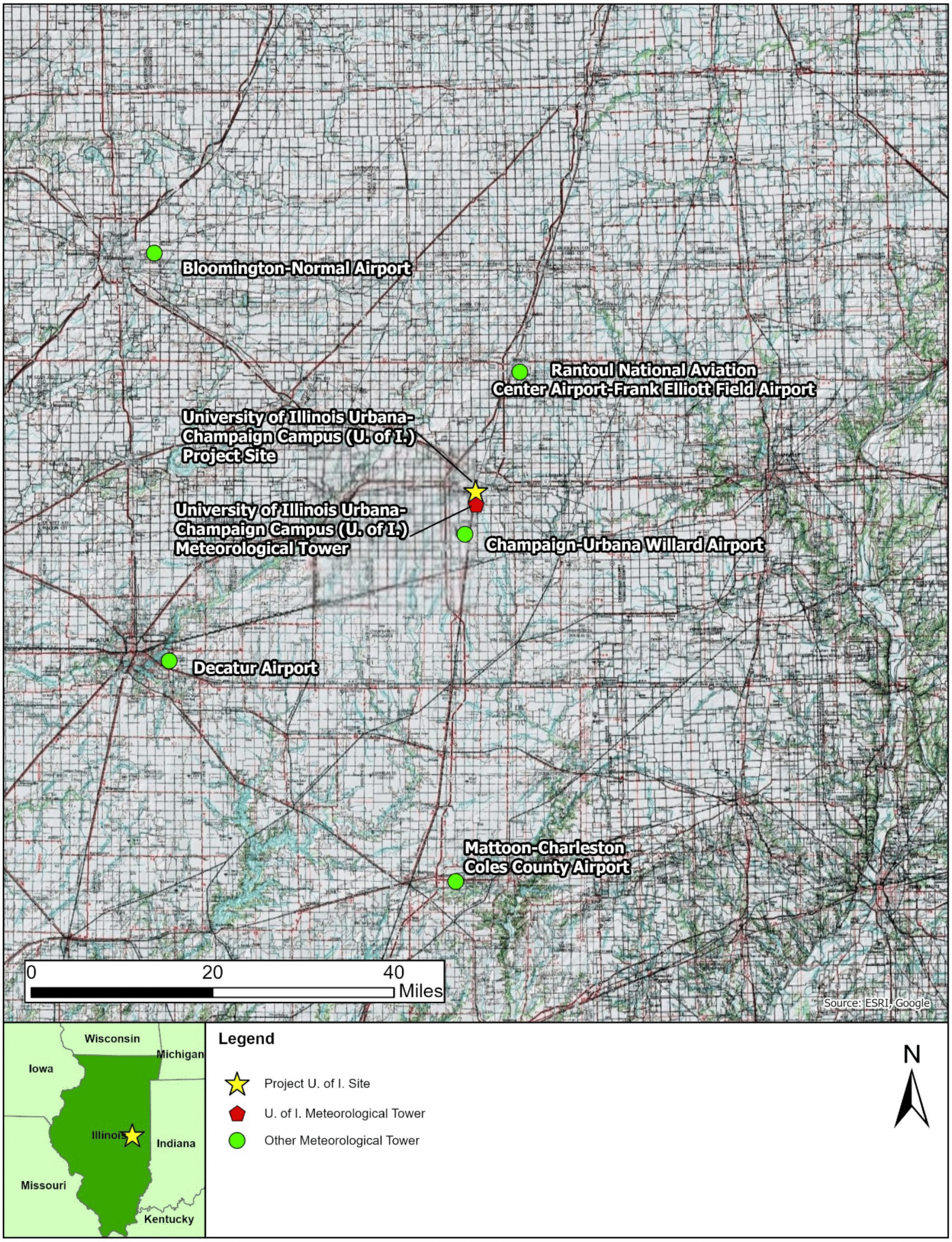


Figure 2-21 50-mile (80-km) Region Surrounding the U. of I. Research Reactor Site

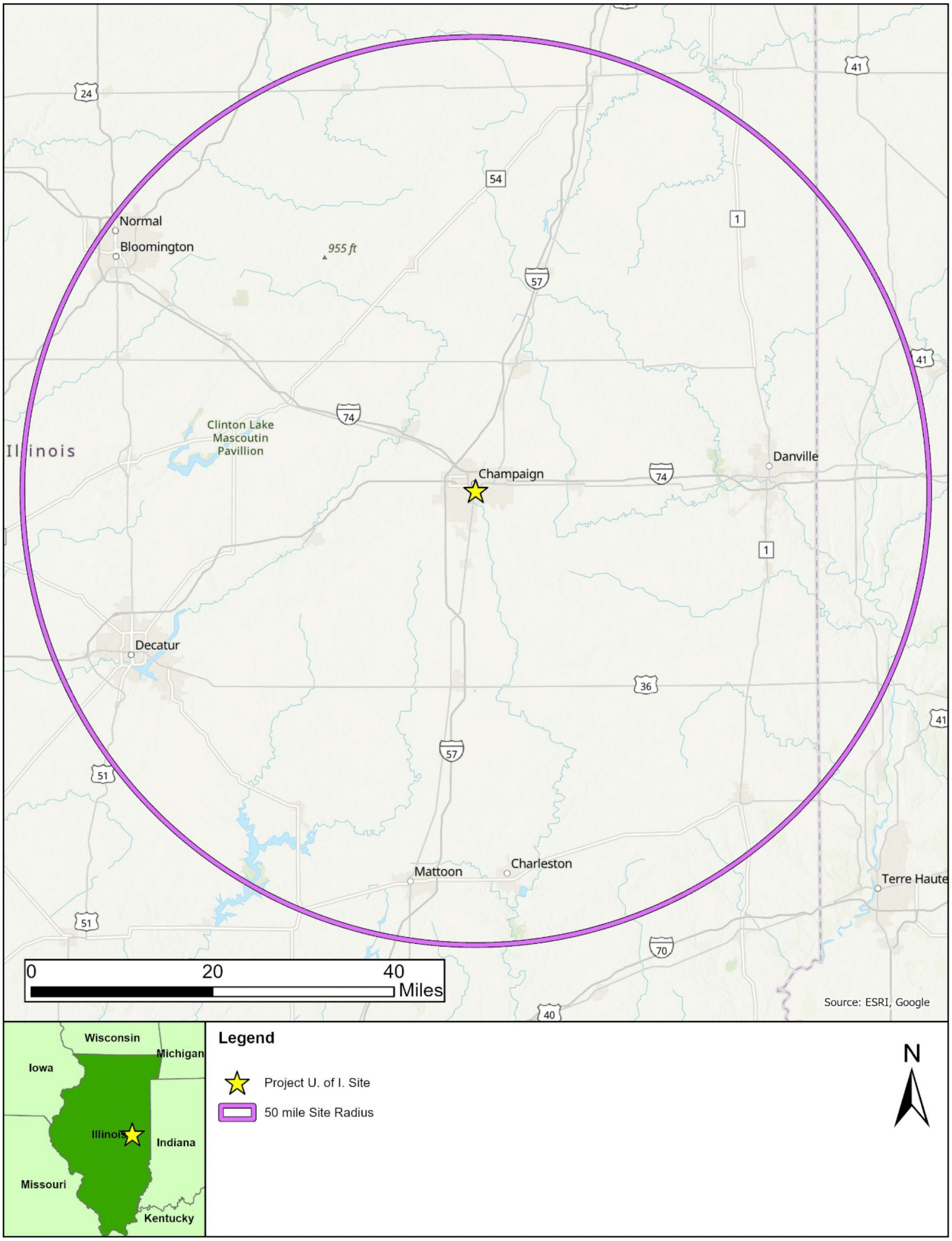
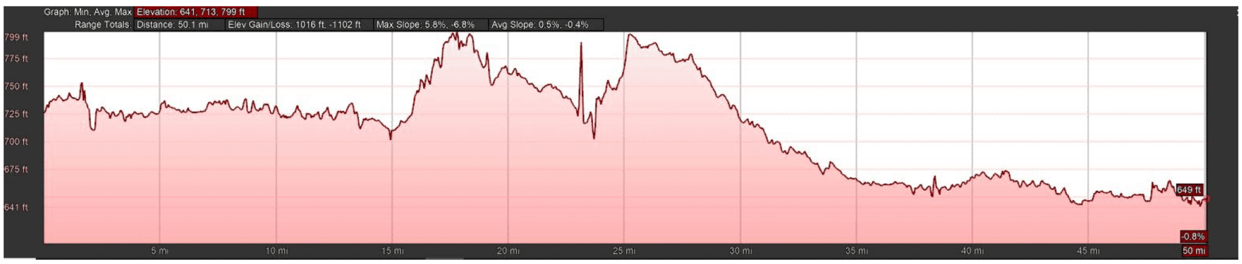


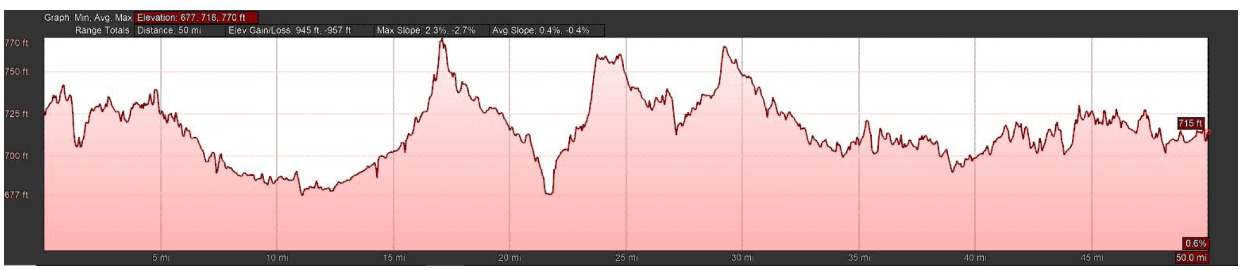
Figure 2-22 Terrain Elevations Within 50 Miles North and North-Northeast of the Project Site

North

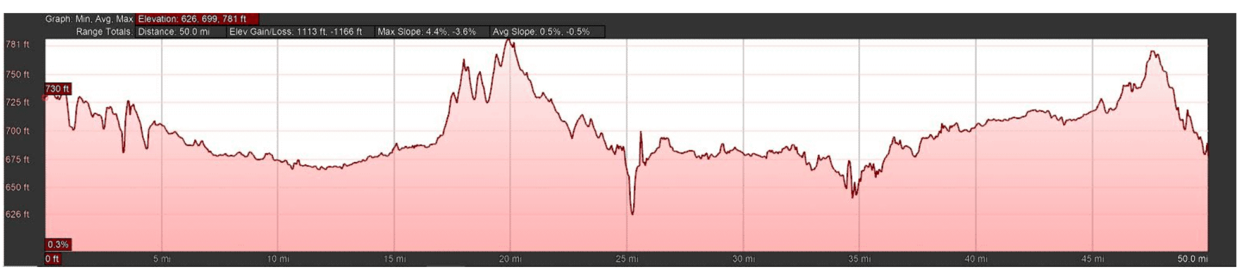


North – Northeast

Figure 2-23 Terrain Elevations Within 50 Miles Northeast and East-Northeast of the Project Site

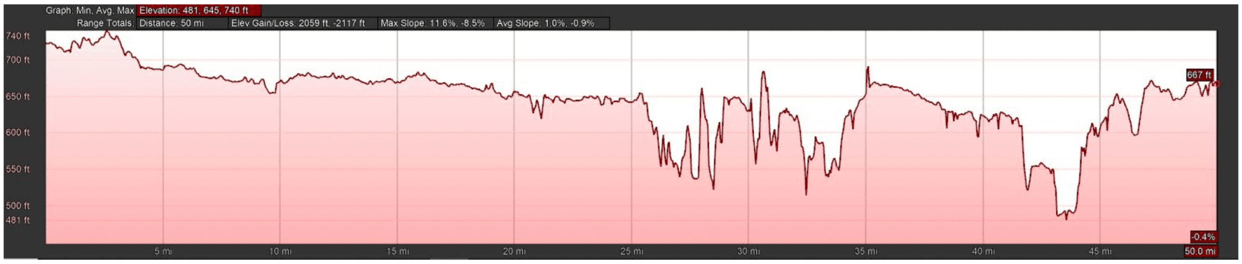


Northeast

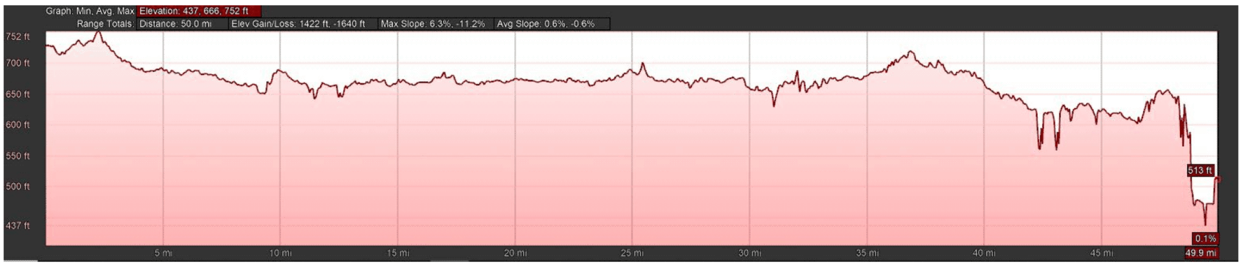


East – Northeast

Figure 2-24 Terrain Elevations within 50 Miles East and East-Southeast of the Project Site

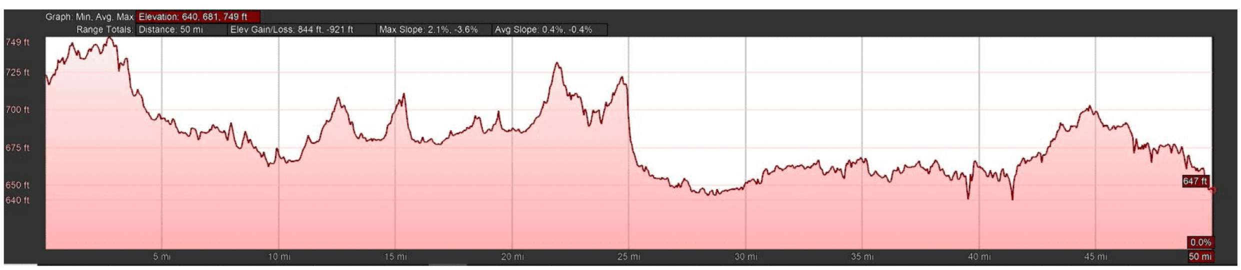


East



East-Southeast

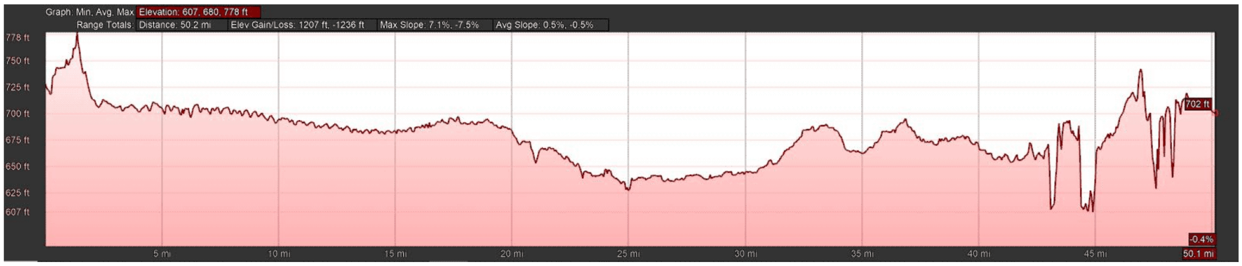
Figure 2-25 Terrain Elevations within 50 Miles Southeast and South-Southeast of the Project Site



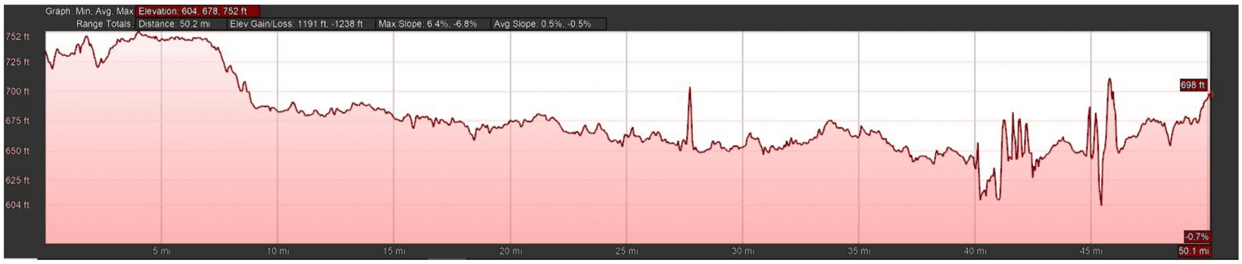
Southeast

South – Southeast

Figure 2-26 Terrain Elevations Within 50 Miles South and South-Southwest of the Project Site

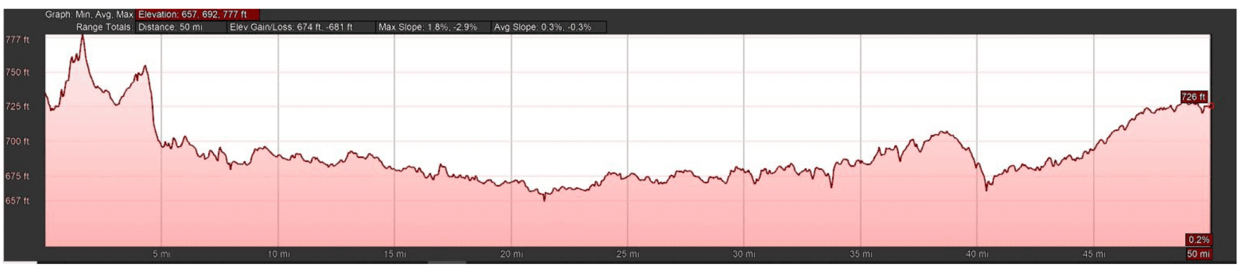


South

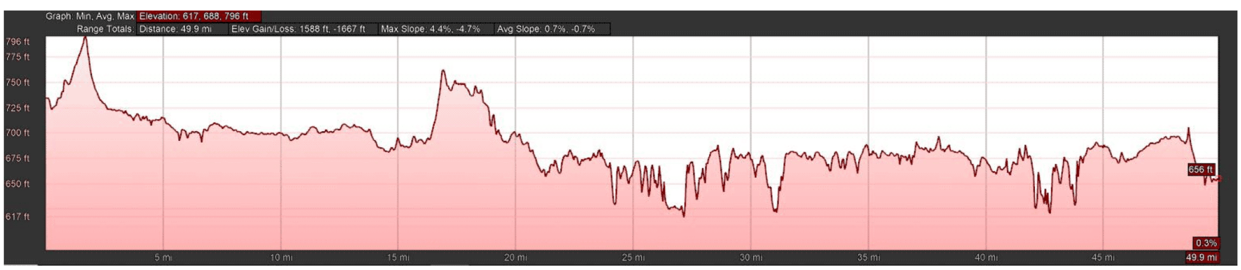


South – Southwest

Figure 2-27 Terrain Elevations Within 50 Miles Southwest and West-Southwest of the Project Site

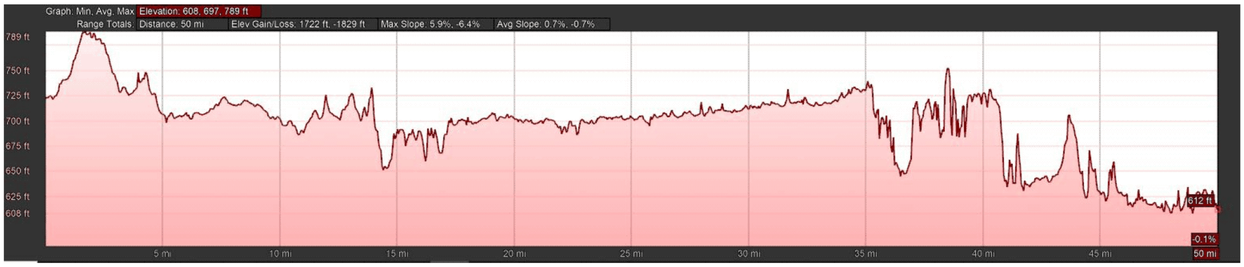


Southwest

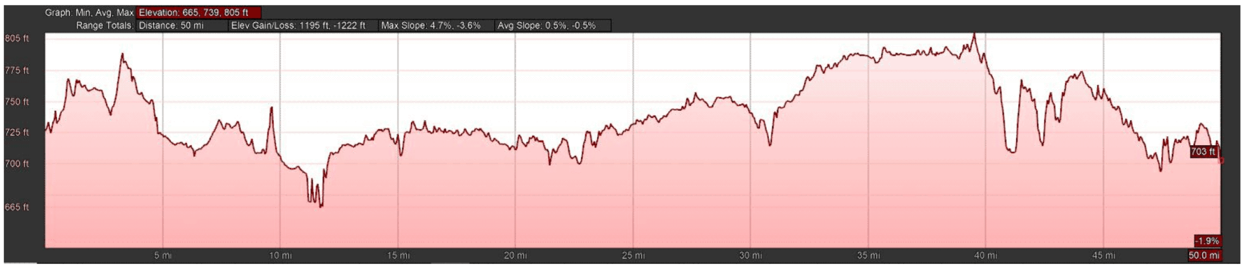


West – Southwest

Figure 2-28 Terrain Elevations Within 50 Miles West and West-Northwest of the Project Site

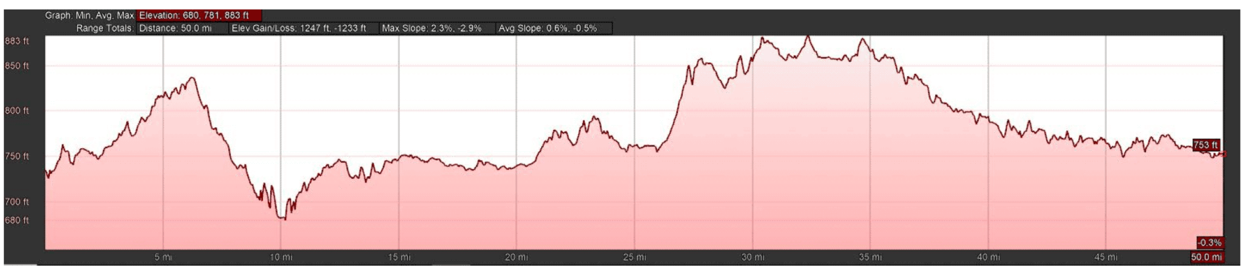


West

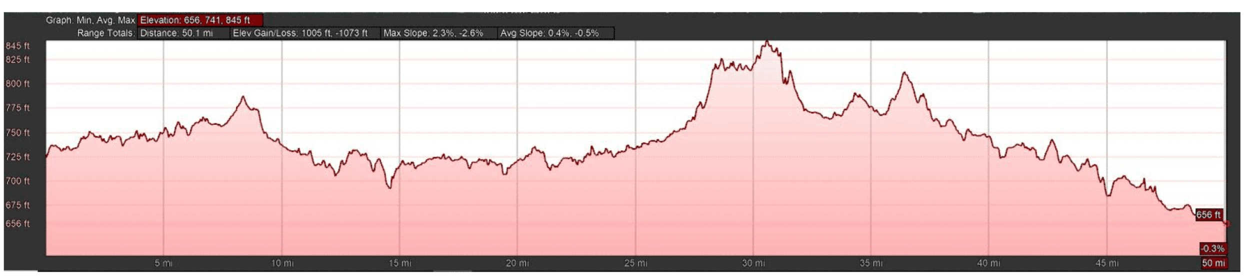


West – Northwest

Figure 2-29 Terrain Elevations Within 50 Miles Northwest and North-Northwest of the Project Site



Northwest



North - Northwest

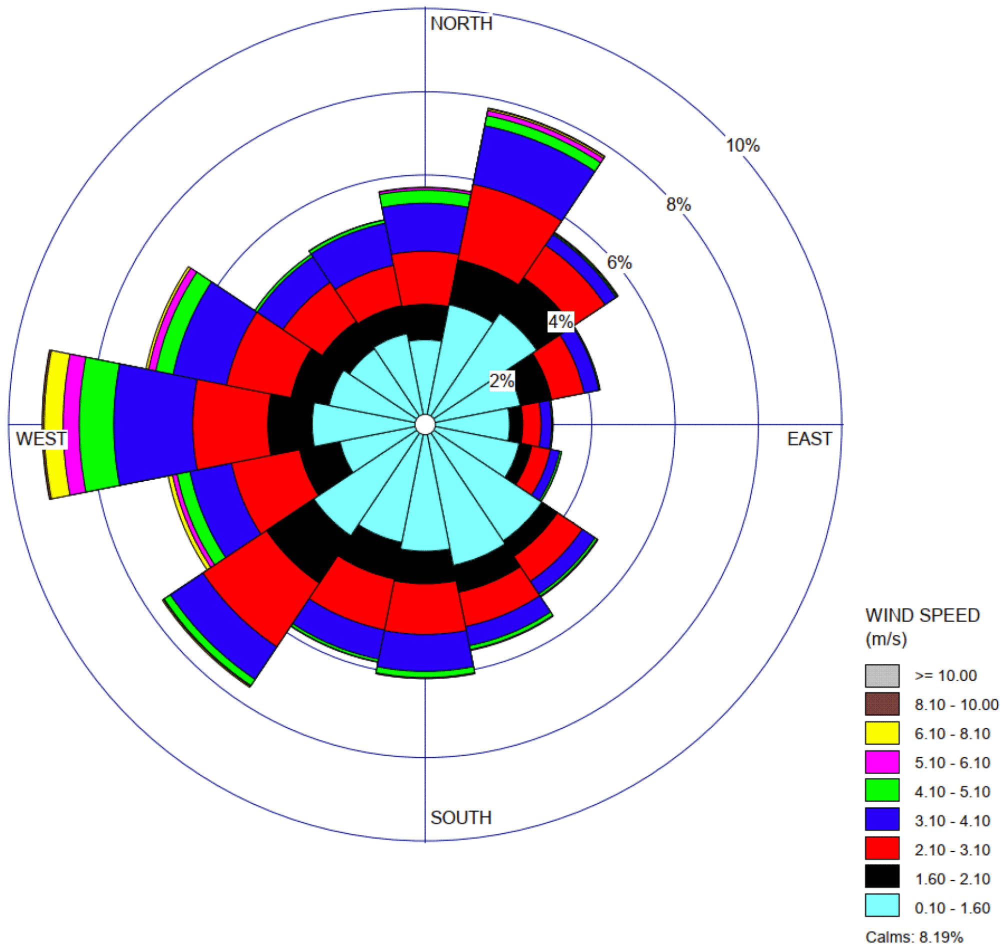
Figure 2-30 10-Meter Wind Rose – Meteorological Tower at U. of I.

WIND ROSE PLOT:

2021-2025 Wind Rose for Onsite 10 m Meteorological Tower
University of Illinois Champaign-Urbana Campus - All Hours

DISPLAY:

Wind Speed
Direction (blowing from)



COMMENTS:

DATA PERIOD:

Start Date: 1/1/2021 - 00:00
End Date: 12/31/2025 - 23:00

COMPANY NAME:

MODELER:

CALM WINDS:

8.19%

TOTAL COUNT:

43819 hrs.

AVG. WIND SPEED:

1.76 m/s

DATE:

1/26/2026

PROJECT NO.:

WRPLOT View - Lakes Environmental Software

Figure 2-31 Wind Direction by 2025 Quarter for Meteorological Tower at U. of I.

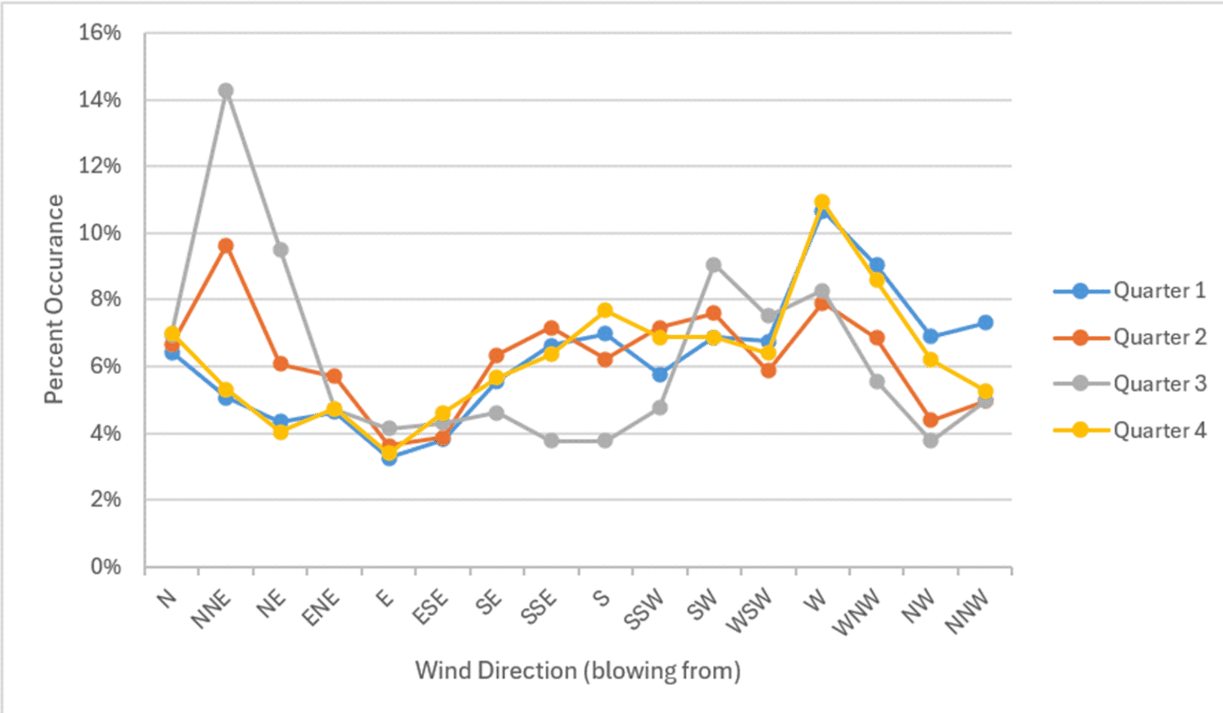


Figure 2-32 Daytime Wind Rose for Meteorological Tower at U. of I.

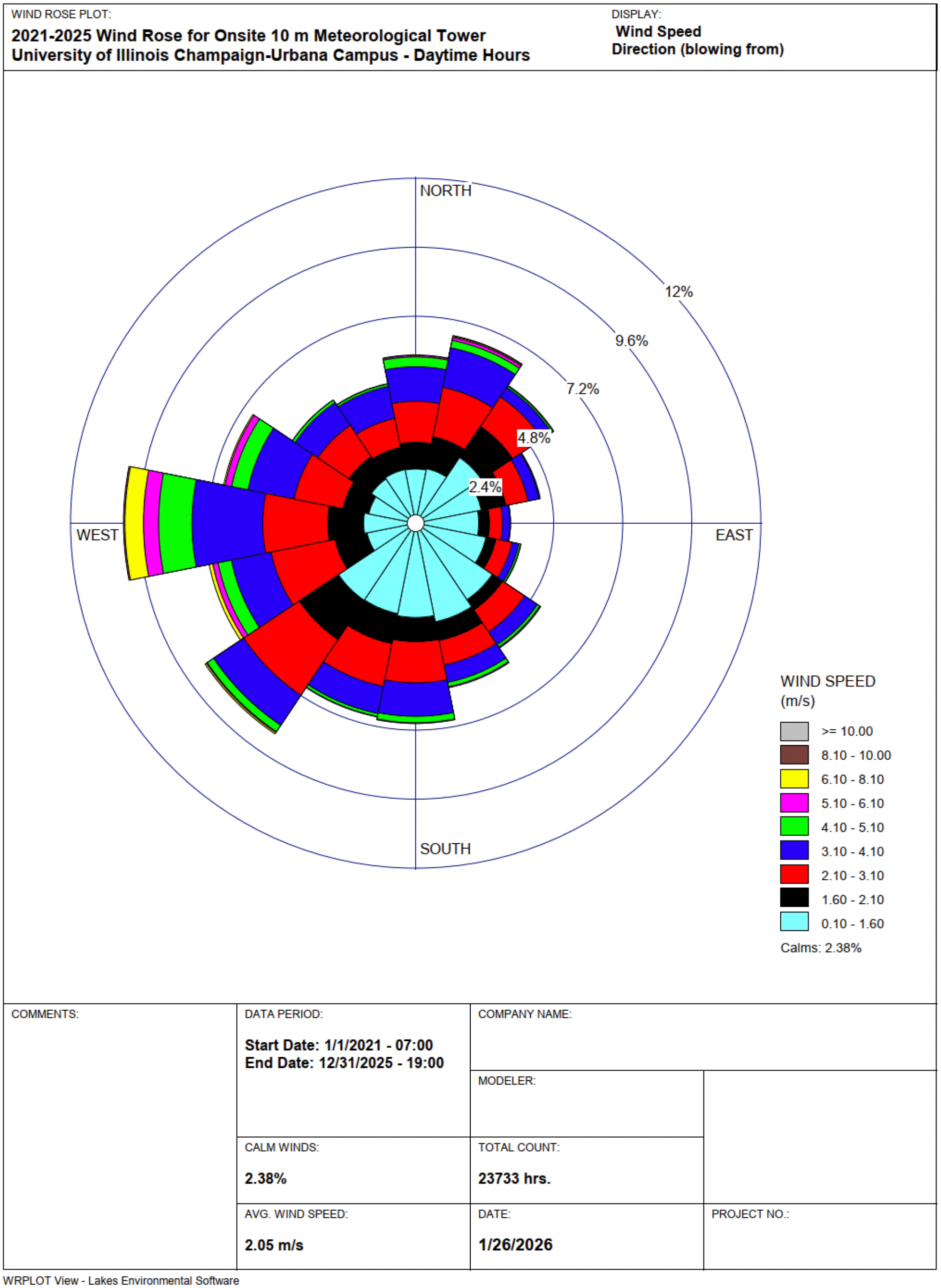


Figure 2-33 Nighttime Wind Rose for Meteorological Tower at U. of I.

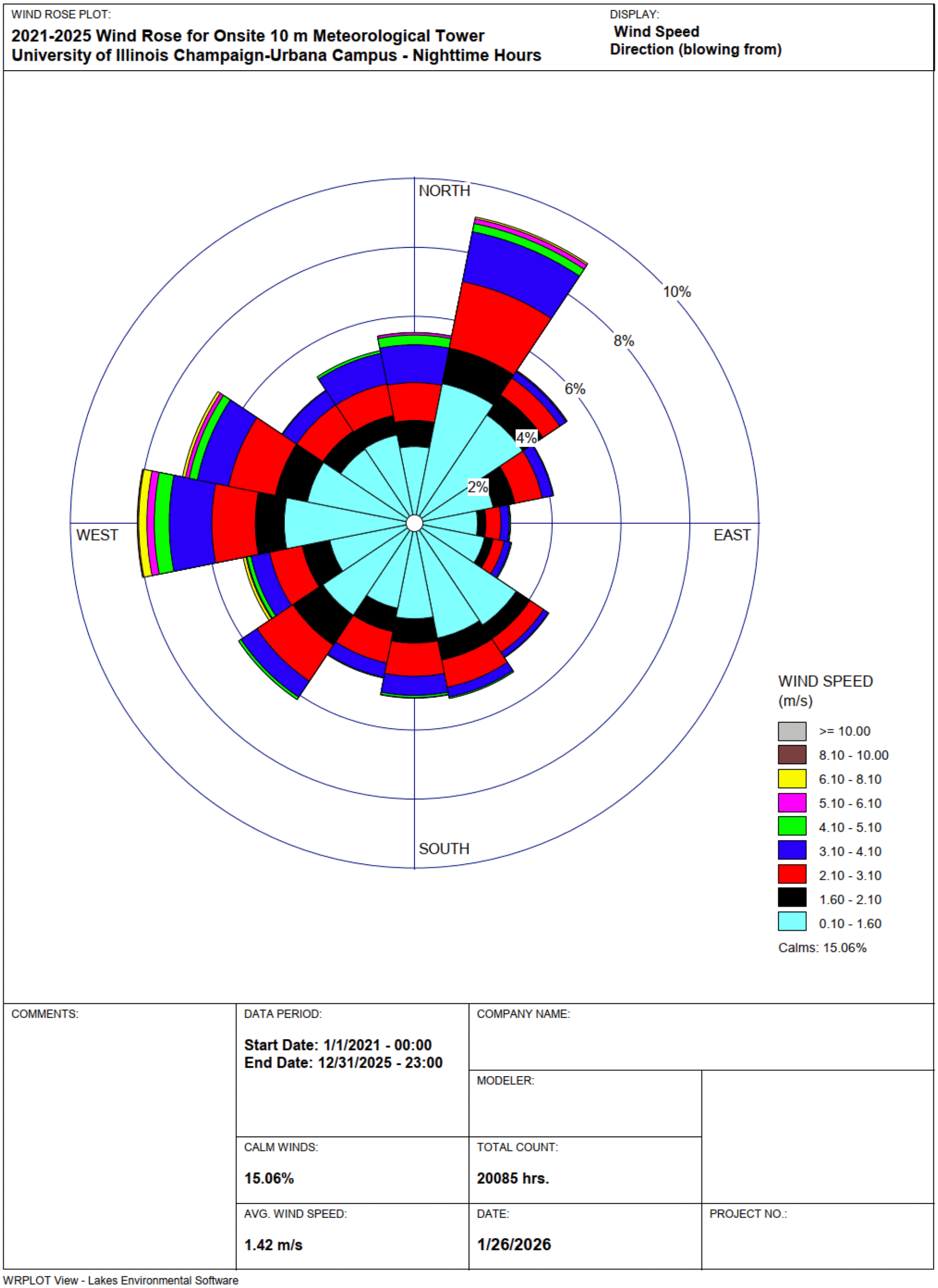
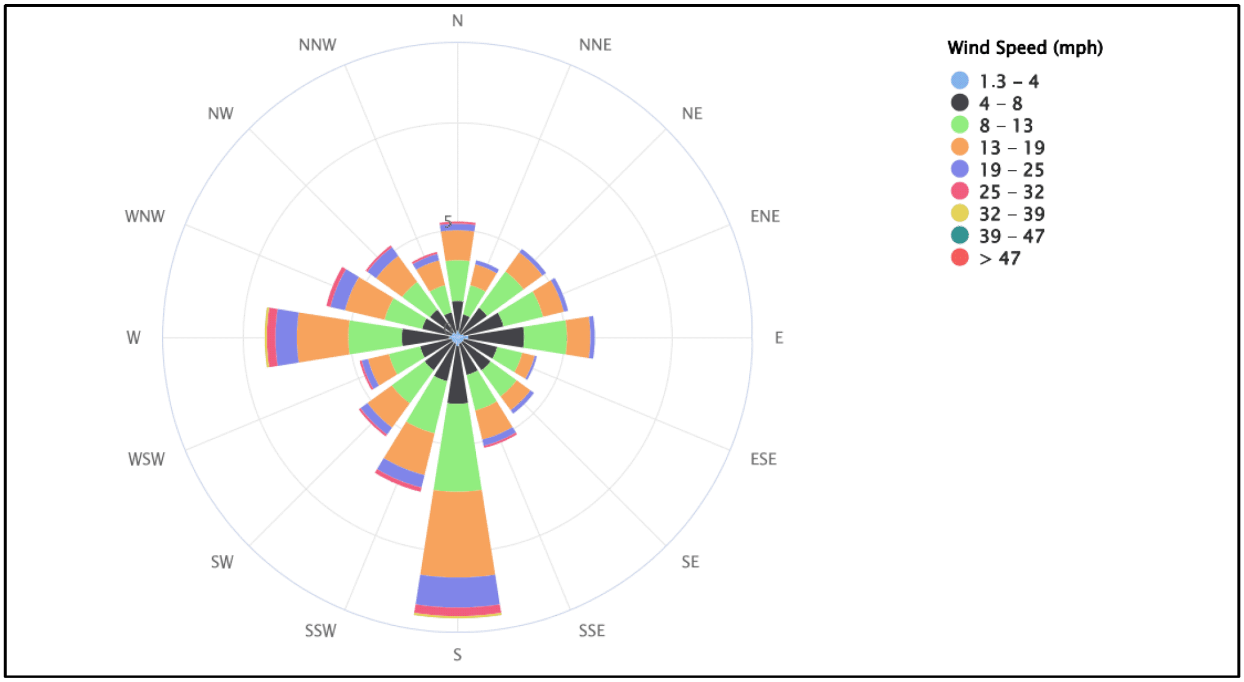


Figure 2-34 Champaign Urbana Willard Airport Wind Rose (July 1, 1996 – December 2025)



Source: Midwest Regional Climate Center 2025

Figure 2-35 Precipitation Wind Rose for Meteorological Tower at U. of I.

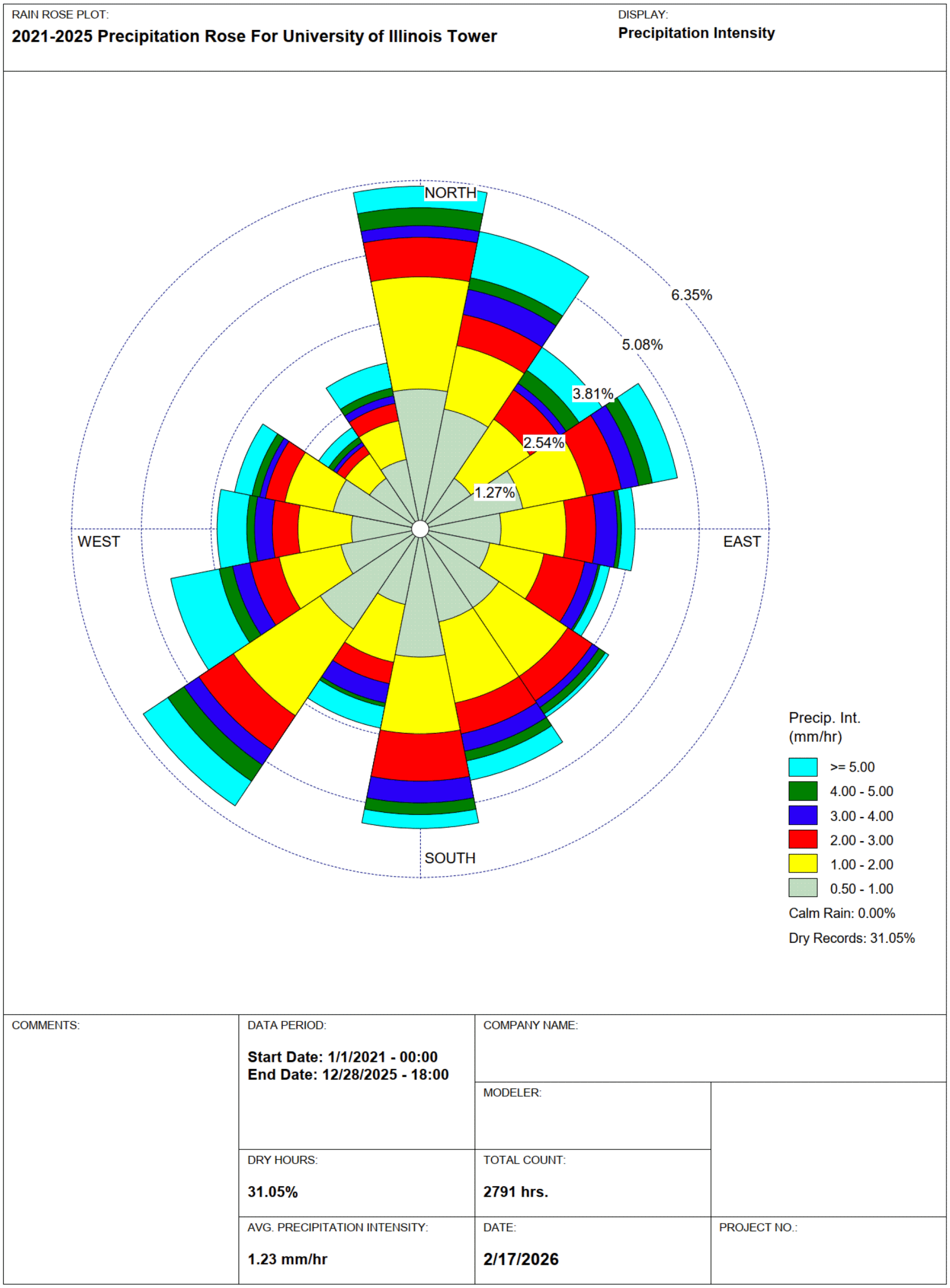


Figure 2-36 Photo of Meteorological Tower at U. of I.

2.4 HYDROLOGY

The guidance in NUREG-1537 discusses that an evaluation of relevant hydrologic, hydrogeologic, and solute transport risks associated with nuclear facilities, both during their operational phase and post-closure, must be provided. This chapter specifically addresses surface water dynamics including hydrologic flows from rivers and streams that may impact the site. Its objective is to identify hydrological processes that could lead to radioactive releases and to characterize the parameters that define these processes.

According to NUREG-1537, the facility design must account for potential leakage or loss of primary coolant into groundwater. The U. of I. research reactor utilizes helium gas for cooling, making a primary coolant spill a non-credible scenario. Nonetheless, to align with the guidance provided in NUREG-1537, the subsequent sections will outline the surface and subsurface hydrology of the site and indicate that the risk of flood near the proposed site is deemed negligible, whether from drainage runoff, ground movement, dam failure, or creek and river blockages.

2.4.1 Hydrological Description

This subsection identifies the site surface water, groundwater aquifers, types of on-site groundwater use, sources of recharge, present known withdrawals and likely future withdrawals, flow rates, travel time, gradients, and other properties that affect movement of accidental contaminants in groundwater, groundwater levels beneath the site, seasonal and climatic fluctuations, monitoring and protection requirements, and man-made changes that have the potential to cause long-term changes in local groundwater regime.

2.4.1.1 General Setting – Surface Water

The proposed research reactor site is situated within a hydrologically complex region influenced by multiple watersheds and rivers, depicted in Figure 2-37 and Figure 2-38, respectively, and obtained from Illinois State Water Survey (Prairie Research Institute). While Champaign County is prominently covered by the Upper Sangamon Watershed, which includes the Sangamon River, the Salt Fork Vermilion River Watershed (Wabash Basin) directly encompasses Champaign and Urbana, making it the most immediate and locally significant drainage basin. The Salt Fork Vermilion River flows eastward, eventually joining the Middle Fork Vermilion River to form the Vermilion River, which drains into the Wabash River and ultimately the Mississippi River.

In addition to these two key watersheds, the Embarras River Watershed originates just south of Champaign-Urbana and flows southward, also draining into the Wabash River. The Kaskaskia River Watershed, another drainage system, flows westward and eventually reaches the Mississippi River directly. Together, these watersheds and their associated rivers – Embarras, Kaskaskia, and Vermilion – highlight the hydrological complexity of the region.

The proposed site is located at the highest elevation in the vicinity, near the boundaries of the Salt Fork Vermilion, Upper Sangamon, Embarras, and Kaskaskia watersheds. While it is primarily within the Salt Fork Vermilion River Watershed, its proximity to the other watersheds underscores its position at a regional drainage divide. This elevated location ensures that no upstream stormwater runoff contributes to the site's hydrology. Instead, all precipitation falling on the site is conveyed as shallow sheet flow into the local storm-sewer system, bypassing natural stream channels.

A key feature of the local hydrology is Boneyard Creek, historically known as Silver Creek, which flows through the urbanized areas of Urbana and Champaign. The U. of I. campus and its environs drain predominantly northward toward Boneyard Creek, situated approximately 2,800 feet north-northeast of the proposed research reactor site. Given the urban nature of the area, storm sewers are extensively utilized to collect surface runoff and convey it efficiently to Boneyard Creek. Approximately two miles northeast of the site, Boneyard Creek joins the Saline Branch Ditch. From this confluence, runoff continues an additional 10.4 miles eastward to the Salt Fork Vermilion River system.

Stream flow rates at Boneyard Creek typically range from 1 to 3 cubic feet per second (cfs) (28.3 to 85.0 liters per second) during non-storm conditions. However, these rates can increase substantially during storm events, with peak flows reaching between 30 and 600 cfs (850 to 17,000 liters per second) during storm events, as depicted in Figure 2-39 (Reference 2-83). This is typical of small urban streams that receive runoff from extensive impervious surfaces. Following the onset of rainfall, runoff from these surfaces – such as roofs, pavements, and sidewalks – enters Boneyard Creek, resulting in sharp rises in stream flow rates, followed by equally rapid decline once the rainfall ceases. U. of I. climatology is discussed in Section 2.3.

2.4.1.2 General Setting – Groundwater

Beneath the site and the broader Champaign County area, groundwater occurs primarily in unconsolidated glacial sand and gravel aquifers. The region's geology features a series of buried bedrock valleys filled with glacial deposits that form productive aquifers. The most prominent of these is the Mahomet Bedrock Valley. This valley contains extensive Pleistocene-aged sand and gravel deposits known as the Mahomet Aquifer, which, along with the coeval Sankoty-Mahomet Aquifer, serve as the principal sources of fresh groundwater for east-central Illinois. These aquifers provide high-quality water and serve as the major freshwater supply for numerous municipalities in the region, including Champaign-Urbana. The Mahomet Aquifer, outlined in Figure 2-40, is part of the lower aquifer system in the area and is known for its prolific water yield. Individual high-capacity wells can yield over 2,000 gallons (7,571 liters) per minute from the thick sand/gravel units in the deepest parts of the Mahomet valley.

Above the Mahomet Aquifer, there are also smaller, shallower sand and gravel water-bearing units within the glacial drift. These include upper aquifers found in discontinuous sand lenses of the surficial (Wisconsinan-age) glacial deposits and middle aquifers in more continuous Illinoian-age sand/gravel layers at intermediate depths. The shallowest aquifer units (commonly 20 to 100 feet [6 to 30 meters] deep) contribute only a few percent of water use and are often utilized where the deeper Mahomet Aquifer is not easily accessible.

Groundwater beneath the site is expected to exist in the glacial drift deposits above the Mahomet Aquifer, likely hydraulically connected to the regional water table. Depth to groundwater in the vicinity is relatively shallow. In general, across Champaign-Urbana, the water table is encountered only about 5–10 feet (1.5 to 3 meters) below the ground surface in low-lying areas near streams, and on higher ground such as the U. of I. campus uplands it is typically 15 to 25 feet (4.6 to 7.6 meters) below the surface. Seasonal fluctuations in the water table are on the order of 5 to 8 feet (1.5 to 2.4 meters) annually, reflecting higher levels in wet spring months and lower in late summer or during droughts. These general ranges are consistent with on-site observations, recent geotechnical borings at the proposed reactor location found groundwater at approximately 10 to 15 feet (3 to 4.6 meters) below grade, indicating a shallow unconfined aquifer beneath the site.

Regional studies describe the shallow glacial deposits as a mix of more permeable sand and gravel pockets and tighter silt and clay layers, so groundwater conditions can vary over short distances. The deeper Mahomet/Sankoty aquifer is generally much more permeable and productive, which is why it supports the areas municipal well fields. Groundwater quality in these aquifers is typically good, with iron and hardness being the most common treatable issues. Importantly, recent investigation at the site did not indicate groundwater contamination based on the field observations and data collected.

2.4.1.3 Present Withdrawals and Known and Likely Future Withdrawals

The design of the research reactor facility does not include groundwater withdrawal or injection, and no planned future injection or withdrawal is expected to have a significant impact on facility operation or safety.

2.4.1.4 Groundwater Flow

The local and regional groundwater flow conditions are characterized by a series of interconnected processes that define the movement of water through the subsurface environment. The groundwater flow pathway consists of three primary components: recharge, underground flow, and discharge. Recharge represents the initial stage of the groundwater flow system and occurs through two primary mechanisms. The first is precipitation and infiltration, where water from rainfall or snowmelt percolates through the ground surface into the subsurface. The second mechanism involves underground recharge, where groundwater enters the system laterally from adjacent geological domains. Following recharge, water transitions into the underground flow phase, which is further divided into unsaturated and saturated flow. In the unsaturated zone, water moved through pore spaces in the soil and rock that are not filled with water. Upon reaching the water table, the water enters the saturated zone, where all pore spaces are fully saturated, allowing for more efficient movement. The final stage of the groundwater flow pathway is discharge, where water exits the subsurface system. Discharge may occur through natural processes, such as the emergence of groundwater at springs, wetlands, or streams, or through human activities, such as the extraction of water from wells. These discharge points can serve as pathways for contaminants to reach the surface, potentially causing exposure to humans or the environment. In general, contaminants are introduced into the groundwater system during the recharge phase. They are subsequently transported with groundwater during underground flow and may reach exposure points through discharge.

At the regional scale, the groundwater system has been extensively studied and documented in the literature, including the work of Sanderson and Zewde (Reference 2-84). However, at the scale of the site, the available information is derived primarily from site-specific investigations, including drilling, testing and geotechnical analyses.

Recharge of the groundwater flow system at the site occurs through two primary pathways: precipitation and infiltration through the ground surface, and lateral subsurface flow from surrounding background domains. The hydraulic properties of the shallow glacial materials at the site, such as hydraulic conductivity and porosity, have been characterized in the context of broader studies on the Mahomet aquifer system. Studies reveal that the upper aquifer units exhibit variable permeability (Reference 2-84). Coarse sand and gravel pockets facilitate rapid water transmission; while intervening silts and clays act as confining or semi-confining layers, impeding flow.

The Mahomet/Sankoty Aquifer, a significant regional groundwater resource, is composed predominantly of thick sand and gravel deposits. These materials exhibit high permeability and storage capacity, enabling the aquifer to sustain large municipal well fields. Despite the extensive regional studies, site-specific data on infiltration rates at the proposed site remain unavailable. However, Walton (Reference 2-85) reported groundwater recharge rates in Illinois from 0.01 to 1.60 gallons per square foot (0.0035 to 0.56 liters per square meter) per day, depending on soil type.

As precipitation infiltrates the soil layers, water migrates through the unsaturated zone, eventually reaching the water table. From there, it flows within the saturated zone toward potential discharge regions. The rate of groundwater movement is influenced by the hydraulic conductivity of the materials through which it flows. The highest conductivity values are typically observed in the saturated zone, where water movement is relatively rapid. In contrast, water migration in the unsaturated zone is slower, assuming similar material properties in both zones. At the proposed site, the water table was encountered at depths ranging from 10 to 15 feet (3 to 4.6 meters) below the ground surface, as determined from on-site borehole investigations.

2.4.2 Floods

This subsection of the PSAR identifies historical flooding, defined as occurrences of abnormally high-water stage or overflow from a stream, floodway, lake, or coastal area, at the proposed research reactor site.

2.4.2.1 Boneyard Creek Flows

Boneyard creek exhibits a persistent susceptibility to flooding throughout the year, with the most significant risk occurring during the spring runoff when rainfall events coincide with snowmelt. The floodplain delineation shown in Figure 2-41 outlines the area of Boneyard Creek subject to inundation during regulatory flood events. The watershed's extensive urbanization – characterized by impervious roofs, pavements, and roadways – amplifies runoff generation and accelerates the conveyance of stormwater to the creek. As a result, peak discharges can occur in any season following intense precipitation. Hydrograph records indicate that water levels rise rapidly at the onset of rainfall and recede just as quickly once precipitation ceases, leaving the creek at modest base-flow levels between storm events. Continuous-record data from the U.S. Geological Survey (Rev. 83) place typical base flows at approximately 0 to 4 cfs (0 to 0.11 cubic meters per second [m^3/s]), whereas storm flows span a broad range of 10 to 600 cfs (0.28 to 17 m^3/s), as depicted in Figure 2-37.

The nearest stream-gauge station is situated on Boneyard Creek roughly 1 mile (1.6 kilometers [km]) northeast of, and hydraulically downstream from the proposed site. Because the project site occupies one of the highest topographic positions within the drainage basis, it receives no appreciable run-on flow from upslope areas and is not vulnerable to flooding. Consequently, overland flow generated on the facility's parcel is limited to shallow sheet flow directed toward on-site storm-drain inlets and is therefore not subjected to gauging.

2.4.2.2 Flood Record Details and Elevations

The Federal Emergency Management Agency (FEMA) completed a comprehensive flood hazard assessment for Champaign County in 2013. As part of that effort, FEMA's Flood Insurance Rate Map (FIRM) Number 17019C0426D designates the proposed U. of I. research reactor parcel as an "Area of Minimal Flood Hazard," indicating that the probability of inundation at the site is sufficiently low to preclude classification within either the 1 percent annual-chance (100-year) or 0.2 percent annual-chance (500-year) flood zones (Reference 2-86). The project site is situated approximately 0.5 miles (0.8 km) upstream of the nearest mapped segment of the Boneyard Creek floodplain that is subject to a 0.2 percent annual-chance flood.

FEMA's earlier hydrologic and hydraulic analyses of Boneyard Creek, completed in 1980, yielded recurrence-interval discharge estimates and corresponding water-surface elevations for design flood events. Table 2-60 summarizes those predicted peaks flows and stages (Reference 2-87). The analysis established a 100-year flood-elevation of 721.5 feet (220 meters) (National Geodetic Vertical Datum [NGVD] 29) along the reach nearest the project location.

Topographic comparison demonstrates ample freeboard between the creek and the proposed facility. The site's ground elevation – approximately 732 feet (223 meters) (NAVD 88), equivalent to 728.4 feet (222 meters) (NGVD 29) – lies about 15 feet (4.6 meters) above the Boneyard Creek streambed elevation at Fourth Street (712.5 feet [217 meters] NGVD29) and roughly 7 feet (2.1 meters) higher than the mapped 100-year flood stage. Moreover, the parcel stands approximately 10 feet (3 meters) above the calculated 500-year flood elevation of 722 feet (220 meters) (NGVD 29), providing an additional margin of safety. The 100-year Boneyard Creek floodplain is shown in Figure 2-40 (Reference 2-1).

Historical stream-gauge records from 1948 through 2022 corroborate the limited flood risk at the site. The highest recorded instantaneous discharge for Boneyard Creek – 946 cfs (26.8 cubic meters per second) – occurred in 1993, while the peak stage of 20.1 feet (6.12 meters) (equating to an elevation of 715.1 feet [218 meters] NGVD 29) was observed in 2002 (Reference 2-88). Even under those extreme conditions, water levels remained approximately 13.3 feet (4.1 meters) below the finished-grade elevation planned for the research reactor facility.

Collectively, FEMA mapping, historical hydrologic data, and relative topographic relief demonstrate that the research reactor site is positioned well above the flood elevations associated with both the 1 percent and 0.2 percent annual-chance events. Consequently, flooding from Boneyard Creek is not anticipated to pose a significant hazard to the proposed installation.

Local heavy precipitation-induce flooding or effects from standing water will be analyzed in the OLA.

2.4.2.3 River or Stream Flooding

The U. of I. research reactor facility is planned with a finished-floor elevation that closely matches the existing ground surface, nominally 732 feet (223 meters) (NAVD 88). Beneath this grade, the Citadel building, which houses the reactor, will extend approximately 60 feet (18.3 meters) below ground surface, ensuring the protection of safety-related structures, systems, and components.

FEMA FIRM 17019C0426D (Reference 2-86) assigns the project parcel to an Area of Minimal Flood. The 500-year (0.2 percent annual-chance) flood elevation for the nearest mapped segment of Boneyard Creek is approximately 10 feet (3 meters) lower than the site grade. This vertical freeboard is further reinforced by the site's topographic prominence, which places the facility well above the highest historically recorded water-surface elevation (715.1 feet [218 meters], NGVD 29; Reference 2-88) and the 100-year flood stage (721.5 feet [220 meters], NGVD 29; Reference 2-87).

Given this substantial elevation differential, inundation of the proposed facility under regulatory design floods is not credible. Moreover, the 10-foot (3-meter) vertical separation precludes wind-generated waves on Boneyard Creek from overtopping the ground surface at the project location; wave run-up would attenuate well before reaching the finished-floor elevation. Consequently, both floodwater encroachment and wind-induced wave action are not credible threats to the facility.

As noted in Subsection 2.4.2.8, a seismically induced failure of upstream impoundments does not produce water-surface elevations exceeding the mapped 500-year flood level; thus, such an event likewise fails to threaten the site.

2.4.2.4 Surges

The U. of I. research reactor site is not adjacent to a seacoast subject to hurricanes. Consequently, surge due to probable maximum hurricane (PMH) is not a credible threat to the facility. Similarly, PMH wind and maximum windstorm-induced (non-hurricane) wave action is also not applicable to the site. Given the substantial margin that exists between the proposed facility's elevation of approximately 732 feet (223 meters) and 500-year flood elevation, surges due to wave action on a flooded Boneyard Creek are not a credible threat.

2.4.2.5 Seiches

The research reactor site is approximately 108 miles (175 km) from the nearest large body of water (Lake Michigan). Consequently, meteorologically induced seiches in inland lakes, coastal harbors, and embayments are not a credible threat to the facility.

2.4.2.6 Tsunami and Sea-Waves

Tsunami and "sea-waves" hazards would theoretically originate from Lake Michigan, located approximately 108 miles (175 km) to the north northeast of the site. The elevation of the lake in the Kenosha area is approximately 580 feet (177 meters), which is approximately 158 feet (48.2 meters) below the elevation of the research reactor site of approximately 732 feet (223 meters) While large waves may be generated in Lake Michigan, it is not a credible scenario that this wave would be greater than 158 feet (48.2 meters) and then maintain any appreciable height over the more than 108 miles (175 km) to the site. Therefore, the risk of tsunami or locally generated "sea waves" is not credible, including seismic, hillslope failure, and submarine landslide generated tsunami-like waves.

2.4.2.7 Seismically Induced Dam Failures (or Breaches)

Potential dam failures affecting the research reactor site are addressed in Section 2.4.3. Seismic risks for the site are covered in Section 2.5.3. There are not credible dam failures that could impact the research reactor facility. The only dams in Champaign County are small dams for farm ponds, small reservoirs, and low-head dams on minor streams.

2.4.2.8 Flooding Caused by Landslides

Seismically induced flooding typically is the result of landslides (above or below water) that cause flood waves. As discussed in Section 2.5.3, the site is not subject to significant seismic hazards. The U. of I. research reactor site is also not adjacent to a body of water subject to flooding caused by landslides. Dams upstream of the site that could be affected by landslides are addressed in Section 2.4.3. Dam failures induced by landslides will not cause flooding that could reach the proposed site elevation of approximately 732 feet (223 meters). Similarly, landslide-induced dam failure or overtopping would not produce runoff, surge, or seiche floods that could reach the site.

2.4.2.9 Effects of Ice Formation in Water Bodies

Ice formation on waterbodies is not considered a credible scenario in the vicinity of the U. of I. research reactor site. Therefore, there is not a credible scenario under which an ice jam derived flood event would impact the site.

2.4.3 Potential Dam Failures

There are no dams located upstream of the proposed site along Boneyard Creek. The nearest dams in the region include Greenwood Dam, Trautman Lake Dam, Sangamon Valley Dam, Lake of the Woods Dam, Conway Farm Subdivision Lake Dam, Spring Lake Dam, and Homer Lake Dam. These structures are situated approximately 10 to 15 miles (16 to 24 km) from the project site.

None of these dams possess the necessary combination of proximity and impounded water volume to present a credible flood hazard to the proposed facility. Most of these dams function as flood mitigation structures designed to capture and/or delay the release of rainfall runoff for residential developments, golf courses, and other localized areas. Their purpose is primarily to manage stormwater within their immediate vicinity rather than to regulate large-scale hydrologic flows.

Given their distance from the proposed site and their limited storage capacities, failure of any of these dams would not result in water levels sufficient to inundate the U. of I. research reactor site. This conclusion further supports the assessment that flood hazards at the site are minimal and do not require additional mitigation measures related to dam failure scenarios.

2.4.3.1 Flood Waves from Severe Breaching of an Upstream Dam

A review of regional impoundments confirms that no upstream dams exist along Boneyard Creek that could generate a flood wave capable of reaching the U. of I. research reactor site. Several small dams – Greenwood, Trautman Lake, Sangamon Valley, Lake of the Woods, Conway Farm Subdivision Lake, Spring Lake, and Homer Lake – are present within a 10 to 15 miles (16 to 24 km) radius, but they are situated outside Boneyard Creek and lack both the hydraulic head and storage volume necessary to influence water levels at the proposed facility.

These nearby structures function primarily as low-head, tail-water-controlled flood-mitigation reservoirs for residential neighborhoods, gold courses, or small recreational lakes. Their limited impoundment capacity, coupled with substantial travel distances and intervening topographic relief, effectively precludes the transmission of a consequential flood wave to the project location even under hypothetical breach scenarios caused by structural, seismic, or hydrologic failure (e.g., overtopping).

Consequently, there is no credible sequence of events in which a dam breach could elevate water-surface levels at the site sufficiently to threaten safety-related Structures, Systems, and Components (SSCs). This determination, in conjunction with the facility's 10-feet freeboard above the 500-year flood elevation, confirms that dam-failure does not represent a design-basis external hazard for the project.

2.4.3.2 Domino-Type or Cascading Dam Failures

A comprehensive review of regional impoundments confirms that no upstream dams exist within Boneyard Creek, and that nearby dams located 10 to 15 miles (16 to 24 km) from the U. of I. research reactor facility are low-head, small-volume structures whose spillways are tail-water controlled. Because of their limited storage capacities and their hydraulic separation from the project parcel, a breach of any single dam – and, by extension, a sequential or "cascading" breach of multiple dams triggered by an upstream failure – cannot generate a flood wave of sufficient magnitude to reach, let alone inundate, the site.

Consequently, there is no plausible scenario in which successive failures of several dams, precipitated by extreme initiating events such as a probable maximum flood, a landslide-induced surge, strong seismic shaking, or volcanic activity, could elevate the water surface at the proposed site beyond the design-basis flood level. The potential for cascading dam failures to compromise safety-related SSCs is therefore excluded from the set of credible external hazards.

2.4.3.3 Dynamic Effects on Structures

Based on the lack of potential dam failures that would impact the site, dynamic effects of dam failure-induced flood waves on safety-related SSCs are not a credible scenario.

2.4.3.4 Loss of Water Supply Due to Failure of a Downstream Dam

Due to facility design, the water supply of the plant would not be influenced by failure of a downstream dam. All water for facility operation is supplied by the City of Champaign public water supply system. Potable water will be supplied through the U. of I. campus water distribution system, which is serviced by the Illinois American Water Corporation (Reference 2-89). Facility water and sewer pipelines will connect to existing utility pipelines via current utility rights-of-way. Therefore, low water considerations are not applicable.

2.4.3.5 Effects of Sediment Deposition and Erosion

Due to facility design, the effects of sediment deposition or erosion during dam failure-induced flood waves that may result in blockage or loss would not influence the function of safety-related SSCs.

2.4.3.6 Failure of On-site Water Control or Storage Structures

No significant levees, are required for this facility which could induce flooding at the site. While the Reactor Cavity Cooling System has a capacity of approximately {{ }}[a(4)] an external water source is needed to fill the system and it does not require onsite water storage. Therefore, there are no credible hazards associated with on-site water control or storage.

2.4.4 Probable Maximum Surge and Seiche Flooding

2.4.4.1 Probable Maximum Hurricane

The proposed site is not adjacent to a seacoast or region subject to hurricanes. Consequently, surge due to PMH is not a credible threat to the facility.

2.4.4.2 Seiche and Resonance

Seiche and surge flooding are phenomena associated with the oscillation of water surfaces in enclosed or semi-enclosed bodies of water, typically initiated external forces such as atmospheric pressure changes, strong winds, or seismic activity (NUREG/CR-7046). These hazards are relevant to sites located along the edges of lakes, reservoirs, or other large water bodies.

The U. of I. research reactor site is not situated near any open bodies of water or lakes. The site lies approximately 1 mile (1.6 km) from Boneyard Creek, a small river, and 108 miles (175 km) from the nearest large body of water, Lake Michigan. Given this considerable distance, the site is not subject to seiche or surge flooding hazards.

Historical records further support this conclusion. The highest recorded seiche on Lake Michigan occurred on June 26, 1954, near Chicago, producing a wave height of 2 to 4 feet (0.6 to 1.2 meters). This wave height is negligible compared to the significant elevation difference between Lake Michigan and the site. The site’s elevation, approximately 732 feet (223 meters) (NAVD 88), far exceeds the elevation of Lake Michigan, rendering any potential seiche or surge effects inconsequential.

Based on the site’s location, topography, and distance from large water bodies, no credible hazards from seiche or surge flooding exist for the site.

2.4.4.3 Wave Runup

The site is not located near a large body of open water. As such, wind-induced wave run-up under PMH or probable maximum windstorm winds would not impact the site.

2.4.4.4 Effects of Sediment Erosion and Deposition

Sediment erosion and deposition during storm surge and seiche-induced waves that may result in blockage or loss of function of SSCs is not a credible scenario.

2.4.5 Probable Maximum Tsunami Hazards

The research reactor site has not been subjected to tsunami forces due to its inland location. Therefore, a tsunami is not considered a credible accident in the plant design.

2.4.6 Ice Effects

There are no reported ice jams in Champaign or Urbana, IL in the U.S. Army Corps of Engineers' Ice Jam Database (Reference 2-97). Therefore, no significant events have been recorded for Boneyard Creek.

2.4.7 Cooling Water Canals and Reservoirs

Canals and reservoirs used to transport and impound water supplied to the safety-related SSCs are not included in the design for the U. of I. research reactor site.

2.4.8 Channel Diversions

No channel diversions are included in the design of the U. of I. research reactor site.

2.4.9 Groundwater Contamination Considerations

The primary and intermediate coolants for the research reactor are helium gas and a salt-based coolant, respectively. The Helium Purification System removes chemical and radioactive impurities, including tritium, from the primary coolant (helium inventory). Tritiated concentration from this system, estimated to be less than 0.35 cubic feet (0.01 cubic meters) per year would be packaged (collected in filters) and discharged to solid radioactive waste. Normal operating liquid effluents would be discharged to sewage at a volumetric flow rate of less than 35.3 cubic feet (1 cubic meter) per day. Wastewater generated by the facility will be managed through the campus sanitary sewer system, which is a tributary to the Urbana & Champaign Sanitary District. All releases will be below the allowable sewer release limits provided in 10 CFR Part 20, Appendix B. Radioactivity monitoring for radioactive releases will include sensors to monitor for radioactivity and testing to determine release limits.

Chapter 11 provides additional information regarding the environmental monitoring that is included in the Radiation Protection Program.

Table 2-60 Summary of FEMA flood information for Boneyard Creek

Recurrence Interval, R (years)	Annual Probability, P	Peak discharge [a] (cfs)	Water surface elevation [b] (ft, NGVD 29)	Water surface elevation [c] (ft, NAVD 88)
10	0.10	596	720	723.6
50	0.02	1,006	721	724.6
100	0.01	1,264	721.5	725.1
500	0.002	1,550	722	725.6

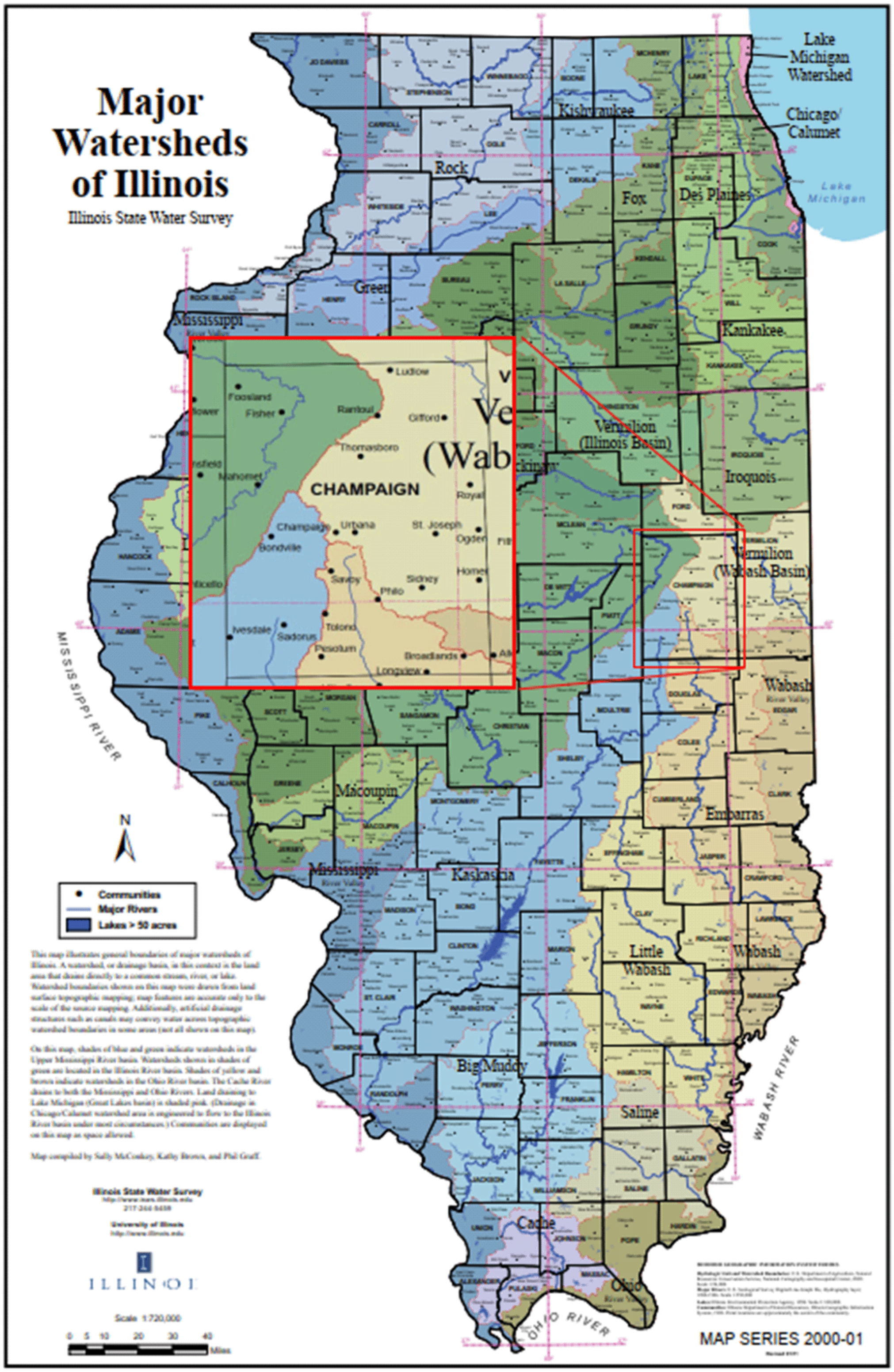
a – *Flood Insurance Study, City of Champaign, Illinois*. USGS Gauge No. 33370, Reference 2-87

b – Elevations are approximate, in NGVD 29 at Fourth Street, Reference 2-87.

c – Elevations are approximate, converted to NAVD 88 (NAVD 88 = NGVD 29 + 3.6 ft).

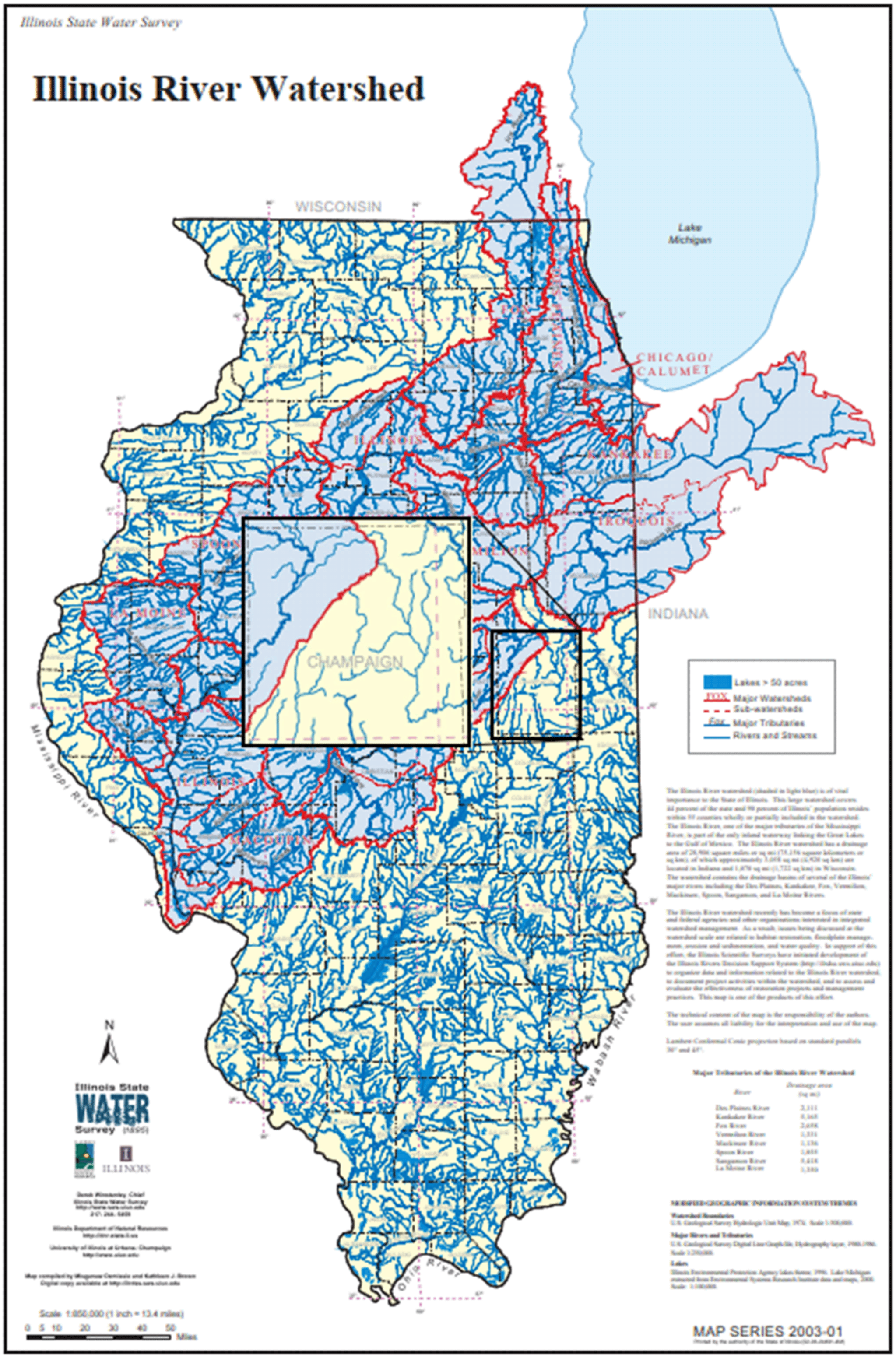
cfs – cubic feet per second

Figure 2-37 Major Watersheds of Illinois



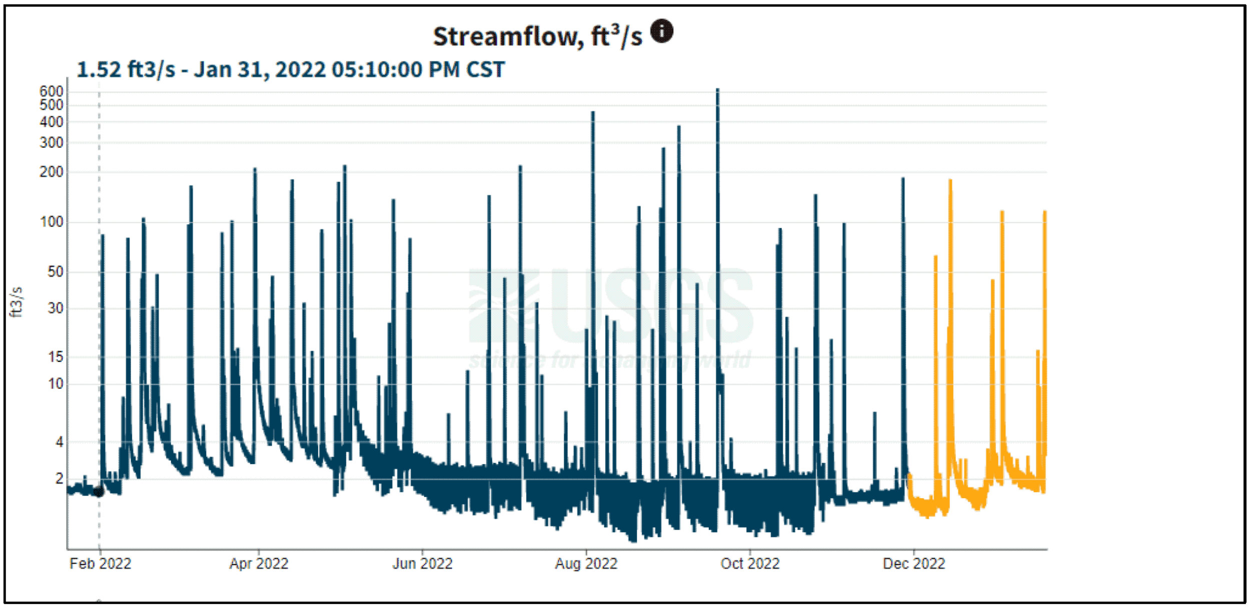
Source: Reference 2-90

Figure 2-38 Illinois River Watershed



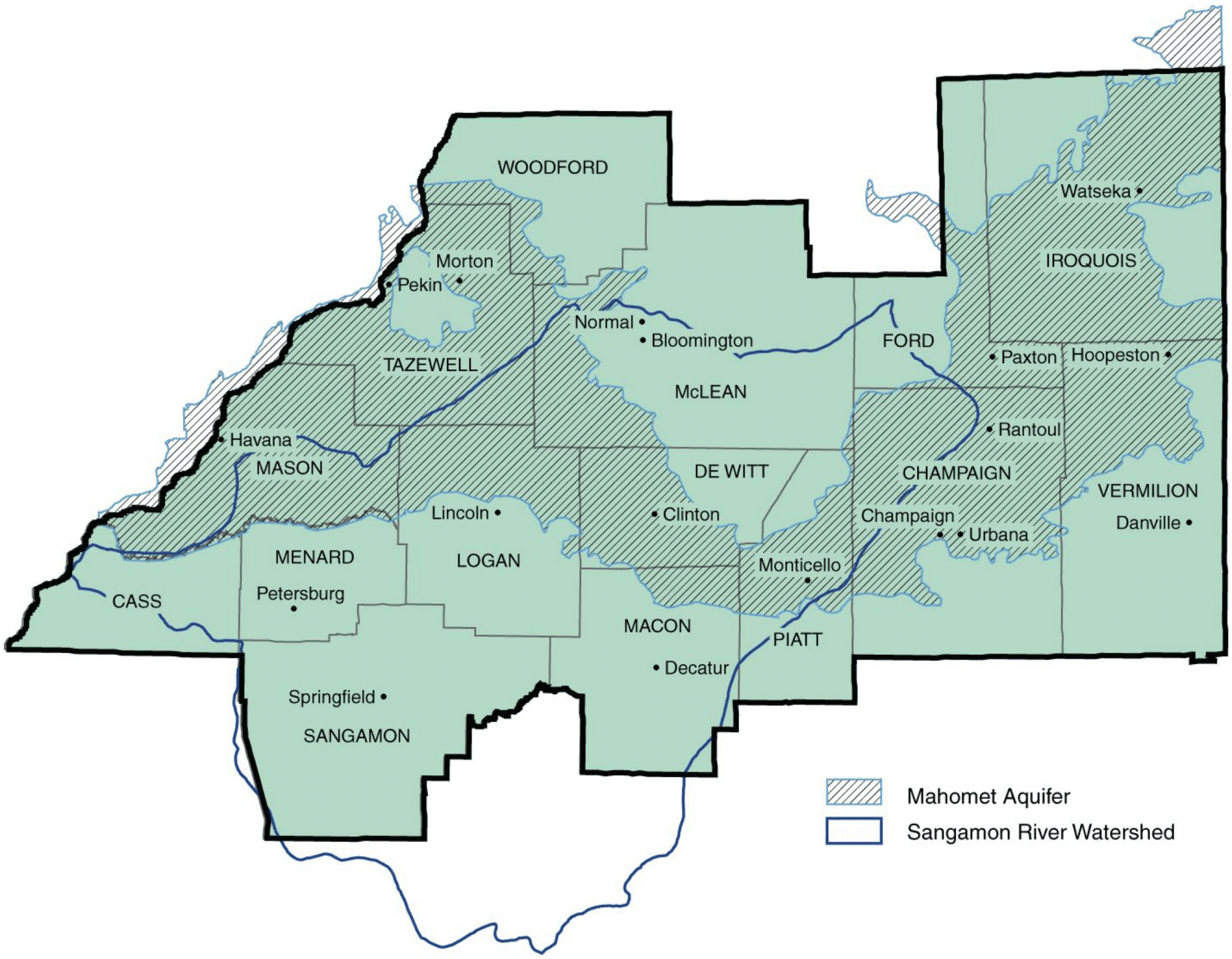
Source: Reference 2-91

Figure 2-39 12-Month Flows for USGS Gauging Station on Boneyard Creek, Urbana IL



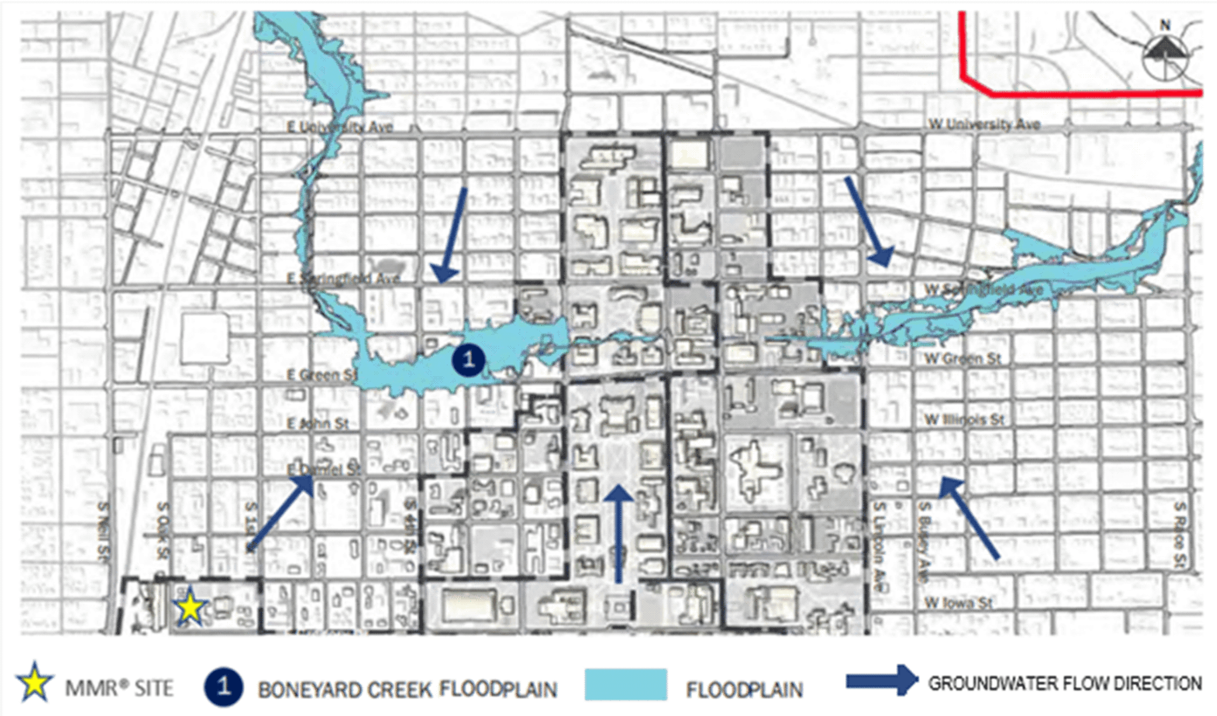
Source: Rev. 83

Figure 2-40 Mahomet Aquifer



Source: Rev. 92

Figure 2-41 Boneyard Creek Floodplain



Source: Rev. 1

2.5 GEOLOGY, SEISMOLOGY AND GEOTECHNICAL ENGINEERING

This section describes the geologic, seismic, and geotechnical characteristics of the proposed research reactor site and the surrounding region. These site characteristics provide the basis for required design inputs for SSCs. The seismic design basis reflects existing information from current seismic hazard publications and data from detailed geotechnical investigations at the site. A regional geologic and seismic overview (within approximately a 100-mile radius of the site) is included to characterize potential seismic sources affecting ground motion at the site.

2.5.1 Regional Geology

The project site at the U. of I. Urbana-Champaign, located near the GSL at South Oak Street and East Gregory Drive, is situated in central Champaign County on a glaciated Quaternary plain formed by multiple Pleistocene glacial advances. The area lies within the Bloomington Ridged Plain, which is characterized by gently rolling to nearly flat topography, with regional ground slopes on the order of approximately 0.1 degrees. The most recent glacial advance affecting the site was during the Wisconsin Glacial Episode, approximately 23,000 years ago. Earlier Illinois and pre-Illinois glaciations also contributed substantial glacial deposits that are now buried beneath younger sediments.

As summarized by Stumpf et al. (Reference 2-93), the surficial stratigraphy across much of the U. of I. campus consists of Wisconsin Episode glacial deposits that, from the ground surface downward, include the Batestown Member of the Lemont Formation, underlain by the Piatt Member of the Tiskilwa Formation, and lower undifferentiated Tiskilwa Formation till. In localized areas, these tills may be overlain by glacial outwash sands of the Henry Formation or by lacustrine clay and silt of the Equality Formation. The glacial sequence is commonly mantled by up to approximately 3 feet (0.9 meters) of Peoria Silt, a windblown loess deposit emplaced following glacial retreat. Beneath the Wisconsin-aged drift, older Illinois Glacial Episode deposits, including the Vandalia Member of the Glasford Formation, overlie pre-Illinois glacial and nonglacial materials that extend downward to an irregular bedrock surface.

Site-specific subsurface data are consistent with this regional geologic framework. Historical borings in the vicinity of the site, including those documented by Burch et al. (Reference 2-94), indicate the presence of approximately 0 to 4 feet (0 to 1.2 meters) of surficial loam or fill, locally containing cinders or coal ash associated with past industrial activity, underlain by stiff glacial till extending to depths of at least 60 to 90 feet (18.3 to 27.4 meters). The near-surface materials correspond to soils developed in Peoria Silt with minor anthropogenic fill, while the underlying unweathered tills are consistent with the Batestown Member in the upper approximately 30 feet (9.1 meters) and the Piatt Member at greater depths. Planned excavation for the facility, anticipated to extend to a depth of approximately 60 feet, is expected to remain entirely within Wisconsin Episode tills, and deposits associated with the Illinois Glacial Episode are not anticipated to be encountered.

Bedrock beneath the site consists of Pennsylvanian-age sedimentary rocks of the Illinois Basin and occurs at depths of several hundred feet below ground surface. Near the site, the uppermost bedrock is described as microgranular limestone and mudstone, with evidence of soft-sediment deformation and an erosional unconformity at the bedrock surface. Because these carbonate units are deeply buried beneath a thick sequence of glacial deposits, karst features are not present at shallow depths, and the potential for sinkholes or other karst-related ground instability is considered negligible. Although the Mahomet Aquifer, a major sand and gravel aquifer within a buried bedrock valley, underlies portions of Champaign County, it does not extend beneath the immediate project site. Overall, the site is located on a stable, nearly level glacial plain with no significant topographic relief or slope instability concern (Reference 2-95, Reference 2-96).

Geology map of the state of Illinois and Champaign is presented in Figure 2-42 and Figure 2-43.

2.5.2 Site Geology

The project site is located on the U. of I. campus in Champaign, Illinois, near the Illinois State Geological Survey Laboratory in the vicinity of South Oak Street and East Gregory Drive. The site is relatively flat, with ground surface elevations of approximately 730 to 733 feet (222.5 to 223.4 meters) and is within a previously developed area. Existing surficial conditions include a combination of paved surfaces and landscaped grass/soil (Reference 2-97).

A geotechnical investigation was completed for the site and included five Seismic Cone Penetration Tests (SCPTu) and fifteen geotechnical borings (Table 2-61). The borings were advanced using sonic drilling methods, with Standard Penetration Tests (SPTs) performed at regular intervals to characterize soil density and consistency and to support the engineering evaluation of subsurface conditions. Laboratory testing was performed on selected soil samples; however, at the time of preparation of this chapter, portions of the laboratory program remain in progress. The SCPTu soundings were advanced to a maximum depth of approximately 70 feet (21.3 meters) below ground surface (bgs), and the boring depths ranged from about 10 feet (3.0 meters) bgs to 125 feet (38.1 meters) bgs (Reference 2-97). The geotechnical exploration plan is presented in Figure 2-44, and the geophysical exploration plan is presented in Figure 2-45.

A geophysical survey was also performed at the site. The geophysical program included two electrical resistivity tests, approximately 500 linear feet (152.4 meters) of Multichannel Analysis of Surface Waves (MASW), Vs30 measurements at five locations, and two P-S suspension logs (Reference 2-97). The geophysical results are summarized in Figure 2-46 through Figure 2-49. The electrical resistivity test results are presented in Figure 2-46, the Vs30 results are presented in Figure 2-47, the MASW results are presented in Figure 2-48, and P-S Suspension logging results are presented in Figure 2-49.

The subsurface materials encountered in the soil borings are described below for the purposes of our discussions in this chapter. It should be noted that these descriptions do not imply the continuity of the materials encountered in the boring. The descriptions of the materials have been established to characterize similar subsurface conditions based on material gradations and parent geology. The material provided below is based on the laboratory results and onsite soil classification (Reference 2-97).

The subsurface materials encountered in the soil borings, along with the range of SPT N-Values which represent the number of blows required to drive a standard sampler 12 inches (30 cm). Up to approximately 4 to 8 inches (10.2 to 20.4 cm) of topsoil and 4 to 6 inches 10.2 to 12.2 cm) of asphalt were encountered at the soil boring locations. These depths should not be considered as stripping depths since the topsoil and asphalt may be thicker at other areas of the site (Reference 2-97). The subsurface soil profiles are presented in Figure 2-50 and Figure 2-51.

2.5.2.1 Existing Fill

All the soil borings indicated the presence of fill soils below the asphalt and asphalt base stone. It is possible that this uncontrolled fill may have been placed during the construction of the existing structures, although no information was available to substantiate this. The existing fill has a thickness of 1 to 10 feet (0.3 to 3.0 meters) (Reference 2-97).

SPT N-values for the uncontrolled fill soils vary between 1 blow per foot (bpf) to 9 bpf, indicating very loose to loose soils (Reference 2-97).

2.5.2.2 Stratum A – Glacial Outwash

Below the existing fill soils, sandy naturally deposited Glacial Outwash soils were encountered in all borings except for ATS-11, with the deepest depth being 45 feet (13.7 meters) bgs in boring ATS-17 (Reference 2-97).

The sandy soils of Stratum A consisted of very loose to medium dense Clayey Sand (SC), Silty Sand (SM), Well Graded Sand with Silt (SW-SM) and Poorly Graded Sand with Silt (SP-SM). The N-values ranged from Weight of Hammer bpf to 23 bpf (Reference 2-97).

The cohesive soils of Stratum A consisted of very soft to very stiff Sandy Lean Clay (CL), Lean Clay (CL), Fat Clay (CH), Silty Clay with Sand (CL-ML), Silty Clay (CL-ML) and Sandy Silt (ML). The N-values ranged from Weight of Hammer to 25 bpf (Reference 2-97).

CPTu (Cone Penetration Test) corrected tip resistance for the existing soils varies approximately between 50 tons per square foot (tsf) and 250 tsf. The shear-wave velocity (Vs) measured from the Vs30 tests (average shear-wave velocity in the top 30 meters of soil) ranged from approximately 500 to 1,500 feet per second (ft/s) (152 to 457 meters per second [m/s]) while the P-S suspension logging results ranged from approximately 800 to 1,200 ft/s (Reference 2-97).

2.5.2.3 Stratum B – Glacial Till

Glacial till was only encountered in the deeper borings (ATS-13, ATS-14, ATS-15, ATS-16A, ATS-17, ATS-18, ATS-19, ATS-20A, ATS-21A, and ATS-23), all of which were advanced to depths greater than 65 feet (19.8 meters) bgs. Glacial till was encountered in these borings to the boring termination depth of 125 feet (38.1 meters) bgs (Reference 2-97).

The sandy soils of Stratum B consisted of dense to very dense Clayey Sand (SC), Silty Sand (SM, Poorly Graded Sand with Silt (SP-SM) and Poorly Graded Sand (SP). The N-values ranged from 32 bpf to 50 blows over 2 inches (5.1 cm). The cohesive soils of Stratum B consisted of stiff to hard Sandy Lean Clay (CL), Lean Clay (CL), Fat Clay (CH), Silty Clay with Sand (CL-ML), Sandy Silt (ML) and Elastic Silt (MH). The N-values ranged from 13 to 84 bpf (Reference 2-97).

The CPTu corrected tip resistance values for the site soils range on the order of about 30 to 250 tsf. All CPTu soundings terminated at refusal well above 100 feet below ground surface, between approximately 55 and 75 feet (16.8 to 22.9 meters) bgs, where a very stiff to very dense glacial till layer was encountered that prevented further penetration (Reference 2-97).

Shear-wave velocity measurements show relatively consistent stiffness across the site. Based on the Vs30 tests, Vs ranges from approximately 1,400 to 1,800 ft/s (427 to 549 m/s). Based on the P-S suspension logging, Vs ranges from approximately 1,000 to 1,800 ft/s (305 to 549 m/s) (Reference 2-97).

2.5.2.4 Groundwater

The groundwater depths were collected from boring and temporary wells installed during the 2025 geotechnical investigation. The groundwater ranged from 11 feet (3.4 meters) bgs to 16 feet (4.9 meters) bgs (Reference 2-97).

2.5.2.5 Static and Dynamic Soil Properties

Select split-spoon soil samples were tested to determine the physical and engineering properties of the onsite soils. The soil laboratory tests were conducted by AECOM in general accordance with applicable ASTM criteria. A summary of some of the laboratory results are provided in Table 2-62 through Table 2-65. Additional laboratory results can be found in the Geotechnical Engineering Report (Reference 2-97).

2.5.3 Seismicity

The site lies within the stable interior of the North American Plate and is far removed from active plate-boundary fault systems. Even so, the central United States does experience occasional intraplate earthquakes. For Illinois and surrounding states, the principal sources of regional seismicity are the New Madrid Seismic Zone (NMSZ) and the Wabash Valley Seismic Zone (WVSZ). The NMSZ, centered in southeastern Missouri, is approximately 300 miles (483 km) south of the site and produced the 1811–1812 earthquake sequence (estimated at M7 or greater). The WVSZ, located along the Illinois-Indiana border, is approximately 150 miles southeast of the site and has generated moderate events in the instrumental record, including the April 18, 2008, Mount Carmel, Illinois earthquake (M5.2) (Reference 2-97).

Seismic activity near Champaign-Urbana is sparse and typically limited to small magnitudes. Central Illinois is generally characterized by low seismicity, and no significant earthquakes have originated in Champaign County in recorded history. Most measurable earthquakes in Illinois occur in the southern part of the state or in adjacent seismic zones. Illinois records a small number of earthquakes each year, most commonly in the M2 to M4 range, and these events are typically associated with the New Madrid or Wabash Valley source regions. Within roughly 100 miles (161 km) of the site, only a limited number of small earthquakes have been documented over the past century, and essentially all were below levels associated with damage (Reference 2-97).

The 2008 Wabash Valley event provides a useful reference point for how regional earthquakes are experienced at the site: it was felt in Champaign-Urbana but did not result in damage locally. Historical records further indicate that the largest earthquake with an epicenter in Illinois was the November 1968 event in southern Illinois (approximately M5.3). Other moderate earthquakes (about M4.0–4.5) have occurred intermittently in southern Illinois, western Indiana, and eastern Missouri. Because seismic waves propagate efficiently through the competent bedrock of the mid-continent, these earthquakes can be felt at considerable distances, even when they are not nearby (Reference 2-97).

Overall, the site is in a low-seismicity area, and earthquakes greater than about M3-4 are unlikely based on the available records, The seismic hazard is controlled primarily by distant regional sources, chiefly the New Madrid and Wabash Valley zones, which can produce infrequent, low-to-moderate ground shaking at the site (Reference 2-97).

2.5.4 Maximum Earthquake Potential

Considering the tectonic setting and historical seismicity, there are no known capable faults within the site area that could generate a site-specific earthquake. The maximum earthquake potential affecting the U. of I. site is therefore governed by the maximum credible earthquakes from the distant seismic zones. The Wabash Valley Seismic Zone has produced earthquakes up to approximately magnitude 5.5 in recorded history, and scientists estimate that this zone could potentially generate an earthquake in the upper magnitude 5 range (and less likely, low 6 range) under rare circumstances. The New Madrid Seismic Zone, although farther away, is capable of larger events (Reference 2-97). All the past earthquakes within the state of Illinois and bordering states are presented in Figure 2-52. Figure 2-53 shows the faults close to the site. The topography of the region is shown in Figure 2-54.

The great New Madrid earthquake is an extremely low-frequency event, but it represents the largest possible earthquake in the broader region. If such an event were to recur, the site could experience mild to moderate shaking despite the large distance, due to the high energy of the quake and efficient wave propagation in the Midwest (Reference 2-97).

For design and safety evaluations, the site earthquake input is based on the Safe Shutdown Earthquake (SSE) defined at the 2 percent probability of exceedance in 50 years (approximately a 2,500-year return period). This level is used because it captures the combined contribution of all relevant seismic sources that can affect the site. For Champaign, the hazard is dominated by distant regional sources, principally a large event in the New Madrid Seismic Zone and, secondarily, a moderate event in the Wabash Valley Seismic Zone, that could produce the highest ground motions at the site (Reference 2-97).

The nearest notable instrumentally recorded event in the region was the 2008 Mt. Carmel, Illinois earthquake (M5.2), located about 130 miles (209 km) from the site. The nearest recognized seismogenic source areas are the Wabash Valley faults in southern Illinois/Indiana, roughly 120 to 150 miles (193 to 241 km) away. Even the larger credible Wabash Valley events (on the order of M5.5) are expected to generate only light to moderate shaking at Champaign due to distance and attenuation. No damaging earthquakes have been recorded at the site in the modern era, and there is no evidence of prehistoric large earthquakes originating in the immediate site area; no capable faults are known at or near the site. Accordingly, the seismic design basis is established from the probabilistic seismic hazard rather than a site-specific deterministic fault source (Reference 2-97). A seismic hazard curve is provided in Figure 2-55.

For context, the largest Illinois earthquake in the past century, the 1968 southern Illinois event (M5.3), occurred more than 180 miles (290 km) from Champaign and did not result in reported damage in Champaign County. The SSE ground motions therefore conservatively bound historical experience, consistent with regulatory guidance and United States Geological Survey (USGS) seismic hazard modeling (Reference 2-97).

2.5.5 Vibratory Ground Motion

Vibratory ground motion at the site was evaluated using a site-specific probabilistic seismic hazard analysis (PSHA). The seismic model is adopted from the comprehensive seismic source characterization of the central and eastern U.S. (CEUS-SSC) developed for nuclear facilities (Reference 2-98). This was updated, particularly sources in the NMSZ, from the USGS National Seismic Hazard model. The PSHA was performed for the horizontal component of response spectral acceleration.

The Next Generation Attenuation – East (NGA-East) ground motion models (GMMs) were used in the PSHA. The NGA-East project considered 30 median GMMs and selected 19 of the GMMs, or "seed" models, for further investigation. These seed models were used to generate a continuous distribution of GMMs for each ground motion intensity measure. This resulted in 17 GMMs, which were then assigned weights for logic tree and PSHA applications (Reference 2-99).

The NGA-East ground motion models are for hard rock (Vs30 ~ 3,000 m/s) and thus need to be adjusted to be consistent with the local site conditions. The hard rock ground motions were converted using the site amplification factors for the CEUS of Stewart et al. (Reference 2-100, Reference 2-101) and Hashash et al. (Reference 2-102). Ramos-Sepulveda et al. (Reference 2-103) developed adjustment factors to allow for joint compatibility between the NGA-East ground motion models and the Stewart et al. (Reference 2-100, Reference 2-101) and Hashash et al. (Reference 2-102) site amplification factors, based on a residual analysis using an expanded CEUS ground motion dataset. The Ramos-Sepulveda et al. (Reference 2-103) adjustment factors were included in the PSHA logic tree with a weight of 0.5.

The MASW data collected at the U. of I. site was used to estimate a site average shear-wave velocity in the upper 100 feet (30 meters) of 337 m/s. ASCE 7-22, and the SPT N-values from the borings provided corroborating evidence of a predominately stiff soil profile. Based on these datasets and the site class criteria in ASCE 7-22 Table 20.2-1 (Reference 2-63), the site is classified as Site Class CD (stiff soil).

The mean site-specific seismic hazard curves for peak ground acceleration (PGA), 0.2 second spectral acceleration, and 1.0 second spectral acceleration are shown on Figure 2-55, with the mean uniform hazard spectra for a range of return periods shown on Figure 2-56 to Figure 2-60. The mean PGA at an average return period of 2,475-years is 0.15 g, with the mean 1.0 second spectral acceleration of 0.21 g. The data points for these figures are provided in Table 2-66. The hazard is controlled largely by the NMSZ and the Wabash Valley seismic source.

2.5.6 Surface Faulting

Based on the available data, there is no evidence of geologic surface faulting at or near the U. of I. site. Illinois is part of the interior cratonic platform and lacks active surface faults. Geologic mapping by the Illinois State Geological Survey shows no mapped faults in Champaign County at the surface. The nearest known faults are associated with ancient structures of the Illinois Basin, primarily in southern Illinois. For instance, the nearest mapped tectonic features include the La Salle Anticlinorium and associated faulted zones, which are over 50 miles (80.5 km) to the southwest of the site, and the Wabash Valley fault system, about 120 miles (193 km) to the southeast. These are deeply buried and ancient faults with no evidence of displacement in Quaternary deposits. Within a 150-mile (124-km) radius of the site, there are no known capable faults (as defined by NRC regulations, i.e. faults that have had movement at or near the surface within the last 35,000 years or that show recurrence within the last 500,000 years) based on the available records. In particular, the New Madrid fault system in Missouri and the Wabash Valley faults in southern Illinois/Indiana, while sources of seismic activity, have not produced surface fault ruptures in Illinois. The New Madrid faults produced ground fissures and sand blows in the Mississippi floodplains during the 1811–12 earthquakes, but those effects were over 300 miles (483 km) from this site. No such features have been observed in central Illinois (Reference 2-97).

A review of geologic and geomorphic data for the Champaign-Urbana area found no lineaments or features suggestive of faulting or differential crustal movement. The region's geology (glacial deposits over flat sedimentary rock) shows continuous, undisturbed stratigraphy in well logs and outcrops. Therefore, the potential for tectonic surface fault rupture at the site is low based on the available data. These data indicate that no specific design features are needed to accommodate surface displacement. This conclusion is consistent with regional seismic hazard assessments and the USGS Quaternary fault database, which contains no entries for active faults in this part of Illinois. Any minor earthquake activity in the area is due to deep-seated slip-on basement faults far below the surface, which does not induce surface rupture (Reference 2-97).

2.5.7 Liquefaction Potential

Liquefaction is a phenomenon in which saturated, loose, granular soils lose strength as excess pore-water pressure develops during seismic shaking. The potential for liquefaction depends on several factors, including soil type and gradation, relative density, plasticity, groundwater conditions, and the intensity and duration of earthquake loading. For this site, liquefaction analyses were performed using peak ground accelerations (PGAs) developed from the site-specific seismic hazard analyses.

Subsurface conditions at the site consist primarily of cohesive glacial deposits with limited sandy intervals. Below the existing fill, Stratum A consists of glacial outwash soils encountered in all borings except ATS-11, extending locally to about 45 feet bgs. The sandy soils in Stratum A include SC, SM, SW-SM, and SP-SM soils and range from very loose to medium dense, with SPT N-values from WOH to 23 blows per foot. The cohesive soils in Stratum A include CL, CH, CL-ML, and ML soils and range from very soft to very stiff, with SPT N-values from WOH to 25 blows per foot.

Stratum B consists of glacial till encountered in deeper borings below about 65 feet bgs and extending to boring termination depths of up to 125 feet bgs. The sandy soils within Stratum B are dense to very dense, with SPT N-values ranging from 32 blows per foot to 50 blows over 2 inches, while the cohesive soils are stiff to hard, with SPT N-values ranging from 13 to 84 blows per foot. CPTu corrected tip resistance values range from about 30 to 250 tsf, and all CPTu soundings terminated at refusal between about 55 and 75 feet bgs upon encountering a very stiff to very dense glacial till layer. Shear-wave velocity measurements further support the relatively stiff nature of the site soils, with Vs30 values ranging from about 500 to 1,500 ft/s in the shallower soils and about 1,400 to 1,800 ft/s in the deeper till. Groundwater levels measured during the 2025 investigation ranged from about 11 to 16 feet bgs.

Overall, the site conditions are not considered favorable for liquefaction. The subsurface profile is dominated by cohesive soils and dense glacial till, and the sandy soils are generally limited, discontinuous, and not laterally extensive. The deeper granular soils exhibit high SPT resistance and high CPTu tip resistance, indicating dense to very dense conditions with low liquefaction susceptibility. No documented liquefaction-related ground failures are known in the Champaign area. Therefore, based on the site-specific hazard analyses and the available geotechnical and geophysical data, the potential for seismically induced liquefaction or significant strength loss at the site is considered low.

Table 2-61 Summary of Borings

Boring Number	**Boring Depth**
ATS-08	10
ATS-09	10
ATS-10	10
ATS-11	10
ATS-12A	10
ATS-13	125
ATS-14	125
ATS-15	125
ATS-16A	33.2
ATS-17	125
ATS-18	100
ATS-19	100
ATS-20A	100
ATS-21A	100
ATS-23	100
CPTu Number	**Boring Depth**
CPTu-01	62.6
SCPTu-02	57.3
CPTu-03A	59.0
CPTu-04	56.3
SCPTu-05	72.0

Source: Reference 2-97

Table 2-62 Summary of Unconsolidated Undrained Test Results

Boring ID	Sample ID	Depth [feet]	USCS	Water Content [%]	Fines Content [%]	Liquid Limit	Plastic Limit	Undrained Shear Strength [psf]
ATS-16A	T-01	10	Lean Clay	14.9	73.4	31	12	1142
ATS-18	T-01	15	Silty Clay	20.8	93.3	21	15	1214
ATS-19	T-01	40	Lean Clay	11.9	63.3	19	11	2390
ATS-20A	T-01	40	Lean Clay	11.6	51.3	20	11	1779

Source: Reference 2-97

psf pounds per square foot

USCS Unified Soil Classification System

Table 2-63 Summary of Consolidated Undrained Test Results

Boring ID	Sample ID	Depth [feet]	USCS	Water Content [%]	Fines Content [%]	Liquid Limit	Plastic Limit	Friction Angle []	Cohesion [psf]
ATS-13	T-01	45	Lean Clay	11.6	56.9	20	11	31.6	295
ATS-14	T-01	50	Lean Clay	12.1	66.0	19	11	30.4	445
ATS-15	T-01	40	Lean Clay	11.4	63.3	19	11	37.8	115

Source: Reference 2-97

° degree

psf pounds per square foot

USCS Unified Soil Classification System

Table 2-64 Summary of Consolidation Summary Test Results

Boring ID	Sample ID	Depth [feet]	USCS	Water Content [%]	Fines Content [%]	Liquid Limit	Plastic Limit	OCR	Cr
ATS-13	T-01	45	Lean Clay	11.6	56.9	20	11	4.3	0.031
ATS-14	T-01	50	Lean Clay	12.1	66.0	19	11	4.1	0.046
ATS-18	T-01	15	Silty Clay	20.8	93.3	21	15	12.6	0.019
ATS-19	T-02	55	Lean Clay	10.6	66.4	22	11	3.8	0.036

Source: Reference 2-97

Cr Recompression Index

OCR Over Consolidation Ratio

USCS Unified Soil Classification System

Table 2-65 Summary of Cyclic Test Results

Boring ID	Sample ID	Depth [feet]	USCS	Water Content [%]	Fines Content [%]	Liquid Limit	Plastic Limit	CSR	Effective Stress [psf]	PGA	Number of Cycles
ATS-13	S-04	20	Sandy Silt	16.1	54.4	-	-	0.11	2188	0.17	32
ATS-21A	T-01	15	Silty Sand	16.0	16.5	-	-	0.18	1875	0.31	47
ATS-17	T-01	20	Silt	21.6	73.6	NP	NP	0.22	2188	0.35	21
ATS-18	S-05	30	Poorly Graded Sand	15.0	2.9	-	-	0.35	2814	0.52	1

Source: Reference 2-97

CSR Cyclic Stress Ratio

ML sandy silt

NP Non-Plastic

PGA peak ground acceleration

psf pounds per square foot

USCS Unified Soil Classification System

Table 2-66 Spectra Data

HORIZONTAL						VERTICAL					
	AFE:	0.001	0.000404	0.0002	0.0001		**AFE:**	0.001	0.000404	0.0002	0.0001
	Return Period (yrs):	1000	2475	5000	10000		**Return Period (yrs):**	1000	2475	5000	10000
T (s)	**Freq (Hz)**	**SA (g)**	**SA (g)**	**SA (g)**	**SA (g)**	**T (s)**	**Freq (Hz)**	**SA (g)**	**SA (g)**	**SA (g)**	**SA (g)**
PGA		9.396E-02	1.452E-01	1.969E-01	2.612E-01	PGA		5.630E-02	8.680E-02	1.180E-01	1.570E-01
0.010	100.000	1.023E-01	1.596E-01	2.186E-01	2.950E-01	0.010	100.000	6.070E-02	9.530E-02	1.310E-01	1.760E-01
0.020	50.000	1.160E-01	1.845E-01	2.576E-01	3.559E-01	0.020	50.000	6.910E-02	1.110E-01	1.550E-01	2.150E-01
0.025	40.000	1.264E-01	2.019E-01	2.832E-01	3.931E-01	0.025	40.000	7.520E-02	1.210E-01	1.710E-01	2.390E-01
0.030	33.333	1.341E-01	2.139E-01	3.010E-01	4.179E-01	0.030	33.333	8.000E-02	1.280E-01	1.820E-01	2.550E-01
0.040	25.000	1.550E-01	2.465E-01	3.467E-01	4.793E-01	0.040	25.000	9.290E-02	1.480E-01	2.110E-01	2.940E-01
0.050	20.000	1.746E-01	2.784E-01	3.906E-01	5.371E-01	0.050	20.000	1.040E-01	1.650E-01	2.340E-01	3.230E-01
0.075	13.333	1.987E-01	3.107E-01	4.284E-01	5.794E-01	0.075	13.333	1.170E-01	1.820E-01	2.500E-01	3.380E-01
0.100	10.000	2.274E-01	3.507E-01	4.761E-01	6.320E-01	0.100	10.000	1.280E-01	1.970E-01	2.650E-01	3.490E-01
0.150	6.667	2.387E-01	3.621E-01	4.829E-01	6.277E-01	0.150	6.667	1.320E-01	1.980E-01	2.620E-01	3.360E-01
0.200	5.000	2.368E-01	3.585E-01	4.751E-01	6.124E-01	0.200	5.000	1.280E-01	1.910E-01	2.520E-01	3.210E-01
0.300	3.333	2.239E-01	3.411E-01	4.516E-01	5.797E-01	0.300	3.333	1.230E-01	1.870E-01	2.480E-01	3.170E-01
0.400	2.500	2.067E-01	3.153E-01	4.180E-01	5.367E-01	0.400	2.500	1.160E-01	1.770E-01	2.350E-01	3.020E-01
0.500	2.000	1.897E-01	2.899E-01	3.846E-01	4.947E-01	0.500	2.000	1.080E-01	1.650E-01	2.200E-01	2.830E-01
0.750	1.333	1.570E-01	2.426E-01	3.260E-01	4.214E-01	0.750	1.333	9.290E-02	1.440E-01	1.930E-01	2.490E-01
1.000	1.000	1.310E-01	2.077E-01	2.786E-01	3.612E-01	1.000	1.000	7.820E-02	1.240E-01	1.670E-01	2.160E-01
1.500	0.667	9.674E-02	1.536E-01	2.064E-01	2.659E-01	1.500	0.667	5.390E-02	8.670E-02	1.170E-01	1.530E-01
2.000	0.500	8.005E-02	1.265E-01	1.705E-01	2.194E-01	2.000	0.500	4.460E-02	7.100E-02	9.680E-02	1.260E-01
3.000	0.333	5.403E-02	8.683E-02	1.171E-01	1.529E-01	3.000	0.333	3.010E-02	4.890E-02	6.660E-02	8.820E-02
4.000	0.250	3.937E-02	6.447E-02	8.698E-02	1.126E-01	4.000	0.250	2.200E-02	3.630E-02	4.950E-02	6.520E-02
5.000	0.200	3.144E-02	5.169E-02	7.055E-02	9.149E-02	5.000	0.200	1.750E-02	2.910E-02	4.010E-02	5.280E-02
7.500	0.133	2.106E-02	3.459E-02	4.789E-02	6.271E-02	7.500	0.133	1.180E-02	1.950E-02	2.730E-02	3.620E-02
10.000	0.100	1.328E-02	2.278E-02	3.153E-02	4.110E-02	10.000	0.100	7.410E-03	1.280E-02	1.790E-02	2.370E-02

Notes:

AFE	annual frequency of exceedance	T	time period
g	gravitational acceleration (m/s^2)	s	seconds
Hz	hertz	SA	seismic activity
PGA	peak ground acceleration	Yrs	years

Figure 2-42 Illinois Geology Map

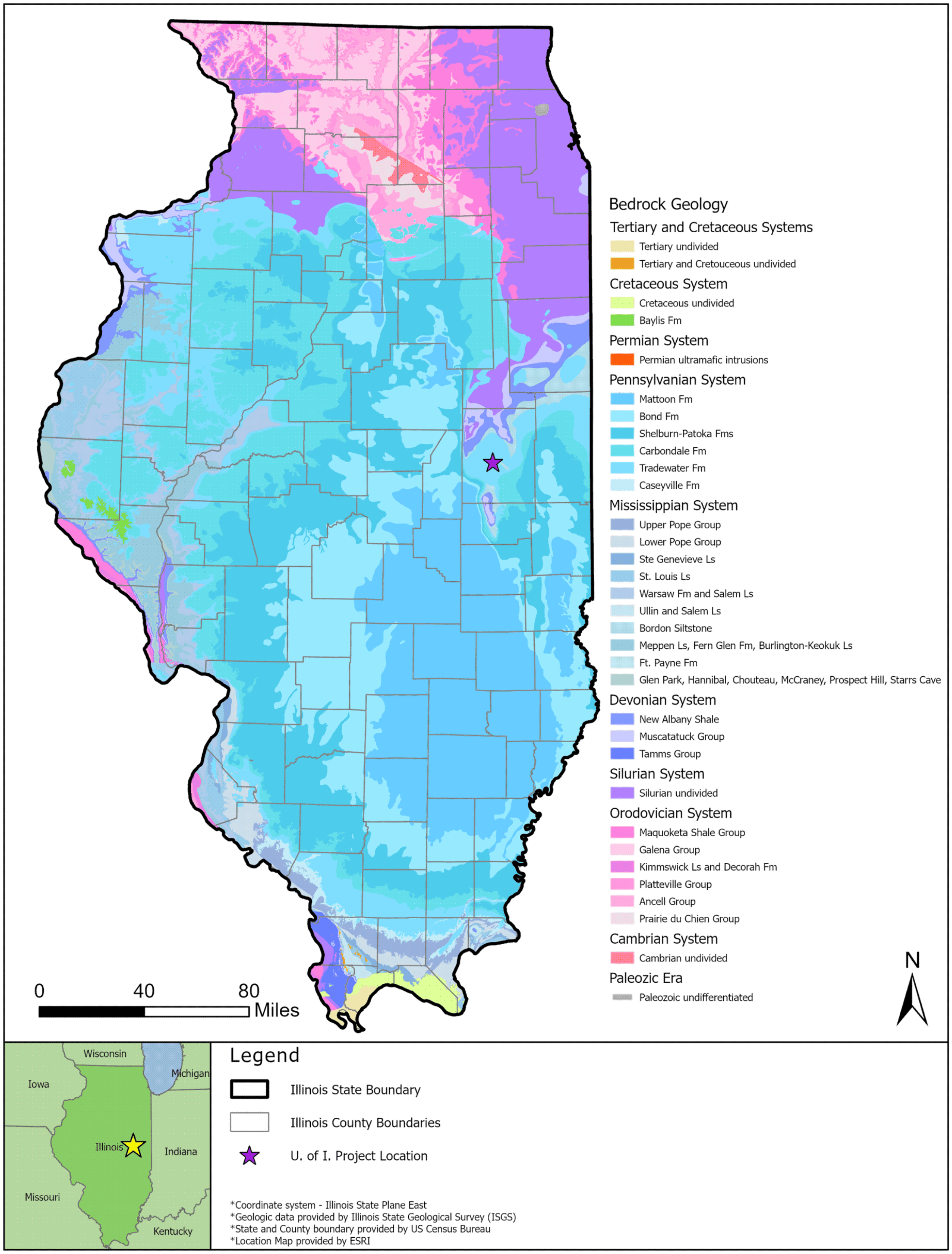


Figure 2-43 Champaign County Geology Map

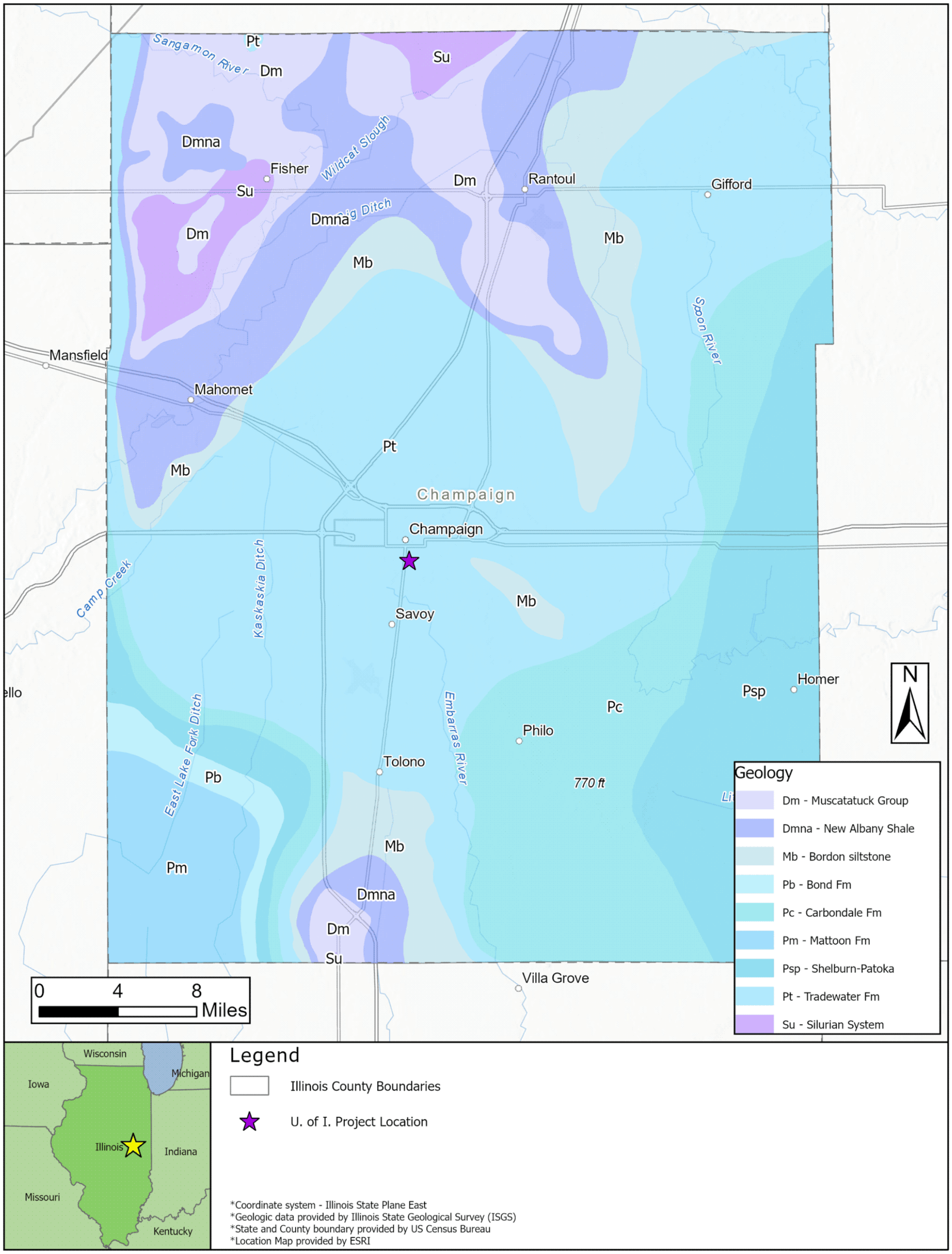
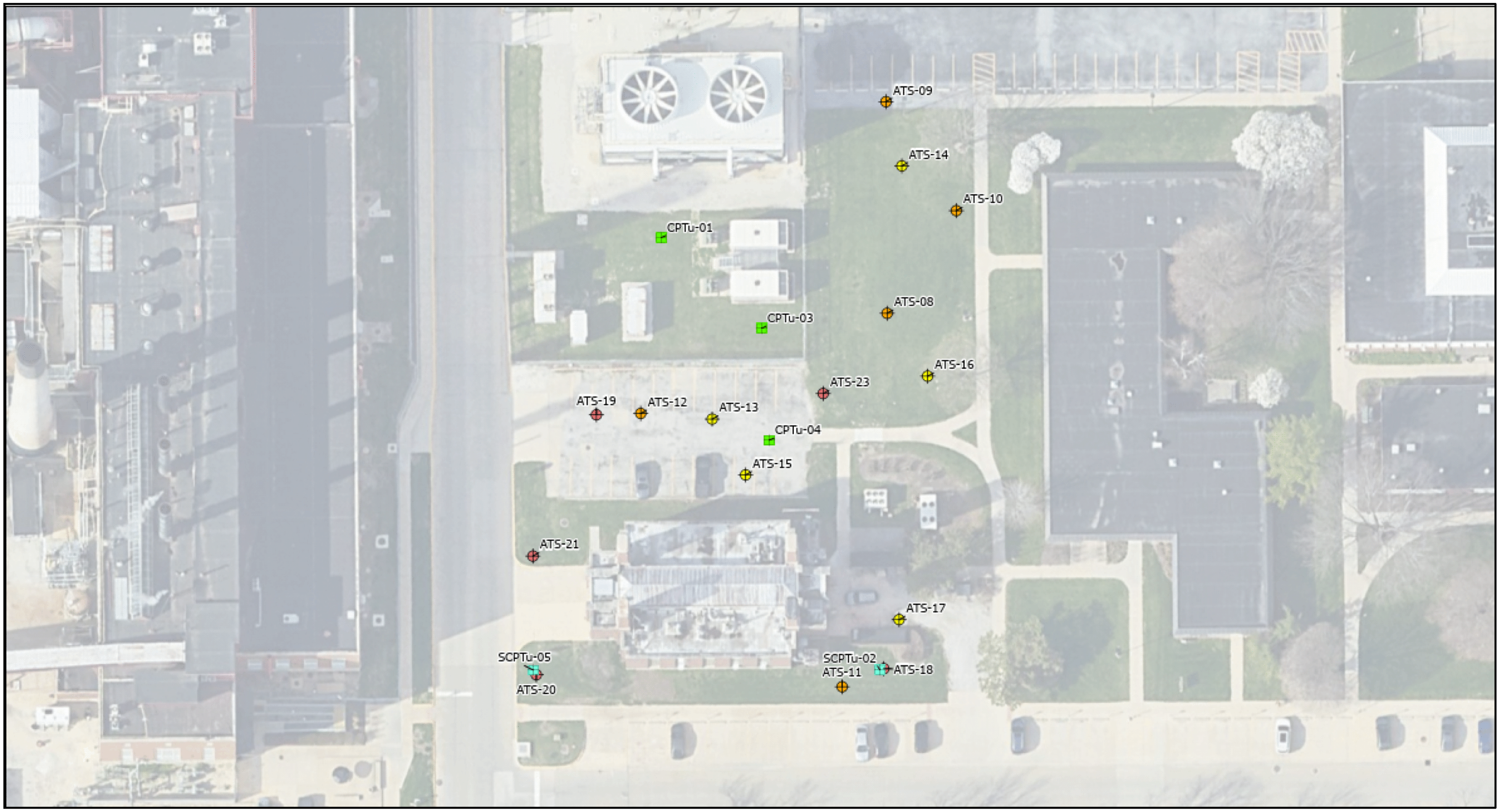
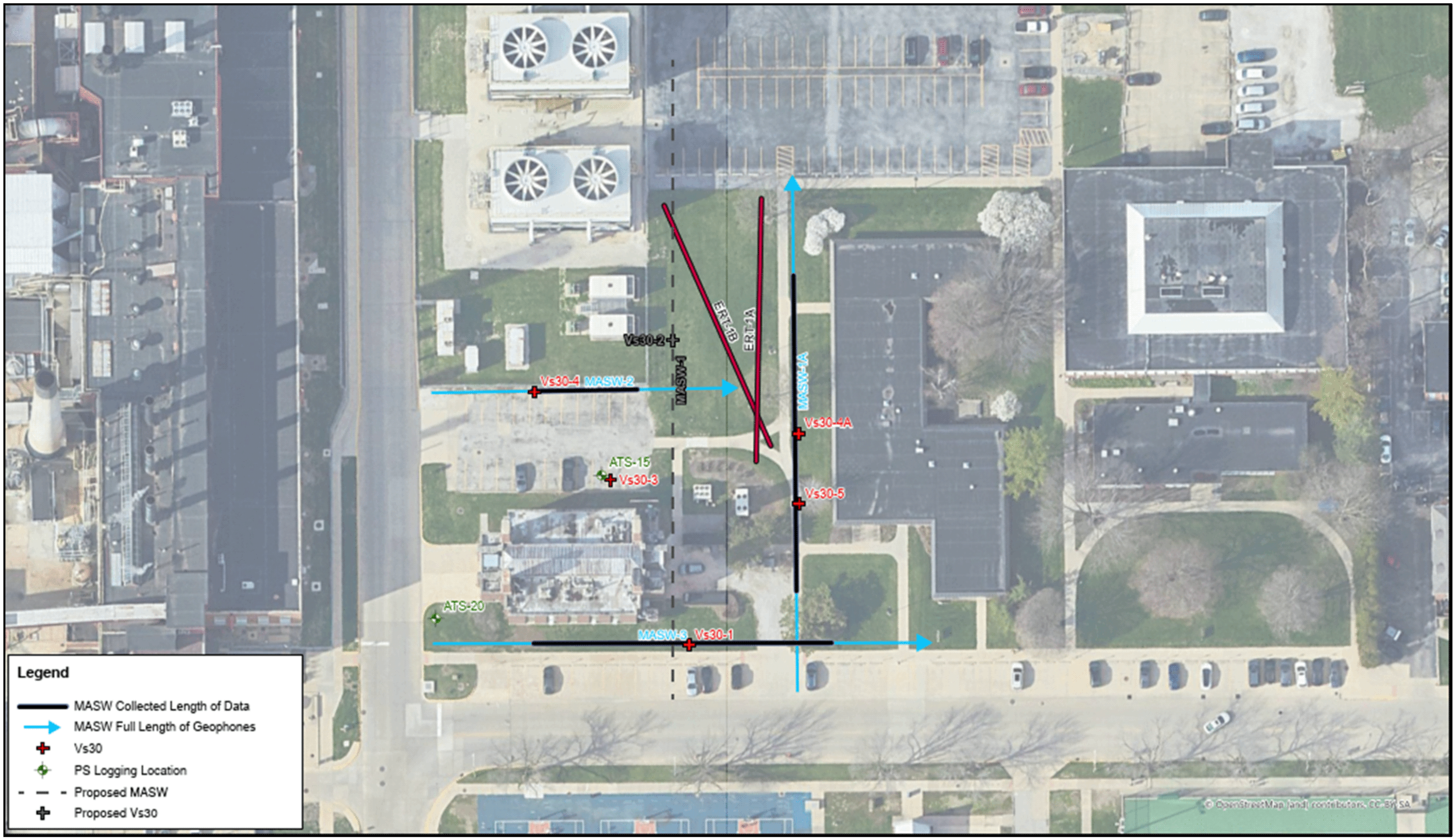


Figure 2-44 Geotechnical Exploration Plan



Source: Reference 2-97

Figure 2-45 Geophysical Exploration Plan



Source: Reference 2-97

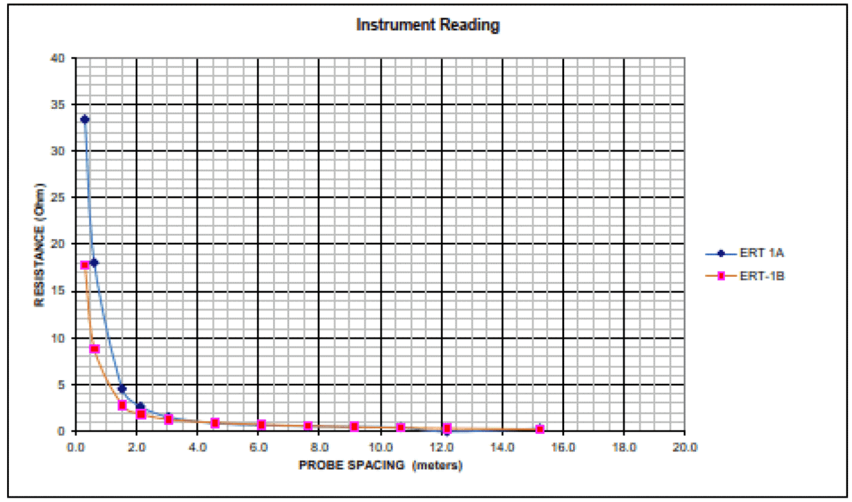
Figure 2-46 Electrical Resistivity Results

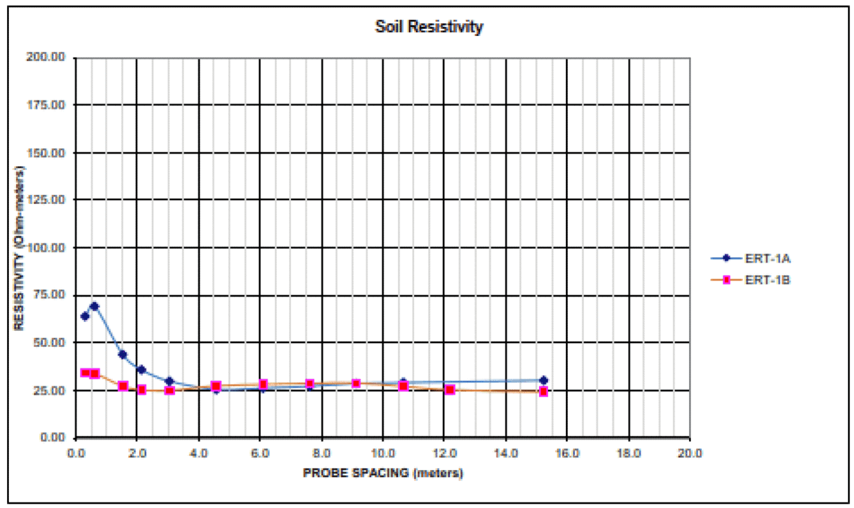
SOIL RESISTIVITY DATA - WENNER METHOD

Test Location: **UIUC**
Test Name: Top Table (ERT-1A); Bottom Table (ERT-1B)
Orientation

TEST POINT	PROBE A Spacing (ft)	PROBE A Spacing (m)	RESISTANCE (Ohm)	RESISTIVITY (Ohm-meter)
1	1	0.3	33.4	63.96
2	2	0.6	18.076	69.24
5	5	1.5	4.581	43.87
7	7	2.1	2.673	35.83
5	10	3.0	1.559	29.86
6	15	4.6	0.885	25.42
7	20	6.1	0.682	26.12
8	25	7.6	0.567	27.15
9	30	9.1	0.498	28.61
10	35	10.7	0.434	29.09
11	40	12.2	NA	
12	50	15.2	0.316	30.26

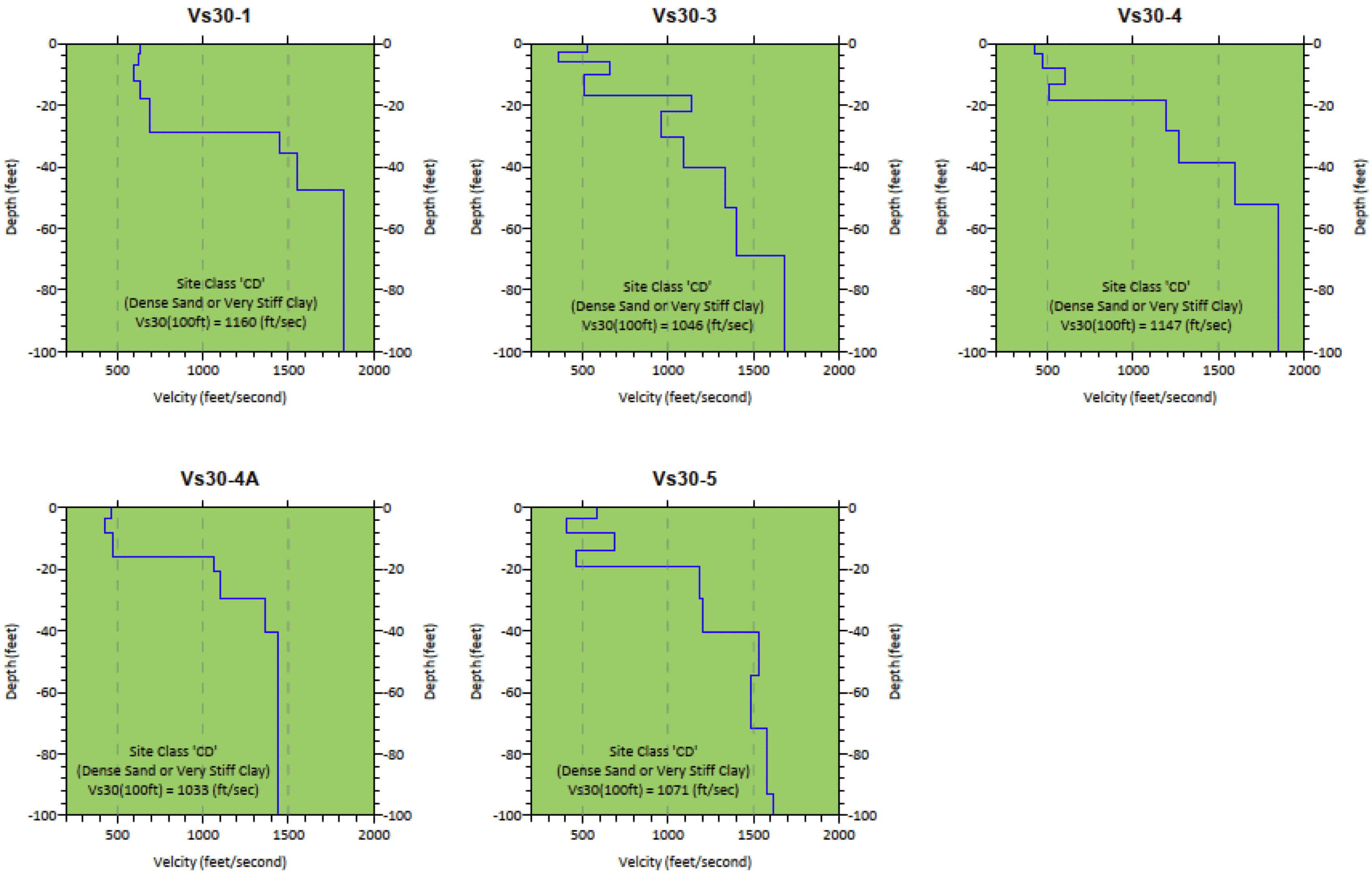
TEST POINT	PROBE A Spacing (ft)	PROBE A Spacing (m)	RESISTANCE (Ohm)	RESISTIVITY (Ohm-meter)
1	1	0.3	17.802	34.09
2	2	0.6	8.86	33.94
3	5	1.5	2.845	27.24
4	7	2.1	1.883	25.24
5	10	3.0	1.301	24.92
6	15	4.6	0.952	27.35
7	20	6.1	0.734	28.11
8	25	7.6	0.598	28.63
9	30	9.1	0.502	28.84
10	35	10.7	0.403	27.01
11	40	12.2	0.329	25.20
12	50	15.2	0.253	24.23





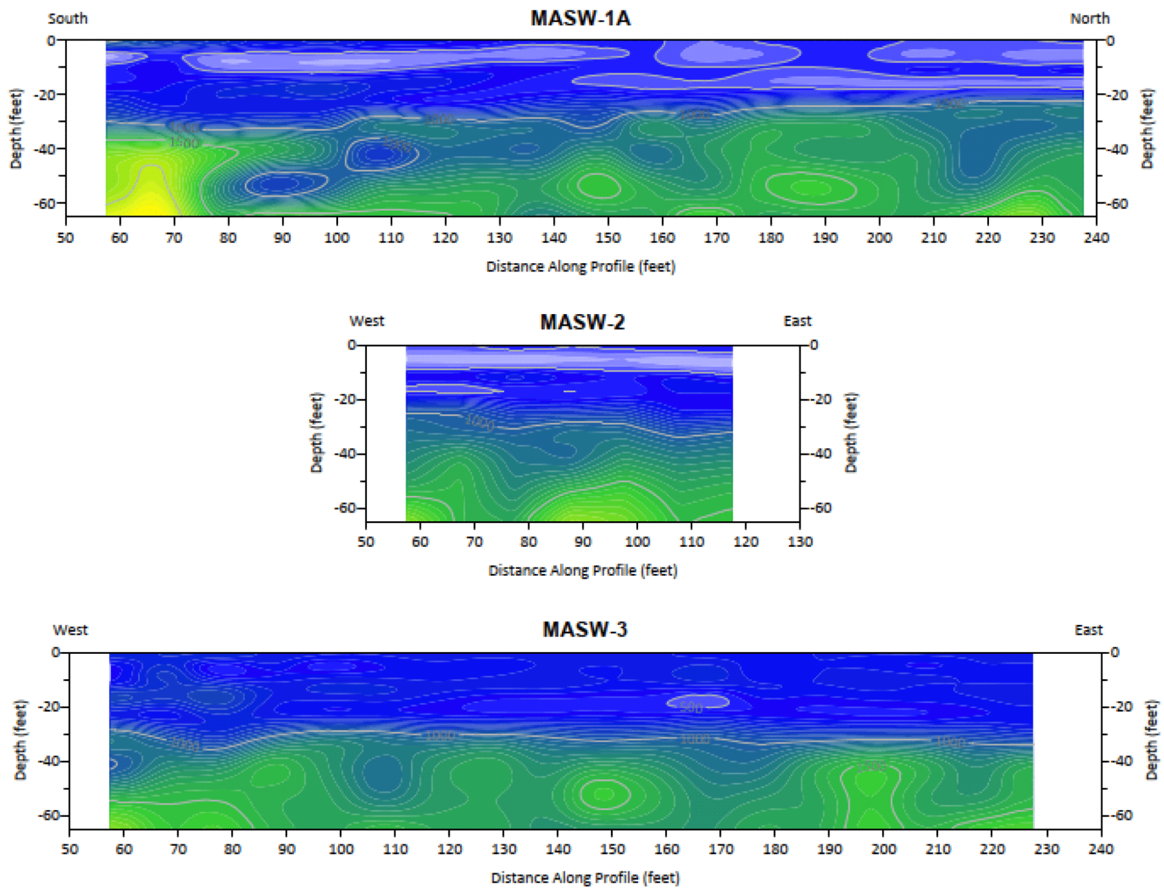
Source: Reference 2-97

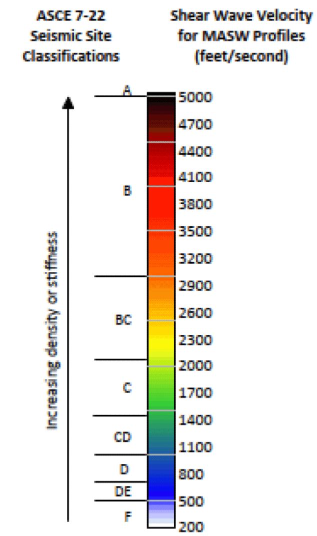
Figure 2-47 Vs30 Results



Source: Reference 2-97

Figure 2-48 MASW Results



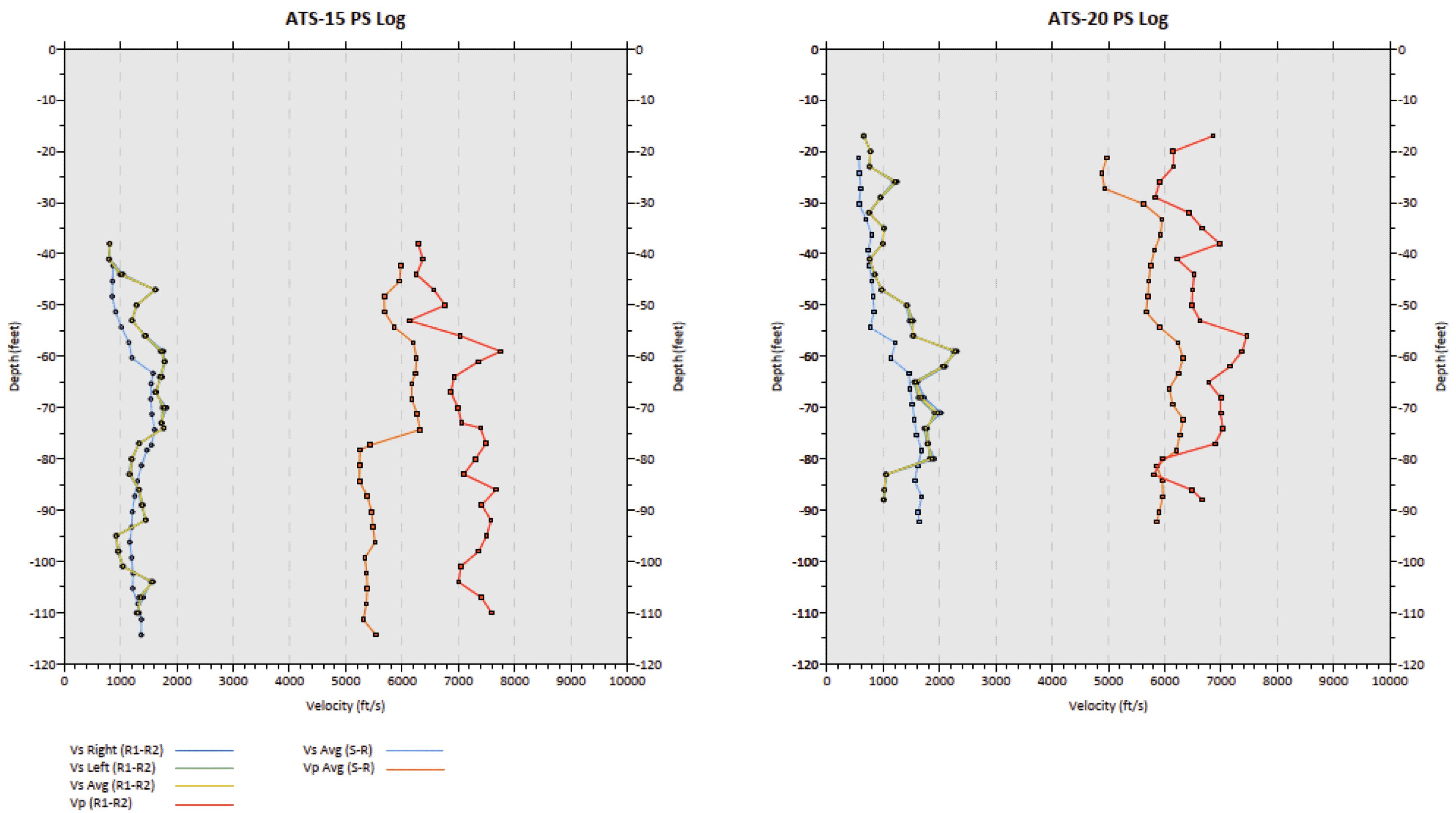


Note: ASCE 7-22 site classes are based on V_{S30} (30-m weighted average). Colored velocity ranges shown here correspond to ASCE Vs thresholds for reference only and do not represent layer-by-layer site class assignments.

Note: MASW profile distance is referenced to the midpoint of each 115-ft geophone spread. Data were acquired using a roll-along geometry, with each new record advanced by 10 feet spacing. Shot locations were placed relative to the active spread, but the displayed point for each record represents the center of the 115-ft array.

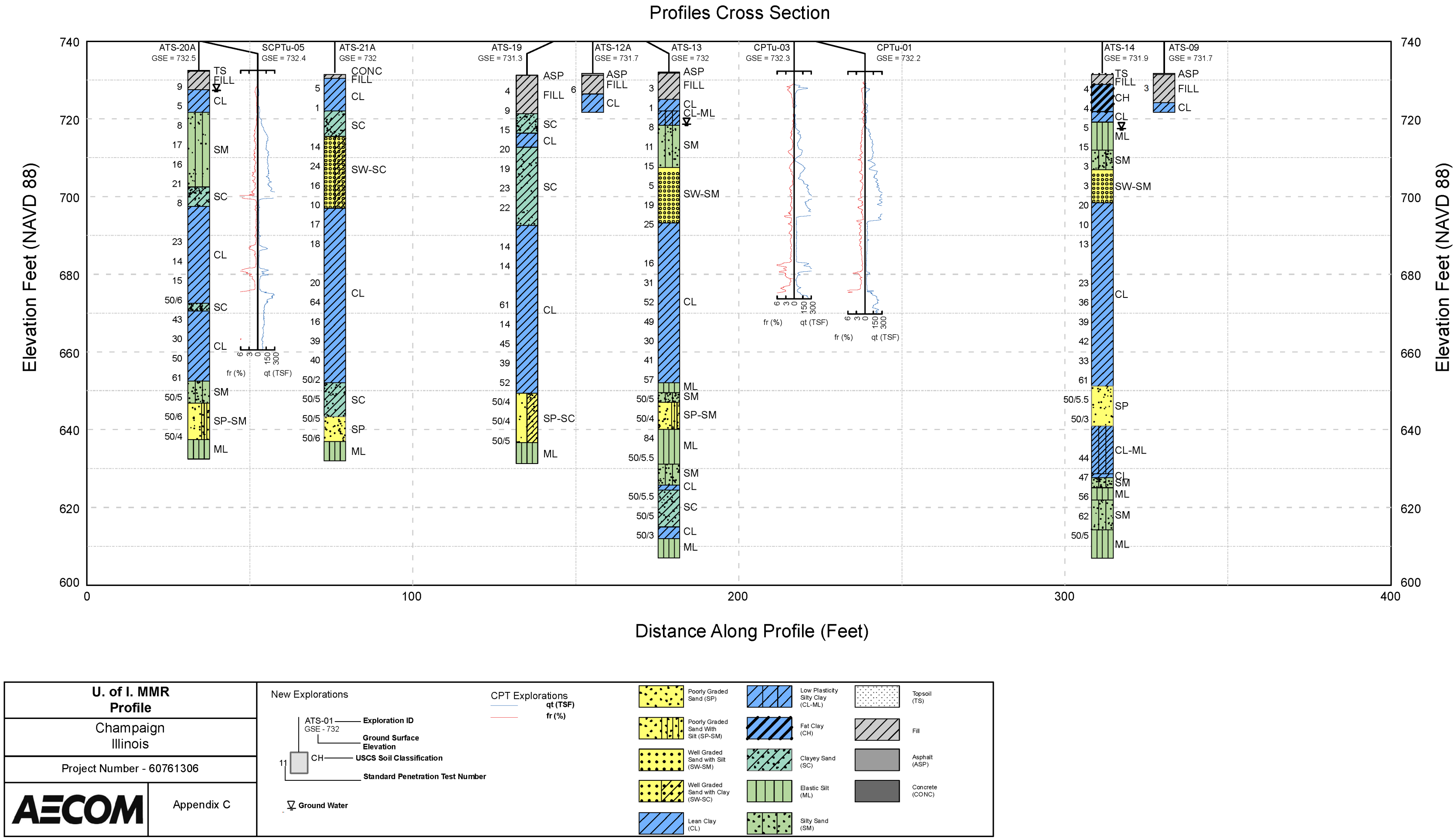
Source: Reference 2-97

Figure 2-49 P-S Suspension Logging Results



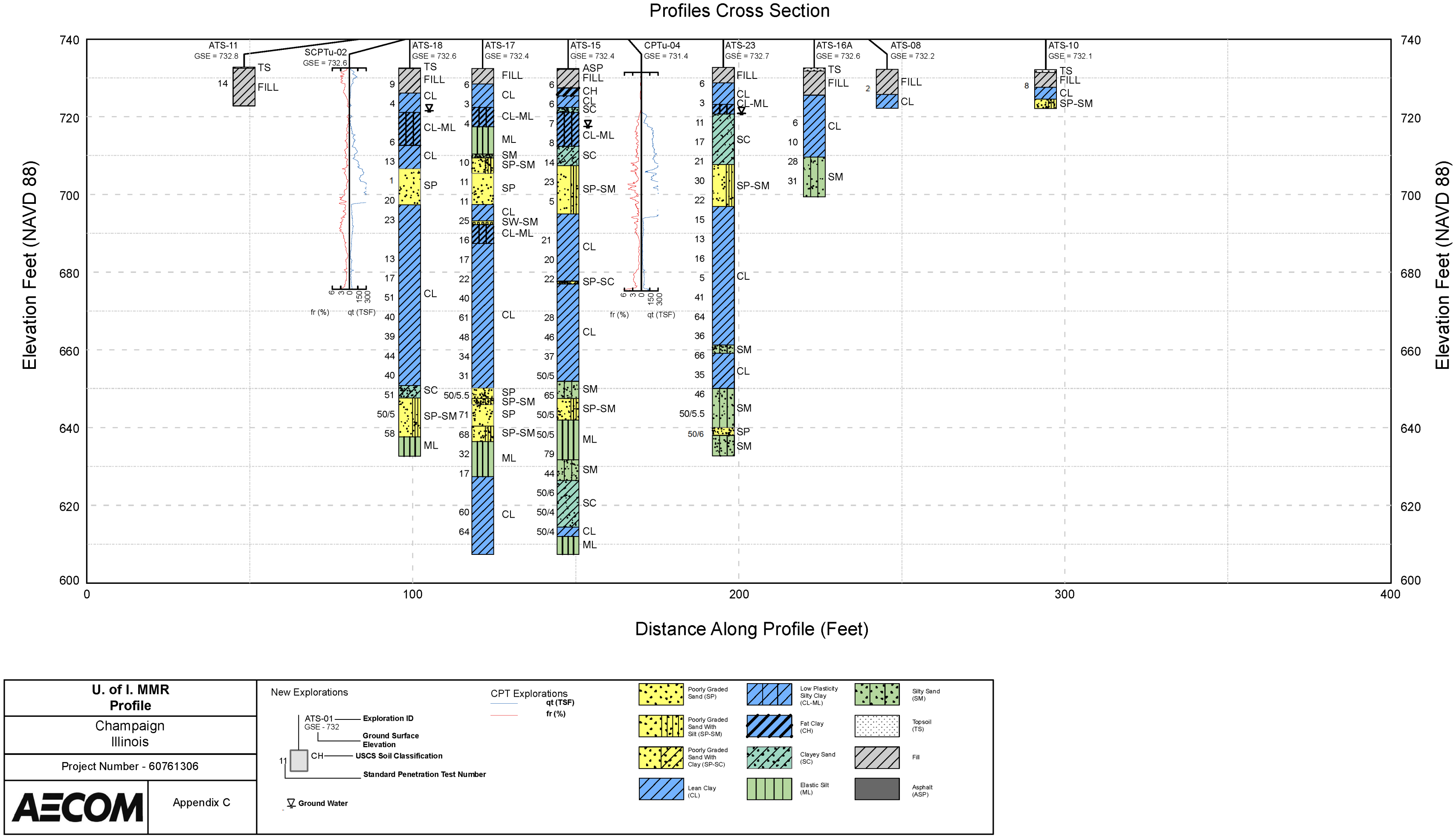
Source: Reference 2-97

Figure 2-50 Subsurface Soil Profile



Source: Reference 2-97

Figure 2-51 Subsurface Soil Profile



Source: Reference 2-97

Figure 2-52 Seismic Activity Map

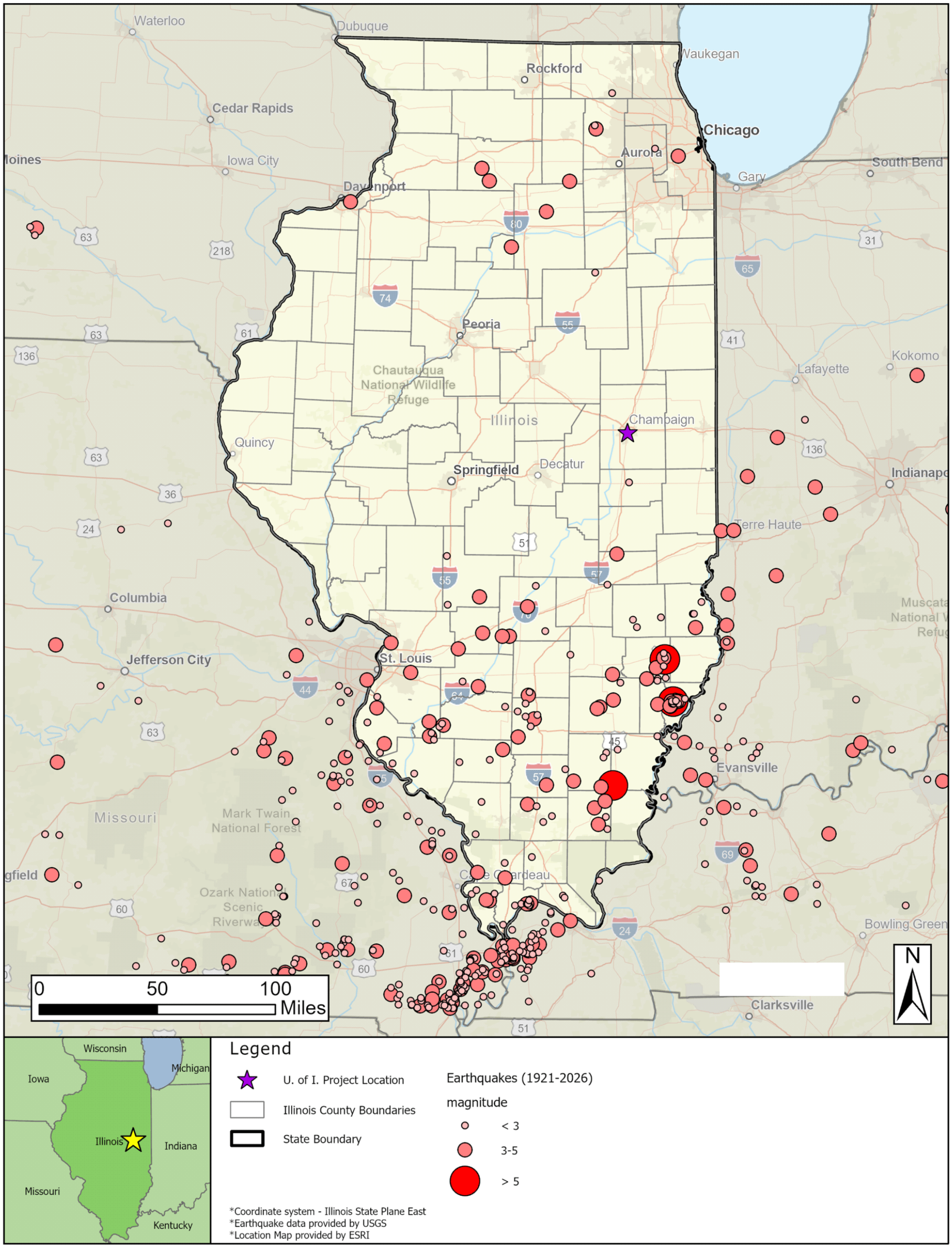


Figure 2-53 Bedrock Faults

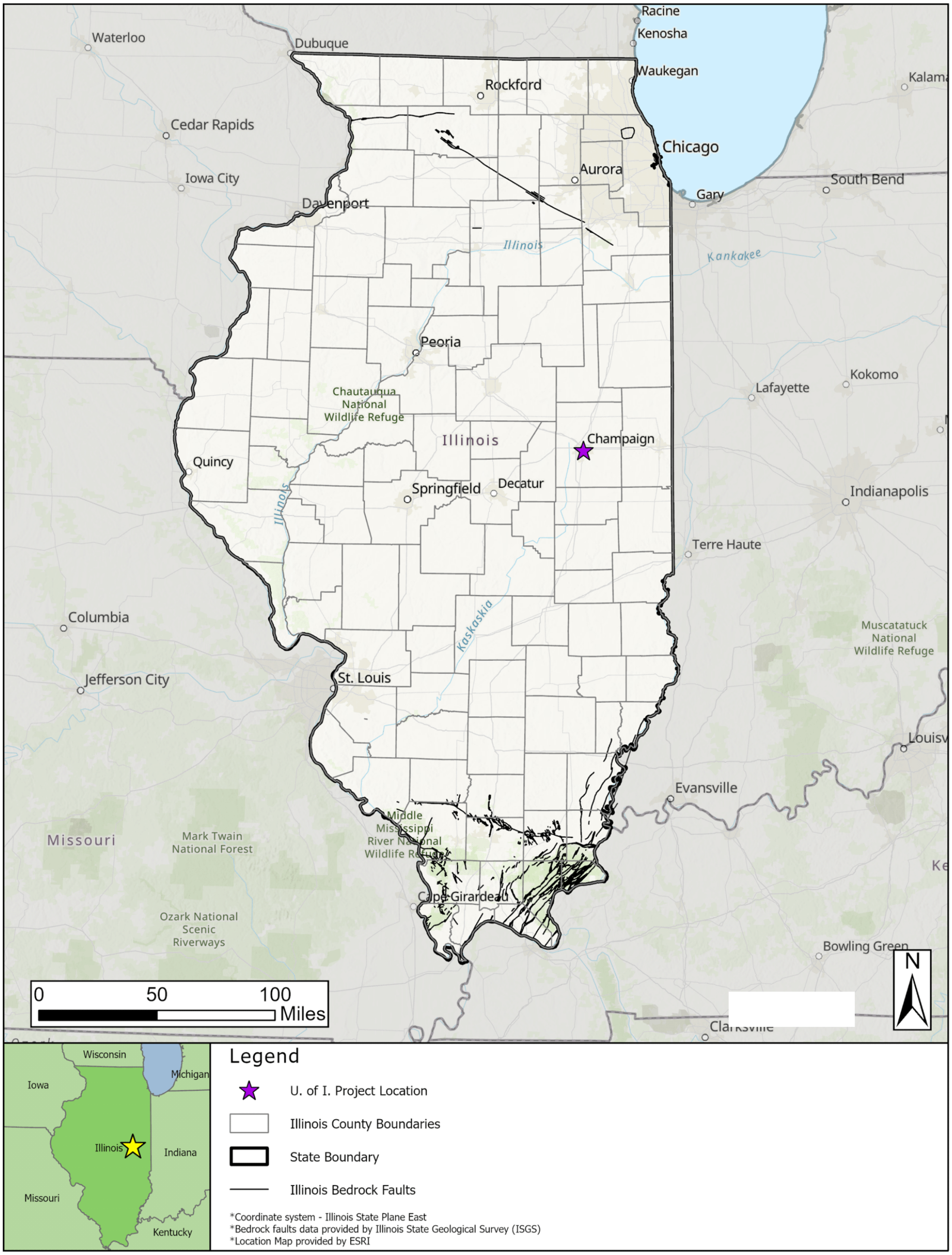


Figure 2-54 Topographic Map

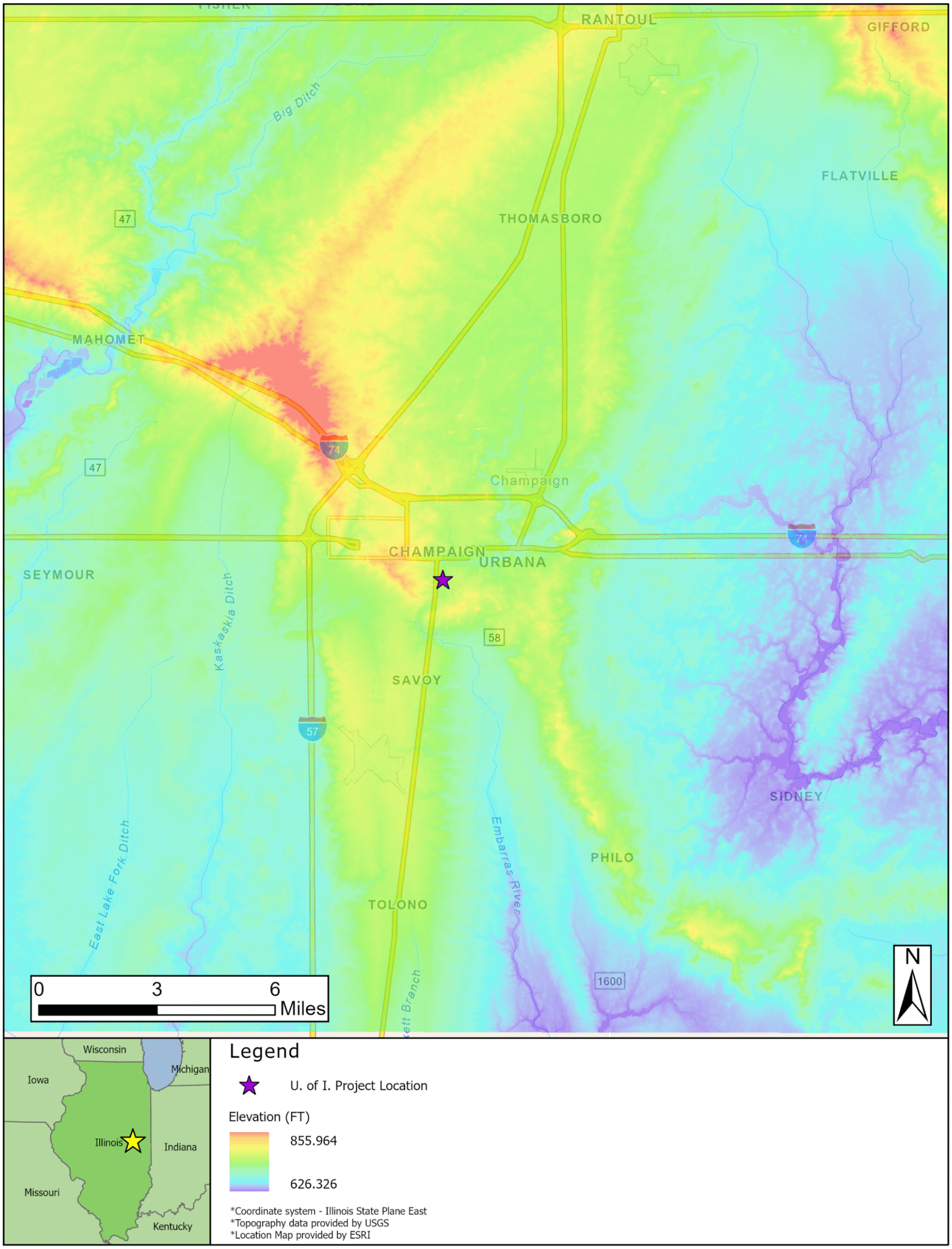
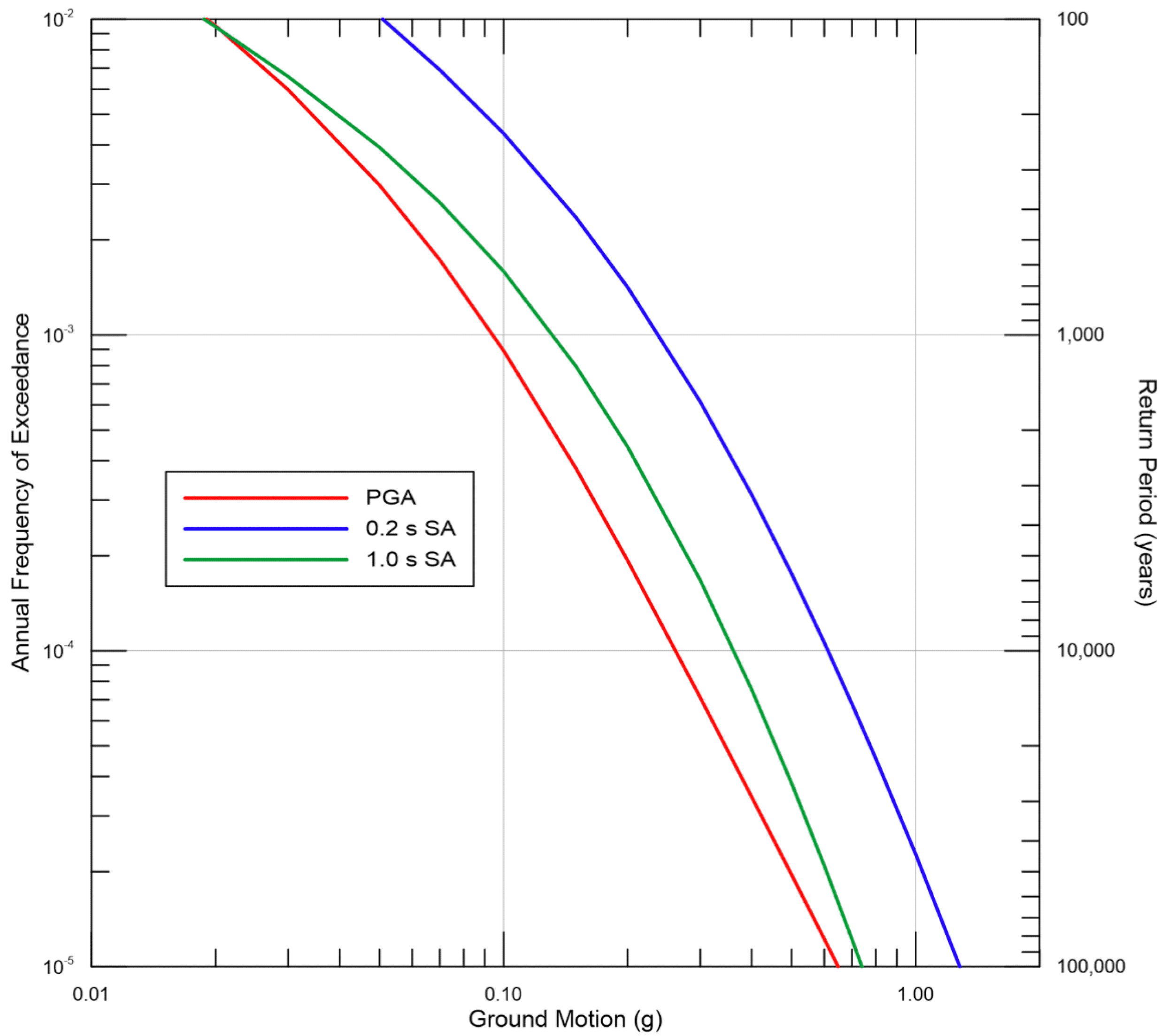
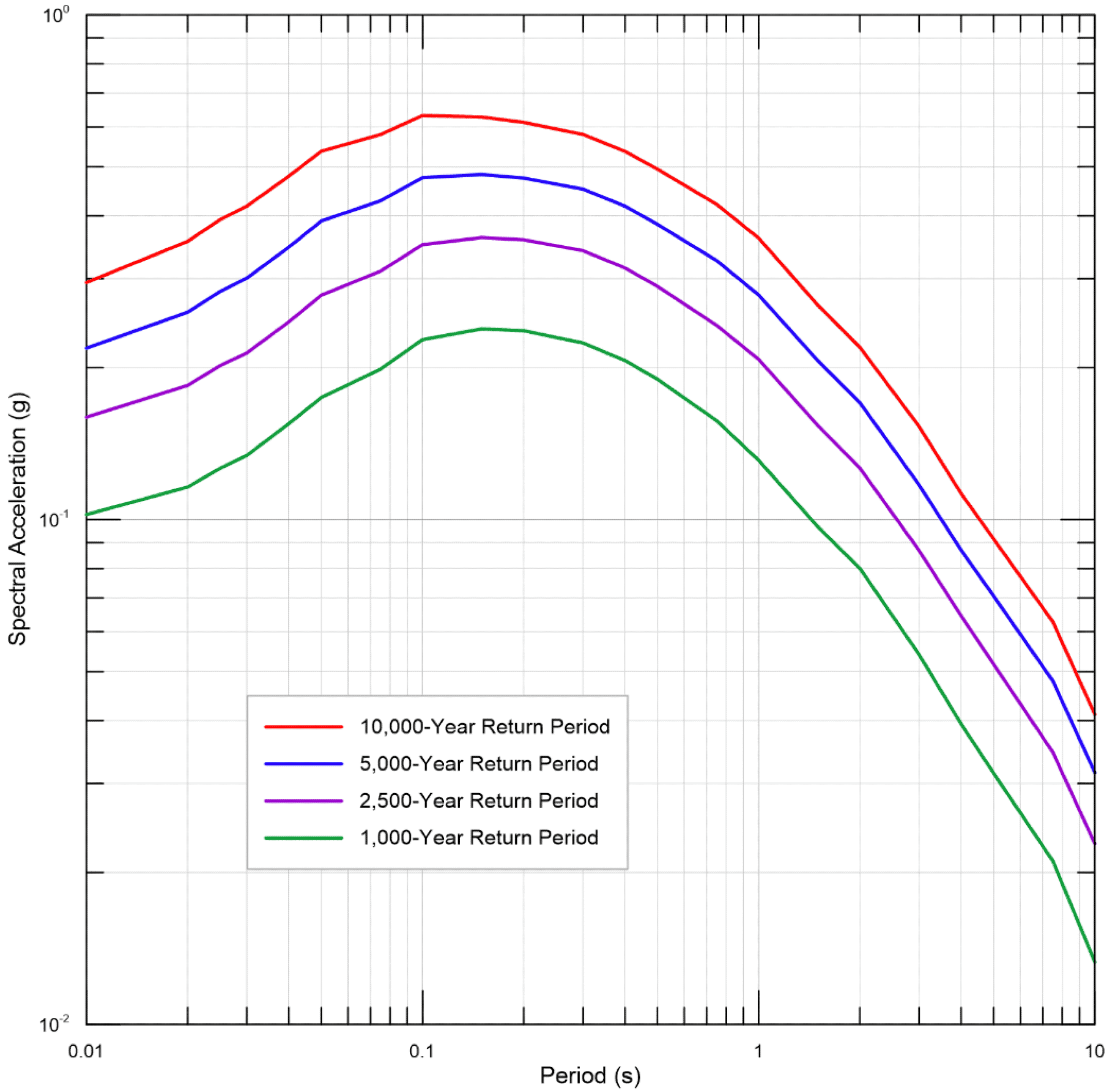


Figure 2-55 Site Specific Hazard Curves



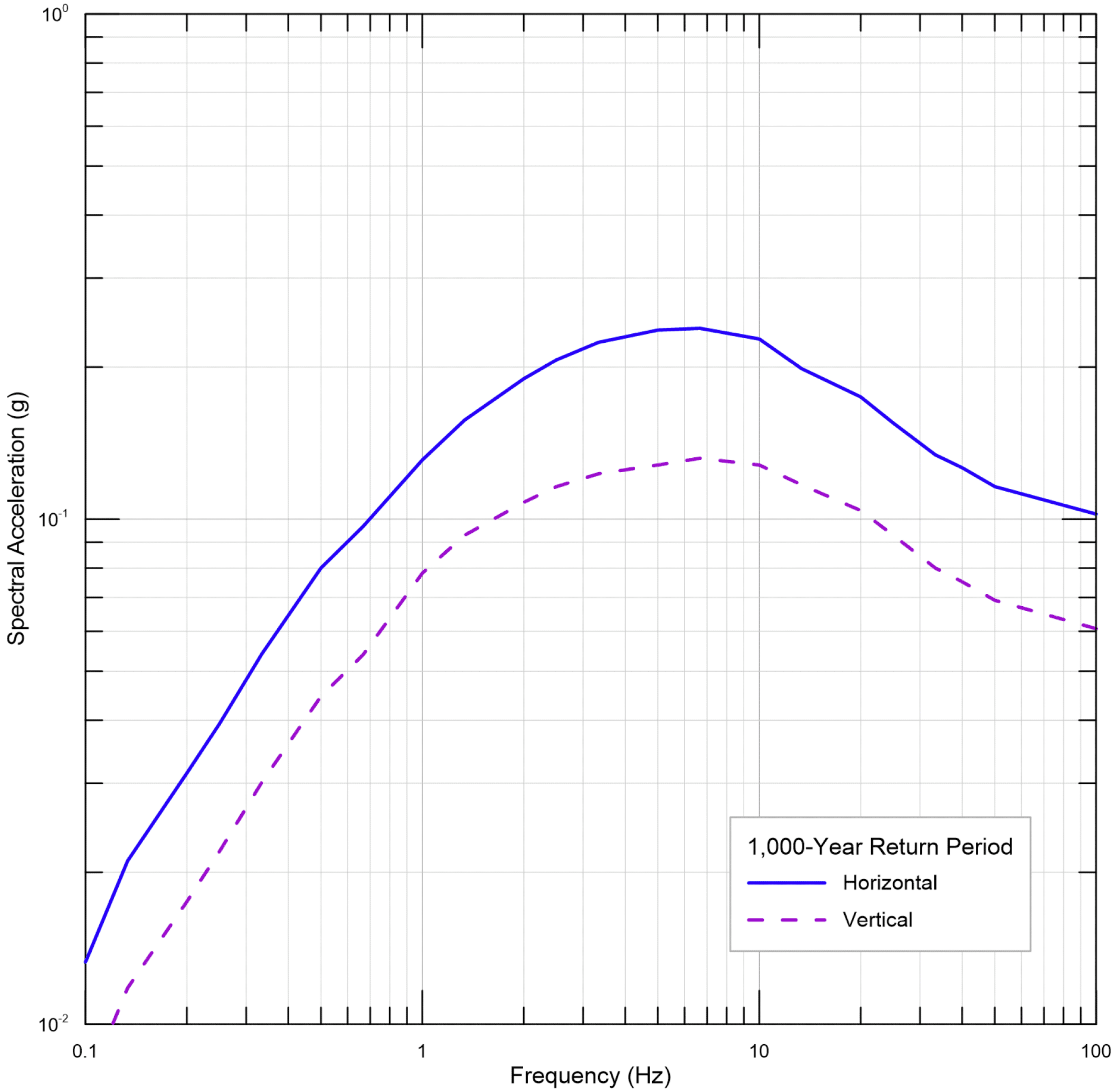
Source: Reference 2-97

Figure 2-56 Site Specific Uniform Hazard Spectra



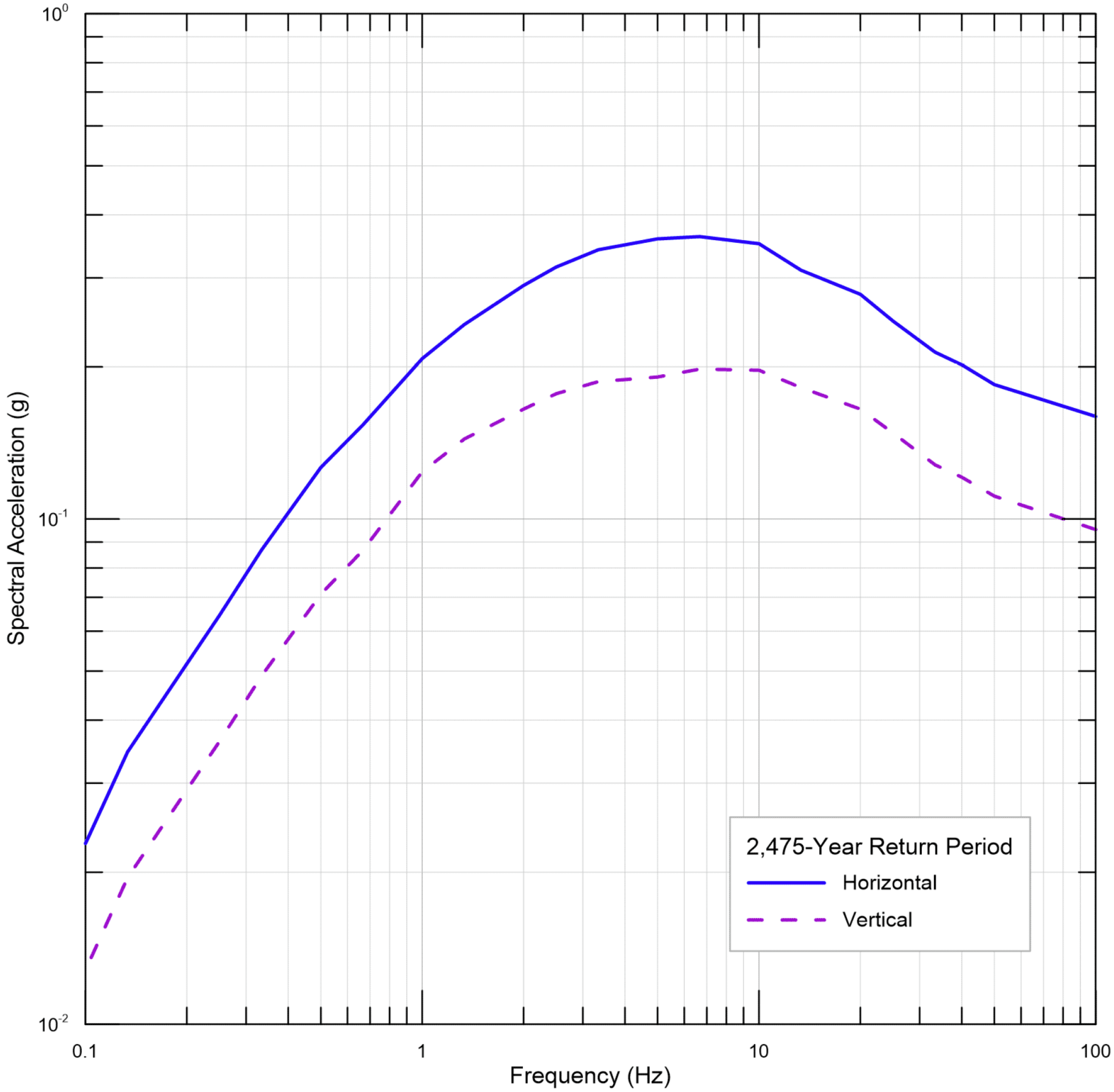
Source: Reference 2-97

Figure 2-57 Site Specific Uniform Hazard Spectra (100-Year Return Period)



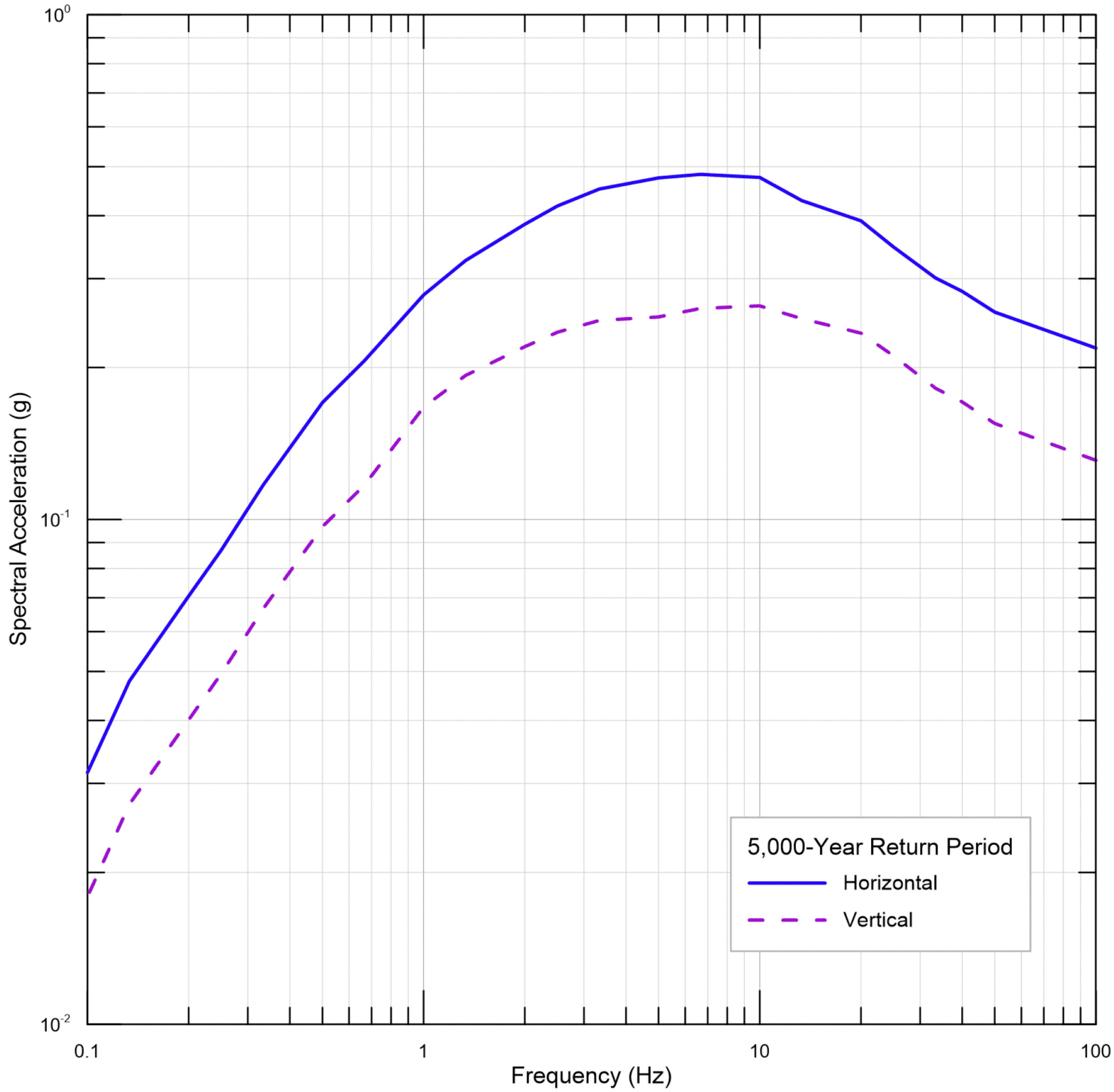
Source: Reference 2-97

Figure 2-58 Site Specific Uniform Hazard Spectra (2,475-Year Return Period)



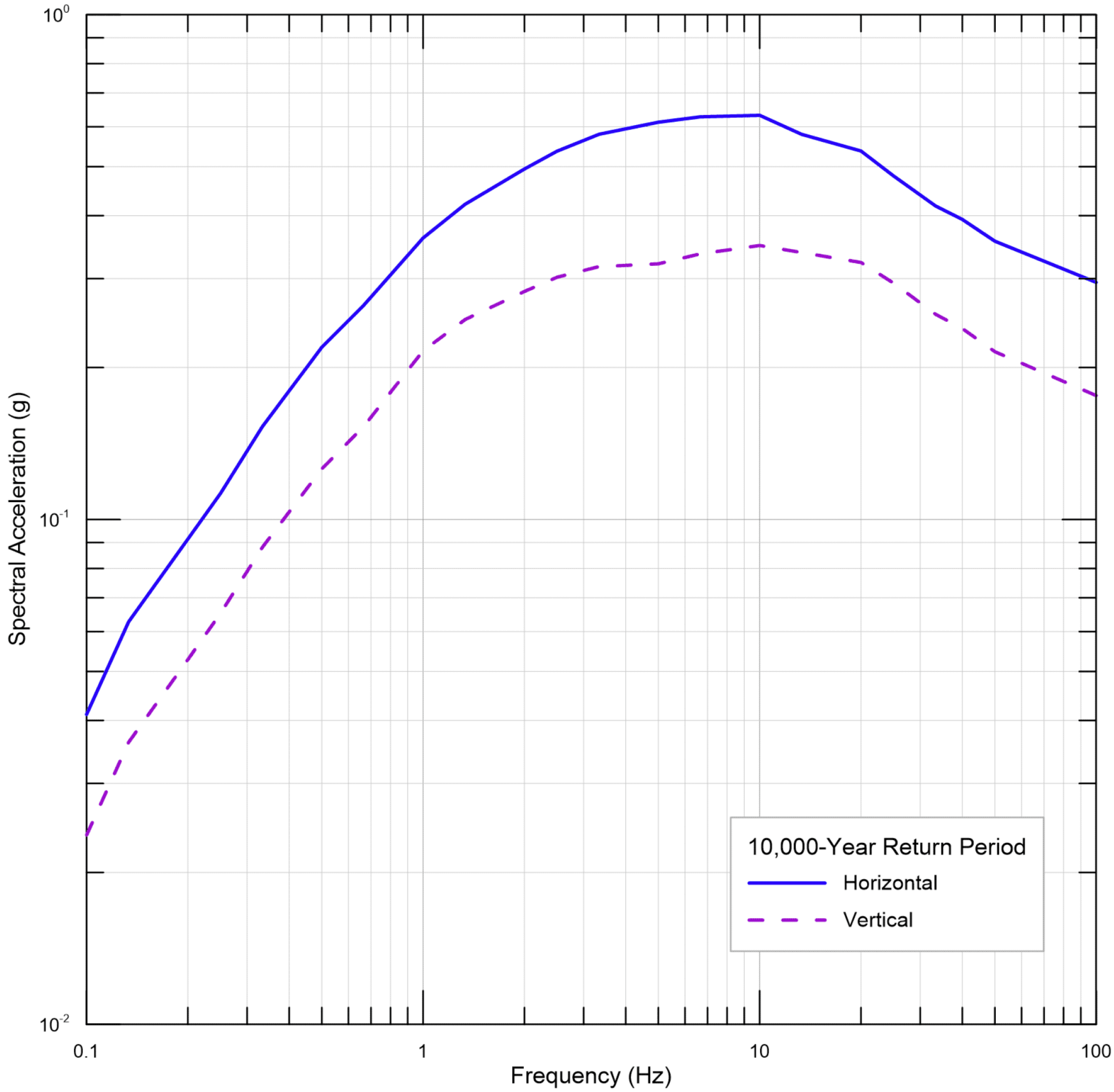
Source: Reference 2-97

Figure 2-59 Site Specific Uniform Hazard Spectra (5,000-Year Return Period)



Source: Reference 2-97

Figure 2-60 Site Specific Uniform Hazard Spectra (10,000-Year Return Period)



Source: Reference 2-97

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PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 3 - DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

Revision 0

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THE UNIVERSITY OF ILLINOIS

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in collaboration with

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to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
AOOs	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
C&C	Components and Cladding
CFR	Code of Federal Regulations
CQC	Complete Quadratic Combination
CRDU	Control Rod Drive Unit
DBE	Design-Basis Earthquake
DRS	Design Response Spectrum
HPS	Helium Purification System
HTGRs	High-Temperature Gas-cooled Reactors
HVAC	Heating, Ventilation, and Air Conditioning
IEEE	Institute of Electrical and Electronics Engineers
IHX	Intermediate Heat Exchanger
I&C	Instrumentation and Control
ISRS	In Structure Response Spectrum
MCER	Maximum Considered Earthquake
MCR	Main Control Room
MHA	Maximum Hypothetical Accident
MHTGR-DCs	Modular High-Temperature Gas-cooled Reactor Design Criteria
MMR	Micro Modular Reactor
MWFRS	Main Wind-Force Resisting System
NFPA	National Fire Protection Association
NPUF	nonpower production or utilization facility
NRC	Nuclear Regulatory Commission
NSR	Non-safety Related
PDC	Principal Design Criteria
PSAR	Preliminary Safety Analysis Report
PSHA	Probabilistic Seismic Hazard Analysis
RCCS	Reactor Cavity Cooling System
RCSS	Reactivity Control and Shutdown Systems
RG	Regulatory Guide
RIS	Reactor Internals System
RPS	Reactor Protection System
SDC	Seismic Design Category
SR	safety related

LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
SSCs	structures systems and components
SSE	Safe Shutdown Earthquake
SSI	Soil Structure Interaction
TR	Topical Report
TRISO	Tri-structural Isotropic
U. of I.	University of Illinois at Urbana-Champaign
USGS	US Geological Survey
VS	Vessel System

CHAPTER 3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 INTRODUCTION

This chapter presents the principal architectural and engineering design criteria applied to the structures, systems, and components (SSCs) that ensure safe operation of the University of Illinois Urbana-Champaign (U. of I.) research reactor and protection of the public. The reactor design relies primarily on inherent safety characteristics, including the use of tri-structural isotropic (TRISO) ceramic annular fuel pellets embedded in a prismatic graphite moderator and helium as the primary coolant. These features provide large passive thermal margins and inherently stable reactor behavior across the full range of design basis conditions. Additional safety related SSCs are designed to ensure that the fuel, reactor core, and reactor coolant boundary remain within acceptable limits during normal operation and postulated events. These SSCs include the safety related portions of the Citadel building, the reactor vessel and internals, the Reactivity Control and Shutdown Systems (RCSS), the Reactor Cavity Cooling System (RCCS), and the Reactor Protection System (RPS).

3.1.1 Design Criteria

U. of I. is pursuing a construction permit for a non-power research reactor under 10 CFR 50. The U.S. Nuclear Regulatory Commission (NRC) regulations in Title 10 to the Code of Federal Regulations (CFR) have been evaluated for applicability to this facility and the results are contained in the U. of I Topical Report (TR), “Applicability of Nuclear Regulatory Commission Regulations” (Reference 3-1), which has been reviewed and accepted by the NRC (Reference 3-2). Presentation and justification of the applicability of design-related regulations to the U. of I. research reactor are provided in the referenced TR.

U. of I. has developed a set of proposed Principal Design Criteria (PDC) for the research reactor, available in the TR “Micro Modular Reactor (MMR) Principal Design Criteria” (Reference 3-3). These PDC have been reviewed and approved by the NRC (Reference 3-4) and apply to the U. of I. research reactor under 10 CFR 50. The discussions in this chapter demonstrate that the reactor design satisfies these PDC. The U. of I. site contains only one reactor unit and no SSCs are shared with any other reactor unit, therefore, PDC-5 is readily satisfied.

The development of PDC is based on the following regulatory and guidance documents:

- 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants” (Reference 3-5) provides the baseline framework for defining PDC. For non-LWRs, the General Design Criteria serve as guidance rather than binding requirements, but their underlying safety objectives remain applicable.
- Regulatory Guide (RG) 1.232, “Guidance for Developing Principal Design Criteria for Advanced Reactors” (Reference 3-6). RG 1.232 provides advanced reactor design criteria, including Modular High-Temperature Gas-cooled Reactor Design Criteria (MHTGR-DCs), that replace direct reliance on the LWR-focused General Design Criteria and can be applied to non-power High-Temperature Gas-cooled Reactors (HTGRs) such as the U. of I. research reactor without the need for exceptions. RG 1.232 instructs applicants to adapt the criteria to their technology while maintaining the underlying safety intent.

- NUREG-1537, Parts 1 and 2, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" (Reference 3-7 and Reference 3-8). NUREG-1537 establishes the licensing framework, including guidance on application content and format for non-power reactors, and informs the treatment of postulated events and SSC classification.

The PDC development methodology is documented in the NRC-approved TR, "Principal Design Criteria" (Reference 3-3). This process included review, and assessment for applicability, of the Advanced Reactor Design Criteria and MHTGR-DCs in RG 1.232, Appendix C (Reference 3-6). Where criteria did not apply, modifications were made to reflect the reactor design features without compromising safety. Departures from the original criteria were documented and are found in Reference 3. To summarize, adaptations were made where RG 1.232 criteria assume conventional LWR features that do not apply to the research reactor design:

- PDC-17, "Electric Power Systems," was modified to reflect the passive safety of the reactor design and the requirement for redundancy already included for the on-site electric power system.
- PDC-19, "Control Room," was modified to reflect the passive cooling nature of the reactor design, which requires/credits no operator actions for maintaining the reactor in a safe condition. The radiation exposure limits were conservatively reduced to align with internationally-recognized conservative dose limits.
- PDC-70, "Reactor Vessel and Reactor System Structural Design Basis," was separated into two distinctive design criteria for these systems, delineating the functions of (a) passive removal of residual heat from the reactor core and (b) retaining structural integrity to allow insertion of control rods.
- PDC-71, "Reactor Building Design Basis," was separated into two distinctive design criteria for the Citadel building, delineating the functions of (a) helium release during overpressure and (b) passive removal of residual heat from the reactor core.

The PDC contained in Reference 3-3 retain terminology from the MHTGR-DCs and LWR design criteria, such as "anticipated operational occurrences (AOOs)," "accident conditions," and "important to safety." This Preliminary Safety Analysis Report (PSAR) is prepared under the non-power reactor licensing framework, in which a maximum hypothetical accident (MHA) is defined, postulated events are not categorized by frequency, and SSCs are classified as either safety related (SR) or non-safety related (NSR), consistent with NUREG-1537. The U. of I. TR, "Event Sequence Identification and SSC Safety Classification Methodology" (Reference 3-9), as reviewed and accepted by the NRC (Reference 3-10), defines the event sequence identification methodology used for this facility and identifies the three fundamental safety functions: control of reactivity, removal of heat from the reactor core, and control of radioactive material release such that public dose limits are not exceeded. The same topical report also provides the methodology for classification of SSCs as either SR or NSR.

Accordingly, for purposes of this PSAR:

- The term "anticipated operational occurrence (AOO)" in the PDC topical report is interpreted as "postulated event."
- The term "accident" or "accident conditions" are interpreted as "postulated event" when used in the design-basis context.

- The term "important to safety" includes all SR SSCs and the subset of NSR SSCs that are required to demonstrate compliance with the PDC in the referenced TR. The definition of "important to safety" is the only adaptation from the TR and does not modify the intent of the PDC but aligns their terminology with the non-power reactor regulatory basis applied to the U. of I. research reactor.

NRC guidance documents are cited in this PSAR where they are used to inform a design topic or evaluation; the discussion below highlights the primary guidance relied upon for the PDC framework.

Regulatory Guides (RGs) in Division 2 and Division 4 generally provide guidance relevant to research and test reactors. Division 1 RGs, which are intended primarily for commercial power reactors, are not directly applicable to non-power reactor facilities. In limited cases, portions of Division 1 RGs contain methods that remain technically relevant. Where such portions were used, they are explicitly identified in the applicable sections of this report.

Applicable codes and standards are identified within the PSAR sections describing each SSC, as available for the current design stage. Additional codes, standards and guidance documents are cited throughout this report as relevant to specific design topics. As the design is finalized, the applicable codes, standards, and regulatory guidance relied upon for SSC design and qualification will be updated and consolidated. This will be reflected in the Operating License Application.

3.2 METEOROLOGICAL DAMAGE

This section describes the approach used to translate design basis meteorological parameters into loads applied to SR structures of the U. of I. research reactor facility. The Citadel building has been identified as the only structure at this facility providing an SR function and only specific portions of the Citadel building perform this SR function. Accordingly, the scope of this evaluation is limited to the SR portions of the Citadel building, as it is the only structure at the facility that houses SR SSCs. Other site buildings do not contain SR SSCs and, therefore, are not required to meet design-basis meteorological load criteria. These parameters are based on the site meteorological characterization presented in Chapter 2.

For the research reactor, the design-basis meteorological loads include normal wind loads, tornado-induced extreme wind loads, hurricane wind and missile loads, and precipitation loads. The Citadel building is located below grade, which inherently limits the exposure of its structural walls to meteorological effects. As a result, only the upper surface of the Citadel building located at ground level (Figure 3-3) is subject to design-basis meteorological loads. This treatment is consistent with ASCE/SEI 7-22 (Reference 3-11) and with PDC-2, which requires protection against natural phenomena commensurate with the site hazards and the safety function of the SSCs.

The determination and application of these design-basis loads are discussed in the following subsections.

3.2.1 Normal Wind Loads

The meteorological characterization of the site (Section 2.3.2) defines the normal and high wind parameters applicable to the facility. This section describes the methodology used to translate the site wind characteristics into design loads applied to the SR portions of the Citadel building. The final structural design under wind loads will be based on the most conservative condition between the site-specific wind characteristics and the applicable code requirements.

The SR SSCs are located within the SR portion of the Citadel building, described in Section 3.5.1. The design of this structure protects these SSCs from the adverse effects of normal and extreme wind events. Wind loads primarily act on the exposed flat roof surface of the Citadel building, corresponding to the upper surface of the structure located at ground level, and are resisted by the structural elements forming the main wind-force resisting system (MWFRS). Components and cladding (C&C) correspond to exterior envelope elements that do not participate in the MWFRS.

3.2.1.1 Applicable Design Parameters

The local building code ASCE/SEI 7-22, "Minimum Design Loads for Buildings and Other Structures" (Reference 3-11), is used as the code of record for wind design. This code specifies risk categories and corresponding design wind speeds. Risk Category IV, the most stringent category, is selected for the SR portions of the Citadel building because these structures support SSCs whose failure could adversely affect public health and safety. Risk Category IV is used to establish conservative design-basis wind speeds and associated loads for structural design.

Based on Figure 26.5-1D of ASCE/SEI 7-22, the basic wind speed for the site is 120 mph (3-second gust) for Risk Category IV. For MWFRS design, this wind speed is converted to pressure using ASCE/SEI 7-22 Section 26.10. For C&C design, wind pressure is determined by using Sections 29.3 and 30.3, respectively. Based on the site characteristics described in Section 2.3.2.3, the site is categorized as Exposure Category C, consistent with ASCE/SEI 7-22.

3.2.1.2 Determination of Applied Forces

In accordance with Equation 26.10-1 of ASCE/SEI 7-22 the velocity pressure is provided in Equation 3-1 below:

$$q_z = 0.00256 \cdot K_z \cdot K_{zt} \cdot K_e \cdot V^2 (\mathrm{lb/ft^2}) \qquad \text{Equation 3-1}$$

Where:

q_z = Velocity pressure at height (z).

K_z = Velocity pressure exposure coefficient at height (z), as determined from Table 26.10-1 of ASCE/SEI 7-22.

K_{zt} = Topographic factor as determined by Section 26.8-2 of ASCE/SEI 7-22 equal to 1.0.

K_e = Ground elevation factor, as determined by Table 26.9.1 of ASCE/SEI 7-22.

V (mph) = basic wind speed (3 second gust) as determined by Figure 26.5-1D of ASCE/SEI 7-22 for Category IV Buildings and Other Structures.

The velocity pressure at the mean roof height q_h is obtained by evaluating Equation 3-1 at z = h, where h corresponds to the mean roof height of the Citadel building above grade. The exposed roof surface corresponds to the exterior {{ }}$^{a(4)}$concrete roof slab located at ground level.

For the Citadel building, where the exposed roof surface is located at ground level, the velocity pressure exposure coefficient K_z is conservatively evaluated at a reference height of 15 ft above grade, consistent with the minimum reference height used in ASCE / SEI 7-22 for low-rise structures.

The velocity pressure q_h is then used to determine the wind pressure acting on the main MWFRS in accordance with ASCE /SEI 7-22.

Because the Citadel building is below grade, the contribution of internal pressure due to wind is considered negligible. Therefore, the net wind pressure applied to the roof surface is determined from the external pressure only, based on Equation 27.3-1 of the ASCE /SEI 7-22 below:

$$p = q_h \cdot K_d \cdot GC_p \qquad \text{Equation 3-2}$$

Where:

P = design wind pressure applied normal to the roof surface (psf)

q_h = velocity pressure evaluated at the mean roof height h

K_d = Wind directionality factor determined in accordance with Section 26.6 of ASCE / SEI 7-22

G = Gust effect factor defined in accordance with Section 26.11 of ASCE / SEI 7-22

C_p = External pressure coefficient for the flat roof surface determined in accordance with ASCE / SEI 7-22

The governing pressure corresponding to the most unfavorable roof zone is conservatively selected for the structural evaluation.

3.2.1.3 Application of Normal Wind Load to Design of Structures

The resulting wind pressure P acts normal to the exposed flat roof surface of the Citadel building, corresponding to the exterior {{ }}[a(4)] concrete roof slab located at ground level, and is applied as a distributed pressure load in the structural model used to evaluate the MWFRS response in accordance with ASCE / SEI 7- 22.

Additional discussion of structural features that address natural phenomena hazards is provided in Section 3.5.3.

3.2.2 Tornado Loading

The meteorological characterization of the site identified the tornado conditions applicable to the facility (Section 2.3.1.7). This section describes the method used to translate the design-basis tornado characteristics into applied loads on the SR portions of the Citadel building. Tornado design parameters include extreme wind speed, atmospheric pressure drop, and tornado-generated missiles. Only the exterior surface of the Citadel building located at ground level is exposed to these loads, as the remainder of the structure is below grade.

3.2.2.1 Applicable Design Parameters

RG 1.76, Revision 1, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants" (Reference 3-12), provides the design-basis tornado characteristics to be applied to the facility. Based on the geographic location of the facility, the parameters associated with **Region I** from Table 1 of RG 1.76 are applicable to the design of the SR portions of the Citadel building.

The design-basis tornado parameters defined for Region I in RG 1.76 include the maximum tornado wind speed, the associated tornado atmospheric pressure drop, and the spectrum of tornado-generated missiles. The maximum tornado wind speed establishes the governing tornado wind field for the facility and is used to determine the tornado wind pressure applied to the SR portions of the Citadel building.

The tornado atmospheric pressure drop defined in RG 1.76 represents the rapid reduction in external atmospheric pressure associated with the passage of the tornado vortex. For the Citadel building, this effect is conservatively applied as a uniform differential pressure acting on the exposed roof slab at grade.

RG 1.76 also defines the spectrum of tornado-generated missiles and the associated maximum horizontal impact velocities that result from the design-basis tornado wind field. The missile types and their corresponding maximum horizontal velocities specified in Table 2 of RG 1.76 are used to evaluate potential missile impact loads on SR portions of the Citadel building.

While RG 1.76 provides the governing tornado wind characteristics and missile parameters, it does not prescribe a specific methodology for determining the corresponding applied wind pressure acting on SSCs. Similarly, NUREG-1537 (Reference 3-7 and Reference 3-8) does not provide detailed guidance for determining tornado-induced wind pressures. Although developed primarily for nuclear power plants, NUREG-0800, Section 3.3.2, "Tornado Loadings" (Reference 3-13) identifies the methodology of ASCE / SEI 7 as an acceptable approach for determining tornado wind pressures and associated structural loads. Therefore, this methodology is adopted.

Accordingly, since ASCE/SEI 7-22 is the code of record for wind design at the site, its wind-pressure methodology is used to compute tornado wind pressures using the wind speeds provided in RG 1.76. The final design under tornado loading is based on the most conservative condition between the site-specific wind characteristics and the applicable code requirements.

3.2.2.2 Determination of Applied Forces

Tornado wind pressures acting on the SR portions of the Citadel building are determined using the methodology described in Section 3.2.1.2 for normal wind load. The velocity pressure associated with the design-basis tornado wind speed is computed in accordance with Equation 26.10-1 of ASCE/SEI 7-22, reproduced as Equation 3-3 in Section 3.2.1.2, with the following modifications for tornado loading:

$$q_z = 0.00256 \cdot K_z \cdot K_{zt} \cdot K_e \cdot V^2 (\mathrm{lb/ft^2}) \qquad \text{Equation 3-3}$$

Where:

V = Maximum tornado wind speed as determined by RG 1.76, Revision 1.

The velocity pressure evaluated at the mean roof height is then used to determine the design tornado wind pressure acting on the structure in accordance with the procedures for the MWFRS defined in ASCE / SEI 7-22. The resulting pressure acts normal to the exposed roof surface of the Citadel building, corresponding to the {{ }}a(4) concrete roof slab located at ground level.

The design-basis atmospheric pressure drop defined in Table 1 of RG 1.76 is applied as a differential pressure load on the exposed roof surface of the Citadel building.

Tornado generated missile loads are determined using the missile spectrum defined in Table 2 of RG 1.76. Missile impact evaluations are performed by converting the kinetic energy of the design-basis missile into an equivalent static load using an energy-balance approach. The equivalent static load is assumed to act locally and normally to the impact surface of the SR portions of the Citadel building. This approach is consistent with the procedures described in NUREG-0800, Section 3.5.3, “Barrier Design Procedures” (Reference 3-13). The resulting equivalent static loads are applied to the exterior structural surfaces at potential missile impact locations to evaluate the effects of tornado-generated missile impacts on the exterior roof slab of the Citadel building located at ground level. The evaluation verifies that the structural barrier maintains its structural integrity and that missile impact does not result in unacceptable perforation, scabbing, or spalling in accordance with the barrier design procedures described in NUREG-0800 Section 3.5.3.

3.2.3 Hurricane Loading

The meteorological characterization of the facility site identified that hurricane wind conditions are not a governing hazard for the site location (Section 2.3.1.8). RG 1.221, Revision 0, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants” (Reference 3-14), provides the methodology for determining design-basis hurricane parameters and associated missile characteristics.

Based on the site meteorological characterization, the facility site is located outside regions subject to design-basis hurricanes as defined in RG 1.221. Therefore, hurricane wind speeds are not a governing design consideration for the facility.

3.2.3.1 Applicable Design Parameters

The guidance from RG 1.221, Revision 0, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” is used on a confirmatory basis to evaluate hurricane wind speeds for the SR portion of the Citadel building. RG 1.221 provides design-basis hurricane wind parameters for the facility that are consistent with the wind-speed definitions used in ASCE / SEI 7-22.

Consistent with ASCE / SEI 7-22 as the code of record for wind loading at the site, the hurricane wind parameters defined in RG 1.221 are reviewed on a confirmatory basis. However, based on the meteorological characterization of the facility site, the site is located outside regions subject to design-basis hurricanes. Therefore, hurricane wind load and hurricane missiles are not governing design conditions for the Citadel building.

3.2.3.2 Determination of Applied Forces

Velocity pressure due to hurricane winds is computed using the ASCE/SEI 7-22 methodology described in Section 3.2.1.2, with the basic wind speed parameter set equal to the ASCE/SEI 7-22 Risk Category IV wind speed applicable to the site.

Hurricane-generated missile impacts are evaluated using the missile guidance in RG 1.221 (Reference 3-14). Consistent with NUREG-0800, Section 3.5.3, Subsection II (Reference 3-13), these missile impact conditions are transformed into equivalent static loads for structural evaluation of the SR portions of the Citadel building.

However, as discussed in Section 3.2.3, the facility site is located outside regions subject to design-basis hurricanes as defined in RG 1.221. Therefore, wind and missile loads resulting from hurricanes are not governing design conditions for the Citadel building.

3.2.4 Precipitation Loads

The meteorological characterization of the site defined the precipitation conditions applicable to the facility (Section 2.3.1.6). Precipitation loads considered in the design include rain, snow, and ice and are evaluated in accordance with ASCE/SEI 7-22. Because the Citadel building is located below grade, precipitation does not accumulate against the SR structural walls. Accordingly, only the exterior surface corresponding to the flat portion of the concrete roof slab of the Citadel building located at ground level is subject to the design-basis precipitation loads.

Site grading and drainage features prevent the accumulation of precipitation on ground surfaces adjacent to the SR portions of the Citadel building. As a result, rain loads are not governing the design. Ice loads are expected to be minimal due to the absence of rainwater accumulation and are bounded by the governing snow load cases described in the following subsections.

3.2.4.1 Applicable Design Parameters

Based on the Risk Category IV classification of the SR portions of the Citadel building (Section 3.2.1.1) and the site climatic conditions defined in Chapter 2 the snow load design parameters are determined in accordance with Chapters 1 and 7 of ASCE/SEI 7-22.

3.2.4.2 Determination of Applied Forces

Snow load acting on the Citadel building are determined in accordance with chapter 7 of ASCE / SEI 7-22. Because the exposed surface of the Citadel building corresponds to a flat {{ }}[a(4)] concrete roof slab located at ground level, the flat roof snow load is used as the basis for design. In accordance with Section 7.3 of ASCE / SEI 7 -22, Equation 7.3-1, the flat roof snow load p_f is determined as:

$$p_f = 0.7 \cdot C_e C_t p_g \qquad \text{Equation 3-4}$$

Where:

p_f = Flat roof snow load (psf)

C_e = Exposure factor as determined by Table 7.3-1 of ASCE/SEI 7-22.

C_t = Thermal factor as determined by Table 7.3-2 and 7.3-3 of ASCE/SEI 7-22.

p_g = ground snow load for the facility site as defined in Section 2.3.1.11.

The resulting flat roof snow load p_f represents a uniformly distributed vertical load acting downward on the exterior horizontal surface of the {{ }}[a(4)] concrete roof slab forming the upper surface of the Citadel building at grade.

In addition, other snow loading conditions required by chapter 7 of ASCE / SEI 7-22 are evaluated where applicable. These include snow drift loads that may develop due to the accumulation of wind transported snow against roof projections and unbalanced snow load conditions resulting from wind redistribution on the roof surface. Where applicable, these loads are determined in accordance with ASCE / SEI 7-22 and applied as additional vertical surface load over the affected roof areas.

3.3 WATER DAMAGE

This section describes the evaluation of water-related loads on the SR portions of the Citadel building resulting from internal and external flooding postulated events. Internal flooding events consider credible water sources within the facility, while external flooding events consider hydrologic conditions at the site as characterized in Section 2.4.

3.3.1 Internal Flooding

Internal flooding postulated events consider credible sources of water within the SR portions of the Citadel building, including piping failures, equipment leaks, and potential discharge from water-containing systems. The evaluation considers the maximum flow rates and volumes that could be released from these sources.

The Citadel building configuration provides passive protection of SR SSCs from internal flooding, including available free volume within the Citadel building compartments and placement of SR SSCs above floor level where applicable, such that postulated internal flooding does not compromise the operability or integrity of SR SSCs. {{

}}[a(4)] The design ensures that a postulated internal flooding event will not compromise the operability or integrity of SR SSCs.

3.3.2 External Flooding Events

The hydrologic evaluation described in Section 2.4 demonstrates that the site is well drained and that the design-basis flood elevation does not exceed finished grade. Therefore, external flooding does not impose loads on exterior surface of the Citadel building.

Consistent with this site characterization and the structural design approach, the effects of external flooding on the below-grade SR structure are evaluated by considering conservative hydrostatic pressure corresponding to the design-basis condition as design loads.

These loads are included in the governing structural load combinations, and the below-grade Citadel structure is demonstrated to maintain structural integrity and continue to support and protect SR SSCs.

{{

}}[a(4)]

3.4 SEISMIC DAMAGE

This section describes the seismic design bases and design methodology applied to SSCs that must maintain their safety function during a design-basis earthquake (DBE). The seismic design ensures that a DBE will not prevent the reactor from shutting down or being maintained in safe shutdown conditions.

Section 2.5 defines the Safe Shutdown Earthquake (SSE) based on a site-specific seismic hazard analysis. For the SR Citadel building, this SSE is conservatively adopted as the DBE for the seismic design of SSCs.

Seismic classification of SSCs is performed in accordance with RG 1.29 (Reference 3-15) which distinguishes between Seismic Category I and non-Seismic Category I SSCs. SSCs classified as SR are designated as Seismic Category I while NSR SSCs are classified as non-Seismic Category I.

A graded, performance-based seismic design approach consistent with ASCE/SEI 43-19, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities" (Reference 3-16) is applied to Seismic Category I SR SSCs. ASCE/SEI 43-19 establishes performance goals, limit states, and Design Response Spectrum (DRS) based on SSC safety function and potential radiological consequences. For SR SSCs credited for safe shutdown or for limiting radiological releases, the seismic design is performed to Seismic Design Category (SDC-) 3, which represents the applicable performance level defined by ASCE/SEI 43-19.

The DBE ground motion is defined based on the site-specific probabilistic seismic hazard analysis (PSHA) described in Section 2.5.5, with the resulting Safe Shutdown Earthquake (SSE) ground motions adopted as the seismic design input. Site characterization data, including shear wave velocity, profiles and seismic site class, are developed based on the geotechnical and geophysical investigation performed at the site summarized in Section 2.5. These inputs are used to develop SDC-3 DRS in accordance with ASCE/SEI 43-19.

NSR SSCs are designed in accordance with the local building code requirements of the 2024 IBC, and ASCE/SEI 7-22, consistent with the guidance provided in NUREG-1537 for research reactor facilities (Reference 3-7 and Reference 3-8).

3.4.1 Seismic Design for Safety Related SSCs

3.4.1.1 Seismic Design Criteria

SR SSCs are designed to withstand the DBE while maintaining the safety functions necessary to shut down the reactor and maintain it in a safe shutdown condition. The radiological consequences associated with DBE are shown in Chapter 13 to remain within the limits of 10 CFR 50 (Reference 3-5). The seismic design approach provides reasonable assurance that SR SSCs will continue to perform their credited safety functions during and after DBE.

SR SSCs are classified as Seismic Category I in accordance with RG 1.29. A graded, performance-based seismic design methodology is applied consistently with ASCE/SEI 43-19. This standard defines performance goals, limit states, and DRS based on safety classification and hazard level. SR SSCs credited for safe shutdown or for preventing radiological releases are assigned to SDC-3 in accordance with ASCE/SEI 43-19.

For SDC-3 SSCs, the DBE ground motion is defined using the site-specific seismic probabilistic seismic hazard analysis (PSHA) results presented in Section 2.5. The resulting ground motion parameters are used to develop SDC-3 DRS in accordance with the methodology of ASCE/SEI 43-19.

This approach is intended to achieve the performance goals specified in ASCE/SEI 43-19 which are expected in terms of acceptable probabilities of unacceptable performance at increasing levels of seismic demand below:

- A target annual performance goal of 1 x 10^{-4} for SDC-3 SSC S, and
- Demonstrating of seismic margin at ground motion levels exceeding the DBE (1.5 times the DBE ground motion).

The objectives are consistent with the target annual performance goal defined in Table 1-1 of ASCE/SEI 43-19.

Acceptable seismic performance criteria are based on the safety function of each SSC and limit states are selected in accordance with ASCE/SEI 43-19.

3.4.1.2 Design Response Motion

The design response motion for SR SSCs is developed in accordance with the seismic hazard characterization approach described in Section 2.5. The PSHA results provide the ground motion parameters that define the seismic input used for SSCs design.

These ground motion parameters are used to develop SDC-3 DRS consistent with ASCE/SEI 43-19, corresponding to a target annual performance goal H_p defined for SDC-3 SSCs, where:

$$H_p = 1 \times 10^{-4}$$

3.4.1.3 Design Response Spectra

Using the seismic input defined in accordance with the seismic hazard characterization approach described in Section 3.4, the 5% damped horizontal and vertical DRS for SDC-3 are developed following Section 2.2 of ASCE/SEI 43-19. These spectra are used to design SSCs to achieve the target seismic performance goals outlined in Section 3.4.1.1. The horizontal and vertical DRS are illustrated in Section 2.5.5.

3.4.1.4 Seismic Response

The seismic response of the Citadel building is evaluated using the SDC-3 DRS developed in Section 3.4.1.1 and Section 3.4.1.3. The structural models are subjected to a three-component seismic input, consisting of two orthogonal horizontal components and one vertical component, to determine the seismic demands for design of SDC-3 SSCs.

Seismic analysis is performed using deterministic linear elastic methods in accordance with ASCE/SEI 4-16, Chapter 4 (Reference 3-18). The analysis uses the 5%-damped horizontal and vertical DRS as the basis for input ground motions.

Soil–structure interaction (SSI) effects are incorporated following the guidance of ASCE/SEI 4-16, Chapter 5. Subsurface material properties used to develop the SSI model are consistent with the site characterization presented in Section 2.5. SSI analyses evaluate the interaction between the Citadel structure and the supporting soils and are used to refine the input motions transmitted to the structure.

The seismic response methodology described in Section 3.4.1.1 through Section 3.4.1.3 defines the process from site seismic hazard characterizations to the determination of structural demands and In-Structure Response Spectra (ISRS) used for the design and qualification of SSCs.

3.4.1.5 Structural Model

The Citadel building is represented using a three-dimensional finite element model developed in accordance with ASCE/SEI 4-16, Chapter 3. The model includes the primary elements of the lateral load–resisting system as well as secondary structural components that may significantly influence the dynamic response. Structural elements are represented using appropriate finite element types consistent with their geometry and structural function. Rigid links and other connectivity requirements are incorporated where necessary to accurately represent load transfer and global dynamic behavior.

The Citadel building is located below grade, with only the exterior roof slab exposed at ground level. The structural model captures the interaction between the embedded structure, surrounding soil, and the foundation system. Boundary conditions at the base and along the embedded portions of the structure are defined to represent the restraint provided by the surrounding soil consistent with the geotechnical characterization presented in Section 2.5. Soil-structure interaction (SSI) effects are represented through appropriate boundary conditions and soil restraint assumptions for static load cases and are explicitly incorporated in seismic analysis in accordance with the methodology of ASCE / SEI 4-16.

Structural mass is assigned to the model to capture the self-weight of structural elements, permanently installed equipment, and other relevant mass contributions associated with design loads. Major equipment masses are represented in the model using lumped masses connected to the supporting structural elements where appropriate. The assigned structural masses are consistent with the guidance of ASCE / SEI 4-16 for seismic dynamic analysis. Modal analysis is performed to determine the natural frequencies, mode shapes, and modal mass participation factors required for the seismic response evaluation.

A concrete cracking evaluation methodology is applied using the SDC-3 DRS to assess the potential for cracking of structural elements at the design level. Where cracking is indicated, element stiffness properties are adjusted in accordance with the guidance of Table 3-2 of ASCE/SEI 4-16. Structural damping is assigned based on the expected level of cracking and response amplitude, using the values provided in Table 3-1 of ASCE/SEI 4-16, consistent with the use of 5% damped response spectra.

Structural demands obtained from the seismic analysis are combined with other applicable loads using load combinations consistent with ACI 349-13 (Reference 3-19) requirements for nuclear-related concrete structures. The governing load combinations used for structural design are summarized in Table 3-1.

The resulting finite element model is used to evaluate structural behavior including structural forces, deformations, overall structural stability, modal characteristics, and ISRS needed for the design and qualification of SDC-3 SSCs. The modeling approach is consistent with the requirements and guidance of ASCE/SEI 4-16 for seismic analysis of SR nuclear structures.

3.4.1.6 Response Analysis

The analysis is conducted using modal response spectrum analysis. Modal responses are combined using the Complete Quadratic Combination (CQC) method for closely spaced modes, consistent with ASCE/SEI 4-16. Sufficient modes are included to ensure that at least 90% of the mass participation is captured in each principal direction of response.

SSI effects are evaluated following the guidance in ASCE/SEI 4-16, Chapter 5. The SSI model incorporates site-specific soil properties from Section 2.5.

Additional details regarding SSI modeling methods, assumptions, and analysis results will be provided in the Operating License Application. The Operating License Application will also summarize the seismic response analysis results, including structural forces and ISRS.

3.4.1.7 Seismic Qualification

Seismic qualification of SR SSCs ensures that these SSCs maintain their credited safety functions during and after the DBE. The qualification process is consistent with the performance-based requirements of ASCE/SEI 43-19 for SDC-3 and uses the seismic demands and ISRS developed in Section 3.4.1.4 and Section 3.4.1.6.

3.4.2 Non-Safety Related SSCs and Seismic Design

With respect to seismic design, NSR SSCs are designed in accordance with the applicable local building code, the 2024 IBC (Reference 3-17). The design basis ground motion for NSR SSCs is defined using the deterministic procedures of the IBC, which reference to ASCE/SEI 7-22 (Reference 3-11).

Site-specific ground motion parameters, including site class and site coefficients, are determined in accordance with Chapter 21 of ASCE/SEI 7-22. These parameters are used to define the risk-targeted maximum considered earthquake (MCER) ground motions for NSR SSCs.

Seismic analysis and qualification of NSR SSCs are performed in accordance with the IBC (Reference 3-17). Seismic design requirements for NSR structures follow Chapter 12 of ASCE/SEI 7-22, while seismic design requirements for NSR nonstructural systems and components follow Chapter 13 of ASCE/SEI 7-22. Any Exceptions to ASCE/SEI 7-22 applicable to NSR SSCs will be identified in the Operating License Application.

3.4.3 Seismic Instrumentation

Seismic instrumentation that enables the prompt processing of the data at the site is installed for monitoring.

The purpose of the instrumentation is to permit a comparison of measured responses of the site with estimated responses corresponding to the design basis ground motion, and permit facility operators to understand the possible extent of degraded performance within the facility immediately following a seismic event. Instrumentation is also used to determine when a design-basis earthquake event has occurred that warrants inspection and maintenance activities.

3.4.3.1 Location and Description of Seismic Instrumentation

The seismic instrumentation consists of tri-axial time-history accelerometers {{ }}[a(4)] The free-field instrument is mounted on rock or competent ground representative of the dynamic characteristics of the site. The instrumentation records time-history data at time increments adequate to capture the range of vibration frequencies contained in the design-basis seismic spectra.

The seismic instrumentation is designed to continue recording in the event of a loss of electrical power. All instruments are housed in weather-resistant and animal-proof enclosures.

3.4.3.2 Seismic Instrumentation Operability and Characteristics

Seismic instrumentation operates during all modes of facility operation. Plant procedures provide for keeping a minimum required number of seismic instruments in service during facility operation. The seismic instrumentation design includes provisions for in-service testing. The seismic instruments are capable of periodic channel checks during normal facility operation and in-place functional testing.

3.5 PLANT STRUCTURES

3.5.1 Description of Plant Structures

Figure 3-1 shows the location and orientation of the licensed area. The plot is approximately 240 ft x 280 ft at its widest. The facility consists of the sub-grade Citadel building, services building located above the Citadel building, the operations building connected to the service building, the molten salt storage tanks, and a steam supply building, which interface with the existing Abbott Power Plant at U. of I.

As shown in Figure 3-2, the services building includes a maintenance enclosure, which encloses the Citadel building and associated fuel handling and storage systems, and provides a suitable environment for refueling operations. Except from the Citadel building and its contents, the services building does not contain any SR SSCs and is designed so that its failure does not interfere with safety functions of SSCs located in the Citadel building.

The operations building is made of several modules that connect together to form the building. {{ }}$^{a(4)}$ The operations building does not contain any SR SSCs and does not interface with any SR SSCs.

The Main Control Room (MCR), {{ }}$^{a(4)}$ serves as the centralized facility for operators to control and monitor the nuclear plant during normal operation and to support monitoring following postulated events. The MCR is not relied upon as a primary radiological shielding structure; as discussed in Section 4.4, radiological protection is provided by the Citadel building and other facility shielding features. The MCR does not contain SR SSCs and does not perform a safety function. Operator actions and automated functions associated with plant control and monitoring are described in Chapter 7.

The Citadel building is the primary structure housing the reactor and associated SR SSCs. The Citadel building is a below-grade, {{ }}$^{a(4)}$ concrete facility designed to provide structural support and radiation shielding for SR SSCs. The Citadel building footprint is approximately 80 ft x 50 ft. The Citadel building protects the SR SSCs from the effects of natural phenomena and external event hazards discussed in Section 3.2, Section 3.2.2, and Section 3.2.3. Figure 3-3 shows the planned layout of the Citadel building.

Figure 3-1 Licensed Area Orientation and Location

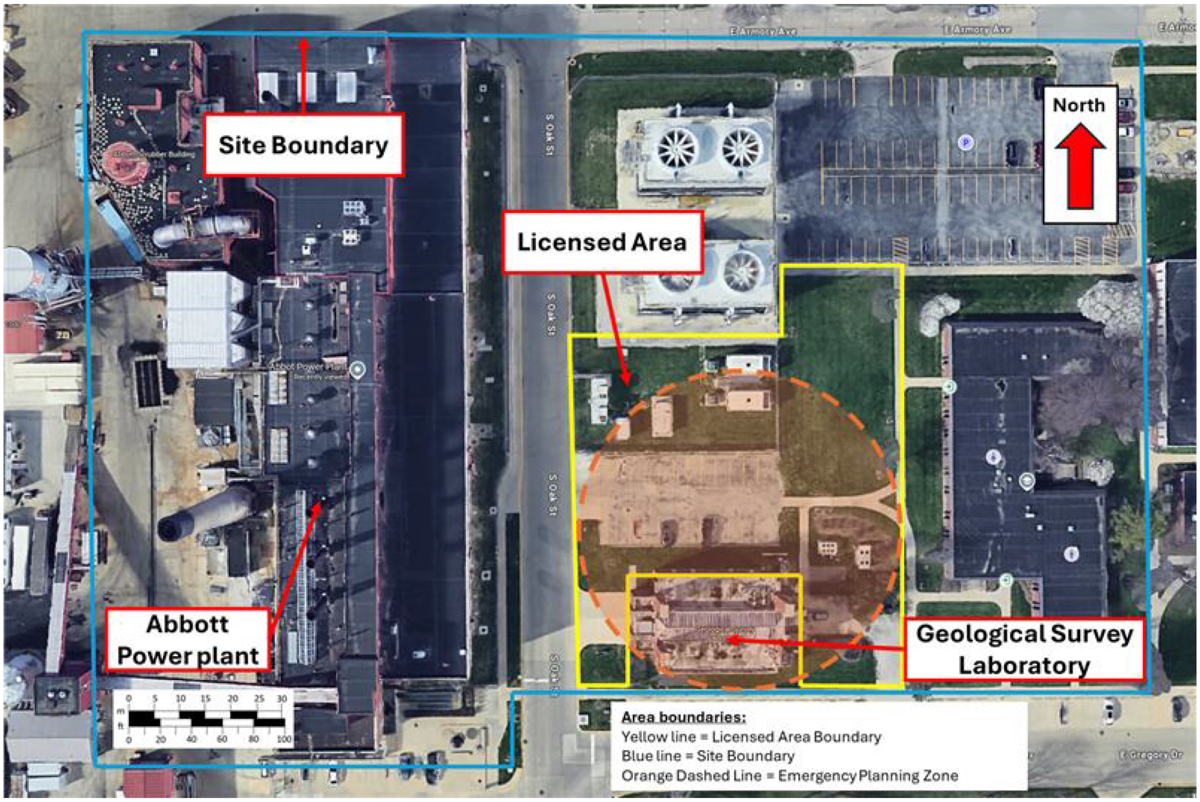


Figure 3-2 Conceptual View of Services Building

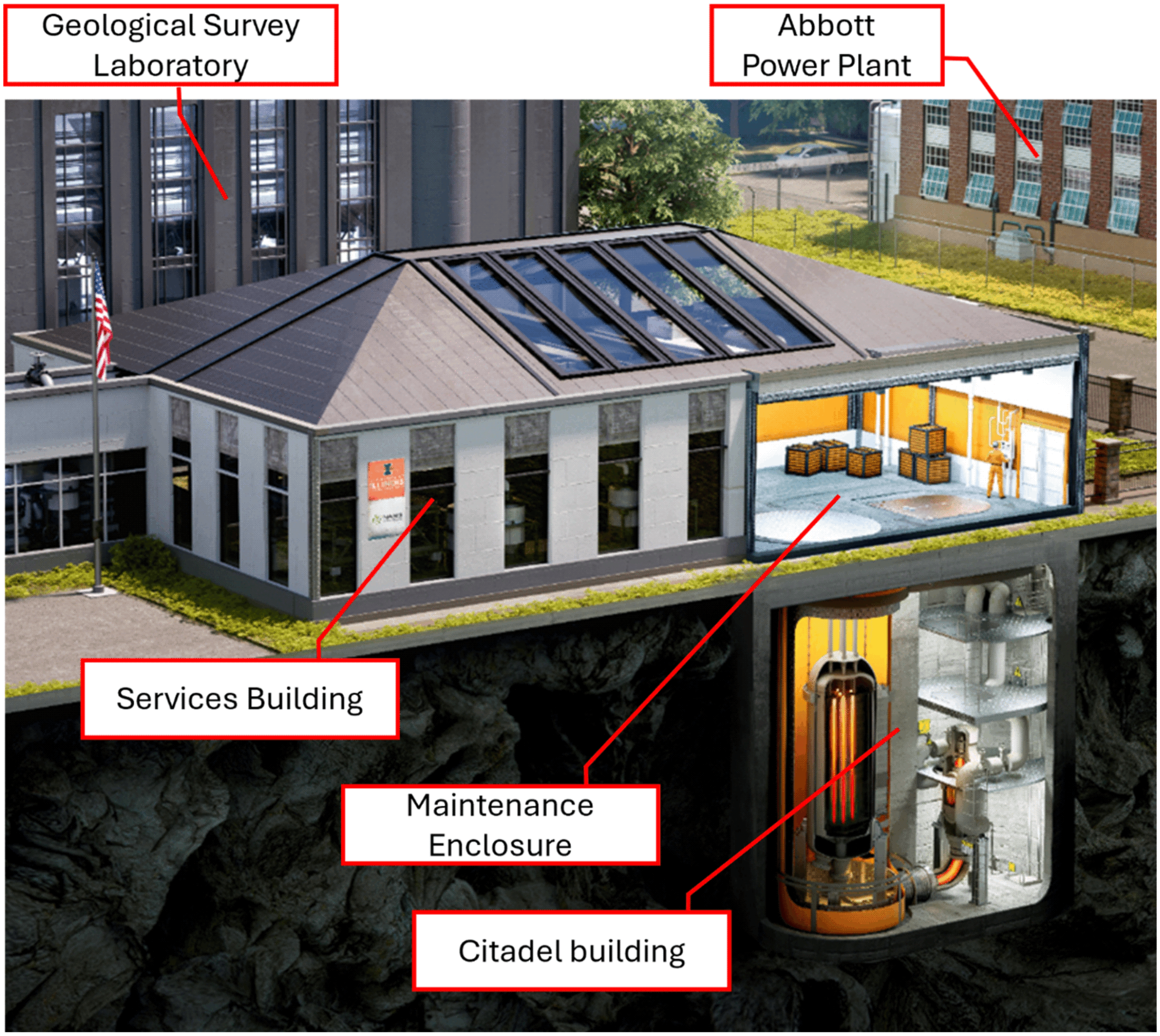
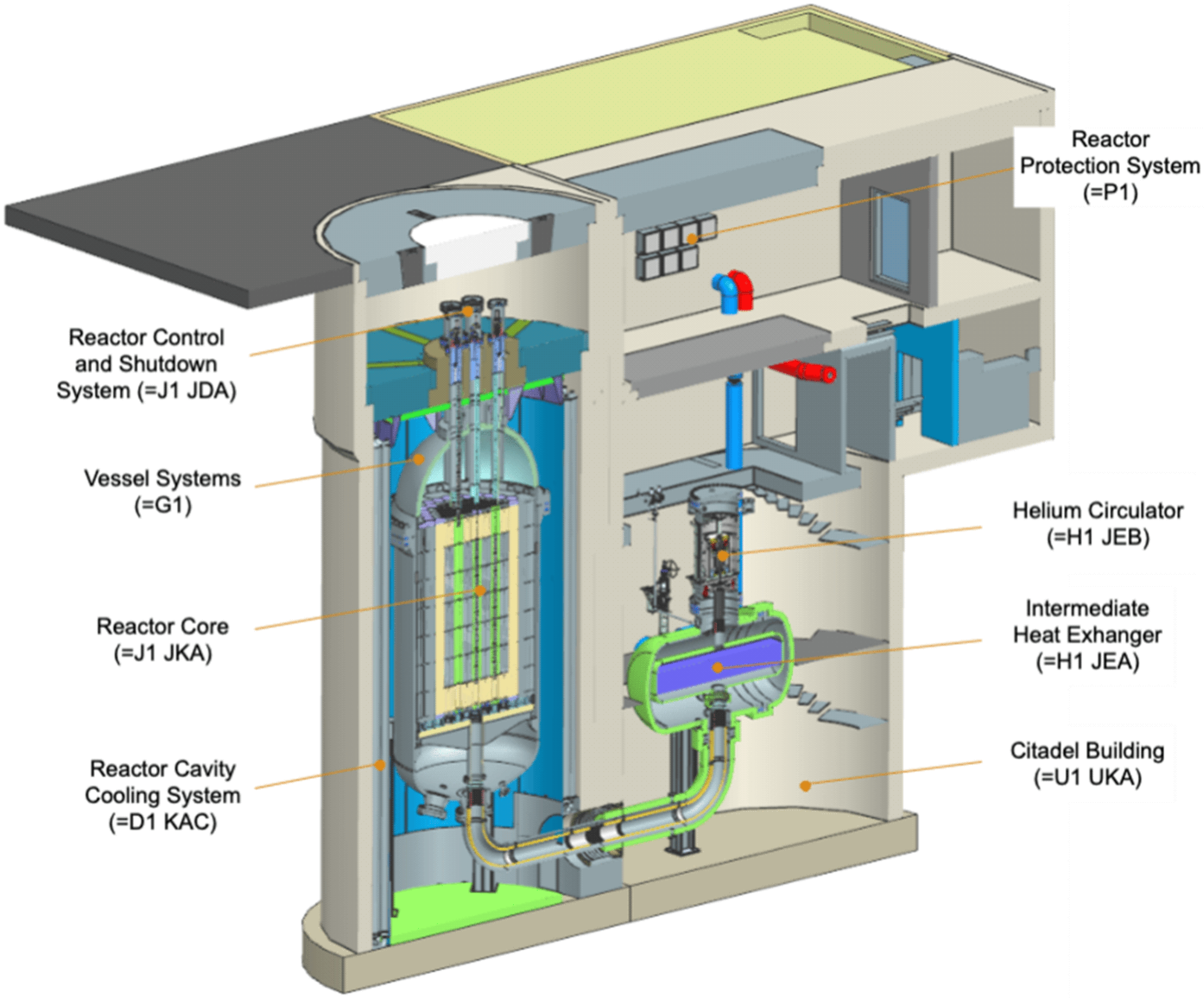


Figure 3-3 Preliminary Layout of Citadel Building (Below Grade)



The Citadel building is classified as a SR structure and contains all the SR SSCs in the facility and selected NSR SSCs. Safety classification of the SSCs is discussed in Section 3.6. The below-grade configuration enhances structural stability, reduces seismic load, and provides inherent protection against tornado missiles. Concrete walls and slabs also contribute to radiation shielding. The rooms of the Citadel building and their associated SR components are as follows:

- Reactor Cavity and Control Rod Drive Unit (CRDU) cavity, containing the Reactor Core, Reactor Internals System, Reactor Vessel and associated supports, Reactivity Control and Shutdown System (RCSS), Reactor Cavity Cooling System (RCCS), and associated SR instrumentation and control. The Reactor Cavity also contains the SR neutron flux detectors of the RPS.
- Intermediate Heat Exchanger (IHX) Cavity, which contains the primary coolant pressure and temperature sensors associated with the RPS.
- Reactor Equipment Room, containing the RPS hardware and associated instrumentation cabinets,

- Battery Rooms, Helium Services Room, {{ }}a(4) and Stair/Access Rooms, which do not contain any SR SSCs.

Radiation zoning and personnel accessibility considerations are discussed in Section 4.4. The safety functions of the Citadel building are:

- Support and protect the SR SSCs installed in the building.

Other buildings on the site do not contain any SR SSCs and, therefore, serve no SR function. This includes the steam supply building.

3.5.2 Design Bases/Principal Design Criteria

The following PDC were used to develop the design bases for the SR Citadel building, consistent with the U. of I. Topical Report (TR), "Micro Modular Reactor (MMR) Principal Design Criteria" (Reference 3-3), as reviewed and approved by the NRC (Reference 3-4):

- **PDC-1,** "Quality standards and records"
- **PDC-2,** "Protection against natural phenomena"
- **PDC-3,** "Fire protection"
- **PDC-4,** "Environmental and dynamic effects"
- **PDC-70a,** "Citadel design basis"
- **PDC-71a,** "Citadel design basis- Residual Heat Removal"
- **PDC-71b,** "Citadel design basis- Release during Depressurization Accident"
- **PDC-72,** "Provisions for periodic Citadel inspection"

3.5.3 System Evaluation

3.5.3.1 Services Building

The services building is classified as NSR because it does not contain SR SSCs and is not relied upon to perform any safety function. The SR SSCs are housed within the Citadel building, which, as detailed in the following section, is designed to support and protect the SR SSCs. Consistent with the scope of Chapter 3, the Citadel building is the only structure performing a safety function, and other site buildings, none of which contain SR SSCs, are not required to meet the load criteria for SR functions. Although the services building is above the Citadel building, it is not credited to provide physical protection to SR SSCs against normal or high winds (see Section 3.5.3.1), the effects of design basis seismic events (see Section 3.5.3.3), or water damage (see Section 3.5.3.2). The Citadel building is designed to be able to perform its physical protection safety functions even if the services building is damaged. Accordingly, the services building is designed such that its failure does not adversely affect the safety functions of the SSCs location within the Citadel building.

The NSR classification and that failure of the services building does not adversely affect safety functions will be demonstrated {{

}}a(4) in the Operating License Application as part of the final design and hazard evaluation for plant structures.

3.5.3.2 Citadel Building

A list of load combinations for the Citadel building is provided in Table 3-1. Note that the design features that mitigate the consequences of potential hazards such as flooding will be further evaluated in the Operating License Application.

Consistent with PDC-1, the Citadel building is designed in accordance with nuclear-grade codes and standards, including ACI 349-13 (Reference 3-19) for {{ }}a(4) concrete and ANSI/AISC N690-18 (Reference 3-20) for steel structures. These standards ensure structural integrity under all design basis conditions. Compliance is maintained through the U. of I. Quality Assurance Program described in Section 12.9, which governs design, fabrication and inspection activities.

Consistent with PDC-2, the Citadel building is designed so that it will be able to perform its physical protection safety functions described in Section 3.5.1, even if the services building is damaged due to the design basis wind, water or seismic events described in Section 3.2, Section 3.2.2, and Section 3.2.3. The system evaluation for PDC-2 is provided in the following subsections.

Consistent with PDC-3, the Citadel building design provides fire protection features such that fires or explosions will not prevent SR SSCs from performing their SR functions, including cases where redundant SSCs perform the same safety function. Noncombustible and fire-resistant materials are used wherever practical, particularly in locations containing SR SSCs. Where fire-rated structural barriers are used to separate fire areas, penetrations through these barriers are protected to maintain the fire-resistive integrity of the barrier and limit fire spread. All fire protection features are designed consistent with NFPA 801 (Reference 3-22). Where redundancy is credited for a safety function, the Citadel building layout and fire protection features provide adequate separation such that a single fire does not disable redundant SSCs credited to perform the same safety function. Fire detection and fire-fighting systems of appropriate capacity and capability are provided to minimize the adverse effects of fires on SR SSCs (see Section 9.3).

Consistent with PDC-4, the Citadel building accommodates environmental conditions associated with normal operation, maintenance, testing, and postulated events. Materials and structural components are qualified for temperature, humidity, and radiation conditions expected during normal operation and postulated events, and the structural design considers dynamic effects such as pipe whip, internal and external missiles, and discharging fluids.

{{

}}a(4)

Consistent with PDC-70a, the Citadel building design supports the reactor vessel and reactor system by maintaining structural stability during normal operation and postulated events. This ensures the geometry required for passive removal of residual heat from the reactor core to the ultimate heat sink is preserved. {{

}}a(4)

Consistent with PDC-71a, the Citadel building is designed to maintain the spatial arrangement of reactor components and heat removal pathways during normal operation and postulated events. Structural, thermal and pressure loads are considered during design of the Citadel building to ensure that the RCCS remains effective in transferring decay heat to the ultimate heat sink.

Consistent with PDC-71b, the Citadel building provides structural protection for the depressurization pathway required during helium release events. Adequate overpressure relief pathways are integrated with the building design to allow controlled discharge of reactor helium in the event of a sufficiently large pipe-break accident, while maintaining structural support and shielding functions.

Consistent with PDC-72, the Citadel building is designed to permit periodic inspection of critical structural areas and depressurization pathways. Visual inspection and embedded strain and temperature monitoring systems are used to support an ongoing surveillance program throughout the facility's operational life. Accessible areas may be inspected on a regular basis, whereas some areas may only be inspected during plant shutdowns or refueling.

3.5.3.3 Conformance with PDC-2 for Meteorological Events

Section 3.2.1 describes the normal wind loads used as design parameters for the Citadel building. Loads from normal winds are in the form of velocity pressure. Section 3.2.2 and Section 3.2.3 describe the high wind loads from tornadoes and hurricanes used as design parameters for the Citadel building. Loads from high winds are in the form of velocity pressure, atmospheric pressure change, and tornado and hurricane missile impacts. Finally, Section 3.2.4 describes the snow loads used as design parameters for the Citadel building.

Consistent with PDC-2, the SR SSCs are located in the Citadel building, which is designed to protect SR SSCs from the effects of design basis normal and high winds and snow. The Citadel building is a {{ }}a(4) concrete structure designed to meet ACI 349-13 (Reference 3-19) with internal steel structures designed in accordance with ANSI/AISC Standard N690-18 (Reference 3-20). Both ACI 349-2013 and ANSI/AISC N690-18 are standards specific to the design of SR nuclear structures and have built-in margin. ACI 349 and ANSI/AISC N690-18 are used to design a structure that can withstand the loads as determined in Section 3.2. By designing the Citadel building in accordance with these two standards, the Citadel building satisfies PDC-2 for design basis loads from normal winds, high winds, and snow, as discussed in Section 3.2.

3.5.3.4 Conformance with PDC-2 and PDC-4 for Internal and External Flooding

This section describes how the design basis for the Citadel building provides reasonable assurance that potential water damage will not preclude SR SSCs from performing their SR functions. Section 3.3 characterized the design basis loads related to external and internal flooding postulated events. This section describes how the Citadel building is designed to address those loads.

3.5.3.4.1 External Flood Design Features

Consistent with PDC-2, all SR SSCs are located in the Citadel building, which is designed to protect these SSCs from the effects of design basis external flooding as described in Section 3.3.

Section 3.3 establishes that, under the design basis flood event, no loads are imposed on the above grade plant structures. The Citadel extends approximately {{ }}a(4) below grade and is engineered to withstand buoyant forces and hydrostatic pressure, including groundwater associated with the design basis flood. The top exterior surface of the Citadel building is located at ground level, and the surrounding ground surface is sloped away from the structure to prevent the accumulation of precipitation or snowmelt on the top of the building. Detailed grading and drainage features will be finalized during detailed design and documented in the Operating License Application.

The Citadel is a {{ }}a(4) concrete structure designed in accordance with ACI 349-2013, which provides nuclear-specific requirements and inherent design margin for SR structures. This standard ensures the Citadel can withstand the postulated external flood loads from Section 3.3, consistent with PDC-2. For buoyant forces acting on the Citadel building during a flood level at or near grade, the structural weight of the Citadel offsets potential uplift forces.

During construction of the Citadel building, {{ }}a(4) will be provided in the Operating License Application. {{ }}a(4)

Defense-in-depth is provided by additional design features including external waterproofing measures and site grading and drainage provisions that limit floodwater ingress and maintain structural integrity under design basis conditions. Small quantities of water ingress/seepage are accommodated {{ }}a(4) These features support management of minor seepage but are not credited as SR features required to maintain SR SSC safety functions.

3.5.3.4.2 Internal Flood and Spray Design Features

This section describes the design features that satisfy PDC-4 with respect to protection from internal flooding and spray hazards for SR SSCs.

SR SSCs that could be vulnerable to water damage from internal spray or floods are elevated above the floor, shielded, or otherwise protected from potential spray sources. The Citadel building's fire suppression system {{ }}a(4) relies on passive fire resistance in inaccessible regions. {{

}}a(4)

To mitigate the potential effect of these hazards, the following features are included in the design of the facility:

- A {{ }}a(4) impingement shield is placed between the Reactor Vessel and RCCS standpipes to prevent water spray from reaching the vessel. Pipe headers and cassette configuration limit water discharge and pumping ceases upon leak detection to prevent replenishment. The Reactor Vessel and IHX Vessel are elevated {{
}}a(4)
- For water systems with SSCs located in the SR portion of the Citadel building, the amount of water is limited by design; water volumes contained within these systems are consistent with the Citadel building design bases pertaining to flood prevention and mitigation. Upon detection of a leak, each system is designed to terminate the flow by closing valves or ceasing to pump, thereby allowing only a limited amount of fluid to spill into the facility. The Unit Monitoring System (Section 7.8) will detect and alert operators to the presence of radioactive contaminants in the Citadel building.
- SR SSCs are raised an adequate distance above the floors of the Citadel building {{
}}a(4)
- Water is directed away from enclosures for SR equipment, and sloped floors and curbs preclude water entry into these areas. {{
}}a(4)
- Molten salt system piping is routed to ensure that any rupture does not affect SR components. Redundant isolation valves limit the volume of salt that can leak. Determination of the cases which are automatic or require operator action will be detailed in the Operating License Application. Because salt freezes at around 230°C, its spread is expected to be limited. Components potentially exposed to molten salt leakage will be evaluated for thermal shock, chemical interaction and degradation.

All pipes, vessels, and tanks with the potential to flood or spray the Citadel building are seismically qualified according to the seismic design category identified in Section 3.6.2.4. There are pressurized piping systems in the SR portion of the Citadel building, so pipe whip effects will also be considered. Further analysis of the impacts of internal flooding and spraying will be provided with Operating License Application. Flood recovery of the Citadel building is supported {{

}}a(4) the facility HVAC system (Section 9.1) will maintain humidity inside the Citadel building within acceptable bounds until such a time as the pump can be repaired.

3.5.3.5 Conformance with PDC-2 for Earthquakes

Section 3.4 discusses the design basis seismic characteristics that are the input for the design of the Citadel building. The Citadel building is designed consistent with the graded approach in ASCE/SEI 43-19 (Reference 3-16). See Section 3.4 for more information about the graded approach. By meeting ASCE/SEI 43-19, the Citadel building protects SR SSCs from design-basis earthquakes, consistent with PDC-2.

3.5.3.5.1 Seismic Design of the Citadel Building

Seismic qualification of SDC-3 structures follows the requirements of Section 5 of ASCE/SEI 43-19. Structural demands are determined based on the results of the response analysis outlined in Section 3.4.1 of this chapter. In addition to the seismic effects, the effects from gravity, operating loads, and other concurrent loading (e.g., snow) are considered on the structural demands.

Seismic acceptance is checked for both strength- and displacement-based criteria summarized in Section 5.2.2 and 5.2.3 of ASCE/SEI 43-19, respectively, for the applicable limit states. Strength-based qualification of structural elements utilizes, when appropriate, the inelastic energy absorption factors discussed in Section 5.1.3 of ASCE/SEI 43-19 and summarized in Table 5-1 of ASCE/SEI 43-19. Allowable drift and rotation limits are based on the discussion in Section 5.2.3 of ASCE/SEI 43-19 and summarized in Tables 5-2 and 5-3 of ASCE/SEI 43-19.

NSR SSCs that connect between the service building and the Citadel building use design features to accommodate differential displacements of the two structures. Design features include flexible features for piping, ducting and conduit, isolation valves, spray and drip shielding, or other similar design solutions. These features minimize the stresses on the elements due to differential motion between the parts of the building during a design-basis earthquake. This is not an SR function, but the features reduce the likelihood that NSR SSCs would adversely affect an SR SSC's ability to perform its safety function during a seismic event.

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3.5.3.6 Conformance with PDC-4 for Other Hazards

Accidental explosions outside the facility (see Section 2.2) and accidental explosions inside the facility are considered in the design of the SR structures. The Citadel building is constructed {{ }}[a(4)] concrete such that credible accidental external explosions do not jeopardize the ability of SR SSCs located in that portion of the building to perform their safety functions.

Potential internal explosion hazards include: (1) energetic combustion events (e.g., deflagration) and (2) rapid pressure transients that can impose dynamic pressure loads on SR SSCs or their supports (e.g., a postulated pressure boundary rupture). The Citadel design limits credible sources of combustible mixtures within SR areas through material selection and fire protection features. Where pressurized systems could discharge into Citadel spaces, relief pathways are provided to minimize overpressure. Consistent with PDC-3, noncombustible and fire-resistant materials are used wherever possible, particularly in locations containing SR SSCs. The use of combustible materials is limited only to those required for equipment function or regulatory compliance. The Citadel building fire protection features are designed consistent with NFPA 801 (Reference 3-21), such that fires and explosions do not prevent SR SSCs from performing their safety functions. Internal explosions and additional combustible source terms will be considered in the Fire Hazard Analysis, to be provided with the Operating License Application.

Within the Citadel building, the principal process fluids are helium (primary coolant), molten salt (secondary coolant), and water. These fluids are noncombustible and therefore do not present a combustion-driven internal explosion hazard. The molten salt secondary coolant is not substantially reactive with water; however, rapid steam generation could occur if molten salt contacts standing water within the Citadel building. {{ }}[a(4)] Potential water inventories that can credibly accumulate in the IHX cavity (e.g., HPS leakage, helium circulator cooling jacket leakage, and groundwater intrusion) are limited through isolation, drainage and inventory controls. {{ }}[a(4)]

Hydrogen generation mechanisms inside the reactor is limited relative to water-cooled reactors because the fuel does not have zircaloy cladding and does not use water as the primary coolant. Tritium is produced from neutron reactions with B_4C Li impurities in graphite, and He-3 impurities in the helium (primary coolant). Bounding calculations indicate that the total lifetime generated inventory of hydrogen-bearing species is several orders of magnitude smaller than the quantity required to create a flammable mixture in the Citadel building rooms, even when neglecting the decay of tritium and its removal from the system by the HPS. One pathway for hydrogen production is through reaction of the graphite with high-temperature steam. {{ }}[a(4)] If the pressure boundary depressurizes below setpoints, the RPS actuates to place the reactor in a safe shutdown condition. For these reasons, hydrogen deflagration is considered a minimal internal explosion hazard.

Burst disks and overpressure relief pathways within the Citadel building are provided to manage postulated dynamic pressure loads associated with a primary or secondary coolant boundary rupture and to limit loads transmitted to SR SSCs. Helium make-up cylinders {{ }}$^{a(4)}$ do not present a credible hazard to the SR SSCs inside the Citadel building.

Electrical hazards that could produce localized energetic events such as electric arc flash are controlled by use of appropriately rated enclosures and electrical equipment designed and installed in accordance with applicable electrical codes and standards. I&C cabinets and enclosures conform to NEMA 250 (Reference 3-22) and IEC 60529 (Reference 3-23), which establish enclosure type and degree-of-protection requirements. These standards define enclosure construction and ingress protection characteristics appropriate for the installed environment and support control of internal fault effects to the extent practicable. Additional programmatic controls for electrical work practices (e.g., arc flash hazard analysis and labeling) will be provided in the Operating License Application.

The U. of I. research reactor will be licensed as a Class 104(c) nonpower production or utilization facility (NPUF) under 10 CFR 50.21(c) (Reference 3-26). Consistent with the U. of I. TR, "Applicability of Nuclear Regulatory Commission Regulations" (Reference 3-1), the requirements of 10 CFR 50.150, "Aircraft impact assessment" are not part of the licensing basis for this facility.

Additional details about the structural design features for the Citadel building informed by the results of the fire hazard and internal explosion analyses will be provided in the Operating License Application.

3.5.4 Testing and Inspections

The Citadel building is a below-grade {{ }}$^{a(4)}$ concrete structure designed in accordance with ACI 349-13 and ASCE/SEI 43-19 for SDC 3.

In compliance with ACI 349-13, prior to and during construction, inspections confirm {{ }}$^{a(4)}$ quality. Waterproofing integrity is verified in accordance with project-specific design requirements and quality assurance procedures.

A monitoring and inspection program will be implemented to verify long-term structural integrity, including periodic evaluation of concrete condition, water ingress prevention systems, and seismic performance. The Citadel building will incorporate inspections to monitor:

- Concrete stress variations and temperature.
- Surface coatings, seals, and gaskets for degradation.
- Steel structures for signs of corrosion and concrete for signs of ageing.

{{ }}$^{a(4)}$ Visual inspection of the internal face of the concrete and steel structures in the IHX Cavity and other parts of the Citadel building will be performed during shutdowns in accordance with the in-service inspection program requirements. Accessible areas may be inspected regularly, whereas some areas may only be inspected during plant shutdowns or refueling. Detailed procedures will be provided in the Operating License Application.

Table 3-1 Load Combinations for the Citadel building

Load Category	Load Combination*
Normal **	$D + F + L + T_o + H + C_{cr} + L_r / S / R$ $D + F + L + H + C_{cr} + L_r / S / R$
Severe environmental	$D + F + L + H + E_o$ $D + F + L + H + W$
Extreme environmental	$D + F + L + T_o + H + C_{cr} + E_{ss}$ $D + F + L + T_o + H + C_{cr} + W_t$
Abnormal	$D + F + L + H + C_{cr} + T_a + P_a$ $D + F + L + H + Y_m + T_a + P_a + E_{ss}$

Table 3-1 Load Combinations for the Citadel building (Continued)

Load Category	Load Combination*
*Load combination presented in this table identifies the types of loads considered to act simultaneously. The load sets are presented in unfactored for. Load factors and detailed load combination rules for the Citadel building are applied in accordance with the applicable design standard. **The governing case is taken as the envelope of the load combinations considering the contribution of L_r, S or R. Piping and equipment loads are conservatively accounted for by representing their effects as enveloped masses applied through equivalent floor loads. **Load Nomenclature:** D Dead loads L Live loads F Fluid loads T_o Thermal loads resulting from temperature distributions within the concrete structure during startup, normal operation, or shutdown conditions. T_a Thermal loads resulting from temperature distributions within the concrete structure during accident conditions H Loads resulting from soil weight and lateral soil pressure C_{cr} Crane load corresponding to the rated lifting capacity of the crane L_r Roof live load S Snow load R Rain load W Wind load under normal operating conditions W_t Load generated by the design basis tornado, including loads due to tornado wind pressure, tornado-created differential pressures and tornado-generated missiles. E_o Loads generated by 1/3 of design basis earthquake (the design basis earthquake (DBE) is also the safe shutdown earthquake (SSE)) E_{ss} Loads generated by SSE P_a Differential pressure load generated by a postulated pipe break accident Y_m Accidental loads resulting from missile impact	

3.6 SYSTEMS AND COMPONENTS

This section describes the design bases for the systems and components of the research reactor required to function for safe reactor operation and shutdown. These SSCs include:

- The reactor core system,
- The Reactor Internals System (RIS),

- The Reactor Protection System (RPS),
- The reactor Vessel Systems (VS),
- The Reactivity Control and Shutdown System (RCSS),
- The Reactor Cavity Cooling System (RCCS), and
- The Citadel building.

The design bases consider normal operating conditions, and postulated events, including internal and external hazards such as seismic hazards.

The design bases and evaluation for the Citadel building are discussed in Section 3.5. Section 3.6.1 describes the fundamental safety functions performed by SR SSCs, and Section 3.6.2 describes the safety classification of SSCs with reference to these functions.

3.6.1 General Design Basis Information

3.6.1.1 Functional Requirements

The SSCs relied upon in the safety analysis to mitigate the consequences of postulated events serve one or more of the three fundamental safety functions listed below:

1. To shut down the reactor and maintain it in a safe shutdown condition.

2. To remove residual heat following a reactor shutdown.

3. To prevent or mitigate the consequences of accidents that could result in unacceptable radiological consequences to workers or the public, such that the radiological dose acceptance criteria for licensing basis events described in Chapter 13 are met.

The SR SSCs each perform one or more of these functions in response to accident conditions and external events:

- The Reactor Core System maintains core geometry and heat transfer paths under seismic loads and loss-of-coolant scenarios, ensuring that reactivity control remains guaranteed and decay heat can be removed. Tri-structural isotropic (TRISO) ceramic nuclear fuel provides inherent radionuclide containment and high thermal margin.
- The Reactor Internals System provides structural support and lateral restraint to the reactor core, ensuring that reactivity control remains guaranteed and that the geometry remains coolable during normal operation and postulated events as well as seismic events. The RIS provides helium flow paths and maintains neutron shielding integrity to limit irradiation effects and activation of the vessel and surrounding structures/environment.
- The RPS initiates automatic trips for postulated events. Redundant and independent fail-safe systems ensure that reactivity can always be controlled by bringing the reactor to a safe shutdown condition, even in the absence of electrical power.
- The VS maintains structural integrity and geometry for the insertion of negative reactivity and to sustain a heat-transfer path for removal of decay heat.

- The RCSS ensures rapid insertion of the control rods by gravity during a reactor trip, thereby controlling reactivity. Actuation of the RCSS requires no power input.
- The RCCS passively removes heat from the Reactor Cavity via water circulation during normal operation and via boil-off during postulated events. The RCCS is credited with decay heat removal in postulated events.
- The Citadel building protects the VS, RCSS, and other SR SSCs from seismic, flooding, and external hazards. See Section 3.5.1 for a description of the Citadel building structure.
- The design of the reactor core also supports the control of reactivity via overall negative reactivity feedback. See Section 4.5 for a description of the core design.

3.6.1.2 Environmental Qualification

SSCs are designed for the full range of environmental conditions expected during normal operation, maintenance, testing, and postulated events. Qualification addresses:

- Design-basis temperature and pressure conditions (including bounding conditions) during startup, steady-state, and transient conditions,
- Radiation exposure over the design life, including cumulative neutron fluence and gamma dose,
- Chemical and physical degradation effects, such as corrosion, erosion, and oxidation,
- Dynamic effects, including missile impact, pipe whip, and fluid discharge, consistent with PDC-4, and
- Thermal cycling and mechanical wear, ensuring materials and joints maintain integrity under repeated load variations.

3.6.1.3 Reliability Considerations

Design bases for SSC reliability include:

- Number of operational cycles, with fatigue analysis per the relevant code (i.e., ASME BPVC Section III, Division 5, Subsection HB for the helium pressure vessels),
- Vibration and dynamic loads from normal operation and seismic events,
- Friction and wear allowances, supported by material selection and lubrication strategies,
- Material strength and fracture toughness under combined thermal, pressure, and seismic loads, and
- Operating environment effects, including irradiation-induced degradation and high-temperature creep for metallic components.

3.6.1.4 Principal Design Criteria

Principal Design Criteria, identified in the U. of I. Topical Report, "Micro Modular Reactor (MMR) Principal Design Criteria" (Reference 3-3), and approved by the NRC (Reference 3-4), were used to develop design bases for the SSCs in the nuclear plant. Global PDC identified as applicable to all SSCs are as follows:

- **PDC-1,** "Quality standards and records"
- **PDC-2,** "Protection against natural phenomena"

- **PDC-3,** “Fire protection”
- **PDC-4,** “Environmental and dynamic effects”

3.6.1.5 Service Levels and Loads

Load combinations and service levels are established to ensure SSCs maintain structural integrity and functional reliability under all anticipated conditions, including cyclic thermal stresses, vibration-induced fatigue, and wear mechanisms. Service levels and loads will be consistent with the requirements of ASME Section III Division 5 (Reference 3-25) and ASCE/SEI 7-22 (Reference 3-11). An example of how loads will be computed in the component design specification is shown in Table 3-1. Loads that will be defined in component design specifications include, at a minimum, the following items:

1. Pressure Loads,
2. Thermal Loads,
3. Seismic Loads (SSE),
4. Piping Loads and Reactions,
5. Nozzle Loads,
6. Bolting Loads due to Flanges, Heads, etc.,
7. Fluid Loads,
8. Fluid Transients,
9. Support Interactions, and
10. Test Loads.

Factors for calculation of individual loadings of the SCCs will be described in detail in the applicable component design specifications during the design of those SSCs.

3.6.2 Classification of Structures, Systems, and Components

SSCs are assigned safety, seismic and quality classifications consistent with their credited safety functions. These classifications are consistent with NUREG-1537 guidance (Reference 3-7 and Reference 3-8) and comply with U. of I.’s Quality Assurance Program (Reference 3-26). The SSC classification methodology for the research reactor is provided in U. of I. Topical Report, “Event Sequence Identification and SSC Classification Methodology” (Reference 3-9) and reviewed by the NRC (Reference 3-10). Table 3-3 summarizes the classification of SSCs. Note that not all SSCs within an SR system may be classified as SR.

3.6.2.3 Safety Classification

SSCs are classified as either SR or NSR. SR SSCs are those relied upon to remain functional during and following design-basis events to ensure achievement of the fundamental safety functions listed in Section 3.6.1.1.

This definition is adapted from 10 CFR 50.2 for high-temperature gas-cooled reactors (Reference 3-27), reflecting the high-pressure helium-cooled design of the research reactor and its reliance on inherent and passive safety features rather than containment pressure boundaries. SSCs that are not credited to perform (or support the performance of) these functions are classified as NSR. Table 3-3 summarizes safety and seismic classification of, and the section of the PSAR which addresses, each SSC.

3.6.2.4 Seismic Classification

SSCs are designed according to their safety classification. The credited safety systems designed to function in a postulated event are described in Chapter 13. Consistent with RG 1.29 (Reference 3-15), SR SSCs are assigned Seismic Category I. These SSCs are designed to withstand the DBE while maintaining their ability to perform their safety functions, so a graded, performance-based seismic design methodology is applied consistent with ASCE/SEI 43-19 (Reference 3-16). SR SSCs credited for safe shutdown or for preventing radiological releases are assigned to SDC-3 in accordance with ASCE/SEI 43-19.

In addition to these requirements, the SR electrical equipment and SR active mechanical equipment are seismically qualified in accordance with IEEE 344-2020 (Reference 3-28) and ASME QME-1 (Reference 3-29), in accordance with RG 1.100 (Reference 3-30), respectively.

NSR SSCs are generally designed in accordance with the local building code (IBC 2024; Reference 3-17) and ASCE/SEI 7-22 (Reference 3-11), as described in Section 3.4.2. Where NSR SSCs interface with the Citadel building, design features (e.g., flexible piping/ducting/conduit features, isolation valves, and shielding) are provided to accommodate differential displacement and reduce the likelihood that an NSR SSC adversely affects an SR SSC's ability to perform its safety function during a DBE.

Certain NSR SSCs are identified as potentially affecting the ability of SR SSCs to perform their credited safety functions if the NSR SSCs were to fail during a DBE. These SSCs remain NSR but are designated Seismic Category I. These selected NSR SSCs are designed in accordance with ASCE/SEI 43-19 to ensure that their seismic response does not adversely affect SR SSCs.

The table below shows the seismic classification and applicable design code for each SSC. For NSR SSCs, selected components may be designed to Seismic Category I criteria where failure of the component could adversely affect the ability of an SR SSC to perform its safety function (e.g., through seismic interaction, induced failure, or loss of required support). Similarly, some SR SSCs include NSR equipment that is designed to Seismic Category I consistent with this approach. This classification approach is summarized in Table 3-2. The Operating License Application will include a component-level listing of Seismic Category I items within NSR SSCs and will document the evaluation basis used to ensure that SR SSC safety functions are not compromised.

Table 3-2 Classification Overview

Safety Classification	Seismic Failure Criteria	Seismic Classification	Design Code
SR SSCs*	Failure Not Permitted	Seismic category I / SDC-3	ASCE/SEI 43-19
NSR SSCs	Failure Not Permitted if it could impact the performance of an SR SSC	Seismic category I / SDC-3	ASCE/SEI 43-19
	Failure Permitted, failure does not impact a SR SSC	Non-seismic category I	IBC 2024 / ASCE/ SEI 7-22
*SR SSCs may have NSR subcomponents which are seismically classified in accordance with the impacts of their failure.			

Table 3-3 shows the classification of SSCs. {{

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Table 3-3 Classification of SSCs

System	PSAR Section	Safety Classification	Classification / Seismic Design Category	Seismic Design Code
Reactor Core System				
TRISO ceramic nuclear fuel	4.2	Safety related (SR)	Seismic category I / SDC-3	ASCE/SEI 43-19
Moderator Graphite	4.2			
Reflector Graphite	4.4			
Reactor Protection System				
Trip Breaker	7.4	Safety-related (SR)	Seismic category I / SDC-3	ASCE/SEI 43-19
Flux Detectors	7.5			
Pressure Transducer	7.5			
Temperature Sensors	7.5			
Helium Flow Meters	7.5	Non-safety related (NSR)	Non-seismic category I	IBC 2024 / ASCE/ SEI 7-22
Inverter Supply	7.4			
Control Cabinet	7.4	Safety related (SR)	Seismic category I / SDC-3	ASCE/SEI 43-19
Readout Monitor	7.6	Non-safety related (NSR)	Non-seismic category I	IBC 2024 / ASCE/ SEI 7-22

Table 3-3 Classification of SSCs (Continued)

<table>
<tr><th>System</th><th>PSAR Section</th><th>Safety Classification</th><th>Seismic Classification / Seismic Design Category</th><th>Seismic Design Code</th></tr>
<tr><td colspan="5">Vessel Systems</td></tr>
<tr><td>Reactor Vessel</td><td>4.3</td><td>Safety related (SR)</td><td rowspan="5">Seismic category I / SDC-3</td><td rowspan="5">ASCE/SEI 43-19</td></tr>
<tr><td>Cross Connection Vessel</td><td>4.3</td><td rowspan="3">Non-safety related (NSR)</td></tr>
<tr><td>IHX Vessel</td><td>4.3</td></tr>
<tr><td>Circulator Vessel</td><td>4.3</td></tr>
<tr><td>Vessel Supports</td><td>4.3</td><td>Safety related (SR)</td></tr>
<tr><td>Pressure Relief Valves</td><td>4.3</td><td>Safety related (SR)</td><td>Seismic category I / SDC-3</td><td>ASCE/SEI 43-19</td></tr>
<tr><td colspan="5">Reactor Internals System</td></tr>
<tr><td>All subsystems</td><td>4.2</td><td>Safety related (SR)</td><td>Seismic category I / SDC-3</td><td>ASCE/SEI 43-19</td></tr>
<tr><td colspan="5">Reactor Control and Shutdown System</td></tr>
<tr><td>Control Rods</td><td>4.2</td><td rowspan="2">Safety related (SR)</td><td rowspan="2">Seismic category I / SDC-3</td><td rowspan="2">ASCE/SEI 43-19</td></tr>
<tr><td>Control Rod Drive Units</td><td>4.2</td></tr>
<tr><td>Controller</td><td>7.3</td><td>Non-safety related (NSR)</td><td>Non-seismic category I</td><td>IBC 2024 / ASCE/ SEI 7-22</td></tr>
<tr><td colspan="5">Reactor Cavity Cooling System</td></tr>
<tr><td>Passive-cooling Components</td><td>6.3</td><td>Safety related (SR)</td><td>Seismic category I / SDC-3</td><td>ASCE/SEI 43-19</td></tr>
<tr><td>Other Components</td><td>6.3</td><td>Non-safety related (NSR)</td><td>Non-seismic category I</td><td>IBC 2024 / ASCE/ SEI 7-22</td></tr>
<tr><td colspan="5">Citadel</td></tr>
<tr><td>All subsystems</td><td>3.5</td><td>Safety related (SR)</td><td>Seismic category I / SDC-3</td><td>ASCE/SEI 43-19</td></tr>
</table>

Table 3-3 Classification of SSCs (Continued)

System	PSAR Section	Safety Classification	Seismic Classification / Seismic Design Category	Seismic Design Code
Other systems with no safety related subsystems				
Primary Coolant Makeup System	5.5	Non-safety related (NSR)	Seismic category I / SDC-3	ASCE/SEI 43-19
HPS	5.4			
HVAC Systems	9.1			
Fuel Handling and Storage System	9.2			
Fire Protection System	9.3			
Water Services Systems	9.7			
Heat Transport Systems	5.2	Non-safety related (NSR)	Non-seismic category I	IBC 2024 / ASCE/ SEI 7-22
Thermal Energy Storage System	5.3			
Information, Instrumentation and Control Systems	7.2			
Reactor Control System	7.3			
Unit Monitoring Systems	7.8			
Electrical Auxiliary Power Supply Systems	8.2			
Communication Systems	9.4			
Equipotential Bonding/ Earthing	9.7			
Lifting and Rigging Equipment	9.7			
Chemical Dosing System	9.7			
Waste Processing	11.2			
Radwaste Drains	11.2			
Access Control System	12.8			

3.6.2.4.1 Seismic Qualification by Analysis

Seismic qualification by analysis follows Section 8.2 of ASCE/SEI 43-19. Depending on the characteristics and complexities of the subsystem or equipment, qualification by analysis is accomplished by either equivalent static analysis methods or dynamic analysis methods.

There are limitations to qualification by analysis. Per ASCE/SEI 43-19:

- Qualification of active electrical equipment by analysis is not performed.
- Qualification of active mechanical equipment by analysis may be permitted if the component is such that the functionality during a seismic event can be established and a margin of loss of functionality during such an event can be quantified.

- Qualification of active mechanical components by analysis shall be justified.

Seismic qualification by analysis is typically implemented for subsystems and equipment structural integrity related capacities (e.g., anchorage, pressure boundary / rupture, serviceability deformations, etc.).

3.6.2.4.2 Seismic Qualification by Testing

Seismic qualification by testing follows Section 8.3 of ASCE/SEI 43-19. Qualification by test is typically used for SSCs for which qualification by analysis is not permitted and for SSCs where dynamic behaviors are not sufficiently understood to support qualification by analysis.

3.6.2.5 Quality Classification

The quality classification for SSCs conforms with the requirements of U. of I.'s Quality Assurance Program (Reference 3-26), which is discussed in Section 12.9.

3.7 REFERENCES

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3-7 U.S. Nuclear Regulatory Commission (NRC), NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Format and Content."

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USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 4 - REACTOR DESCRIPTION

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
AGR	Advanced Gas Reactor
AGR-1	First AGR Irradiation and Safety Testing Campaign
AGR-2	Second AGR Irradiation and Safety Testing Campaign
ALARA	as low as reasonably achievable
B4C	boron carbide
BA	burnable absorbers
BOC	beginning-of-cycle
CCV	cross-connection vessel
CFR	Code of Federal Regulations
CR	Control Rod
CRDU	control rod drive unit
CTC	combined temperature coefficient
CVI	chemical vapor infiltration
DOE	Department of Energy
EFPY	effective full power years
EOC	end-of-cycle
EOL	end-of-life
EPRI	Electric Power Research Institute
FQMTR	Fuel Qualification Methodology Topical Report
FTC	fuel temperature coefficient
HPRS	helium pressure relief system
HTGR	high-temperature gas-cooled reactor
HTS	Heat Transport System
I&C	instrumentation and control
IHX	intermediate heat exchanger
IPyC	inner pyrolytic carbon
LEU	Low-enriched uranium
MOC	middle-of-cycle
MTC	moderator temperature coefficient
MTR	material test reactors
NRC	Nuclear Regulatory Commission
NSR	non-safety-related
ODSL	outer dense surface layer
OLA	Operating License Application
OPyC	outer pyrolytic carbon
PDC	Principal Design Criteria
PLOFC	Pressurized Loss of Forced Cooling

LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
PyC	pyrolytic carbon
RCS	Reactor Control System
RCCS	Reactor Cavity Cooling System
RCSS	Reactivity Control and Shutdown System
RIS	Reactor Internals System
RPS	Reactor Protection System
RTC	reflector temperature coefficient
RV	reactor vessel
SARRDL	specified acceptable radionuclide release design limits
SCA	Secondary Control Area
SCALE	Standardized Computer Analyses Licensing Evaluation
SER	Safety Evaluation Report
SiC	silicon carbide
SR	safety-related
SSC	structures, systems and components
TCR	Transformational Challenge Reactor
TEDE	total effective dose equivalent
TRISO	TRi-structural ISOtropic
UCO	uranium oxycarbide
VS	Vessel System

CHAPTER 4 REACTOR DESCRIPTION

4.1 SUMMARY DESCRIPTION

The U. of I. research reactor is a graphite-moderated, helium cooled high-temperature gas-cooled reactor (HTGR). It is designed with a functional capability to achieve a rated thermal power of up to 45 MWth with a reactor primary coolant outlet temperature of 660°C and an operational lifetime of 40 years. The reactor design employs micro-encapsulated tri-structural isotropic (TRISO) fuel particles embedded in a ceramic matrix to form an annular fuel pellet. The TRISO particles and ceramic pellet matrix perform as functional containment which, alongside the moderator and coolant, provides improved fission product retention, chemical and thermal stability, and strong negative reactivity feedback coefficients under all conditions.

The research reactor is designed for passive safety response to all postulated events and relies on functional containment as the primary means to limit release of radioactivity to the environment, with defense-in-depth provided by confinement through reactor systems and facility design.

The key design features of the reactor are described below:

- The fuel is comprised of TRISO particles, which provide highly effective fission product retention capability. The superior fission product retention capability of TRISO fuel particles enables the concept of “functional containment” in which these particles serve as the first retention barriers when operated within the range of qualification parameters (Reference 4-1).
- The TRISO particles are encased in a matrix made of silicon carbide (SiC) that provides an additional layer of defense-in-depth for the retention of fission products by functional containment. The matrix and the TRISO, together, form the fuel pellets.
- The low power density and large thermal inertia of the core lead to slow heat-up during loss of forced cooling events.
- Low thermal power results in a small inventory available for release of the most limiting short-lived fission products for public safety, such as ^{131}I. The increased inventory of long-lived fission products associated with a long core life is addressed by the defense-in-depth approach to functional containment.
- The low power density also reduces the decay heat that must be managed in case of postulated accidents, facilitating passive decay heat removal.
- Reactivity is controlled by a combination of negative reactivity feedback, movable control rods, and fixed burnable absorbers.
- Heat transfer fluid used for core cooling during normal operation is helium. It is an inert, chemically stable, single-phase gas.
- Safety-related core cooling is passive and capable of maintaining fuel and component temperatures below limits with no helium, electrical power, or operator action.

- Secondary heat transfer is performed by a molten salt loop that effectively isolates the reactor from any of the events or accidents that may occur in the adjacent power conversion system.
- The reactor core, primary coolant system, protection systems and all safety-related systems are located below grade. These are surrounded by a concrete structure (i.e., the Citadel building) which provides protection against external hazards but is not credited as a containment structure.

This chapter provides a description of the reactor core, vessel system, and biological shield. The nuclear design and thermal hydraulic design are described.

Heat is generated in the reactor core through the controlled fission of enriched uranium contained within the TRISO-based ceramic fuel. The generated heat is transferred to the helium coolant circulated within the primary loop. Control of the fission energy generation rate during reactor power operation and assurance of adequate shutdown margin is provided by two separate and independent means, which are movable control rods as well as the negative reactivity feedback that is inherent to the core design. A neutron source is provided for initial startup of the reactor core and to assist with calibration of the ex-core flux detectors. At maximum rated capacity, the reactor is refueled every 3 years, with all fuel replaced in a single refueling outage. Refueling is performed by the remote-operated {{ }}[a(4)] which interfaces with the housing of the control rod drive units (CRDUs) and their standpipes, located on the vessel head.

The design of the reactor core, the reactor internals and the vessels ensure that a coolable and controllable core geometry is maintained under all normal operations and postulated events. The reactor design includes provisions for online monitoring to support control and protection functions, as well as the capability to perform in-service inspection, maintenance, and decommissioning activities. Shielding elements are incorporated with the reactor internal structures, which limits dose to personnel present on site, material degradation of the reactor vessel and concrete structures as well as activation of adjacent components.

A summary of key reactor parameters is provided in Table 4-1, and an overview of the reactor system is shown in Figure 4-1.

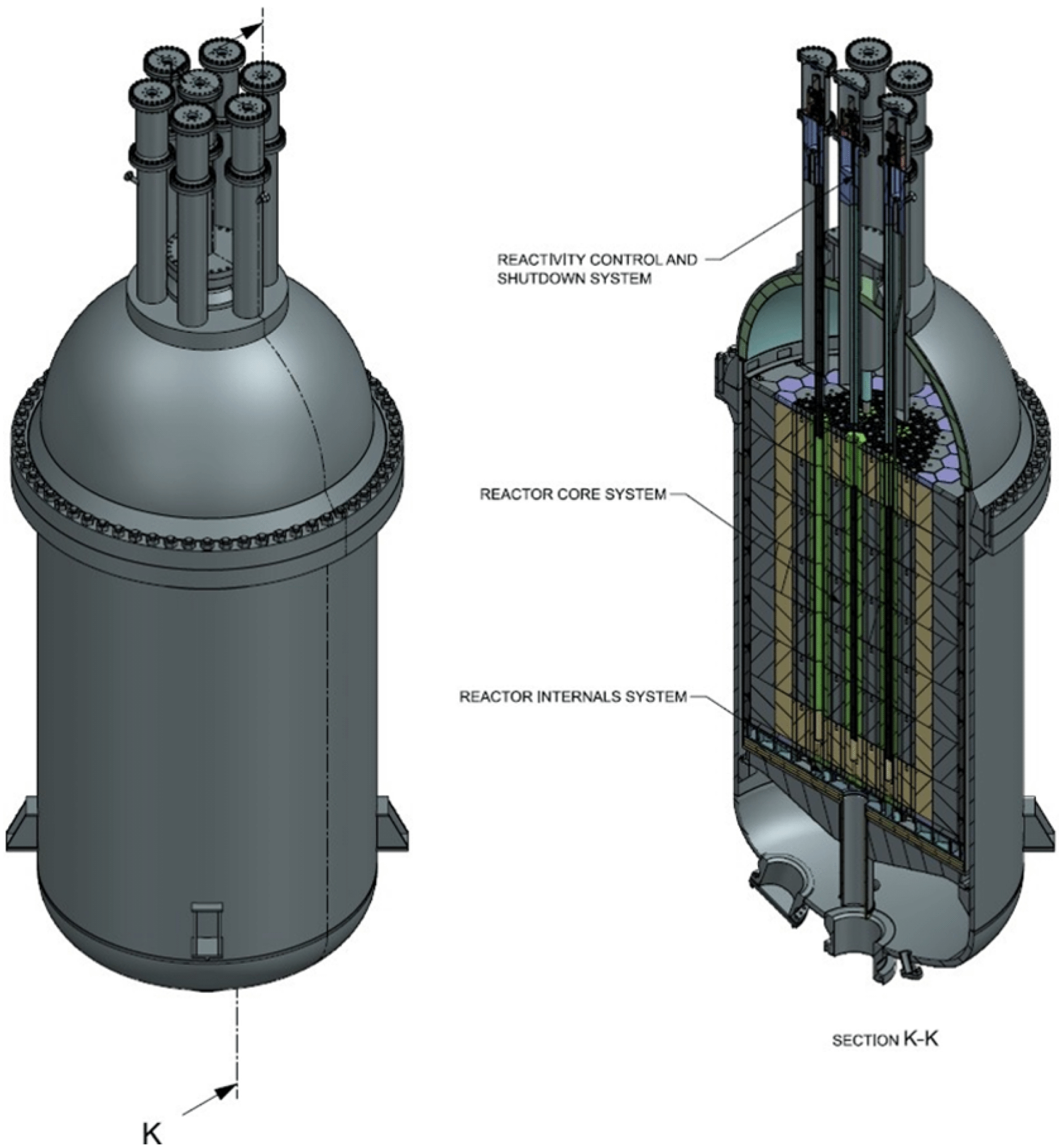
Table 4-1 Reactor Parameters

Feature	Parameter
Fuel Type	TRISO with uranium oxycarbide (UCO) kernel
Fuel Form	Annular pellet with SiC matrix and outer dense surface layer (ODSL)
Fuel Enrichment	Low-enriched uranium (LEU) with ≤9.90 wt% ^{235}U enrichment (referred to as LEU+)
Moderator and Reflector	Nuclear Graphite
Coolant	High-purity helium
Coolant Boundary	Low-alloy steel pressure vessel
Reactor type	Prismatic HTGR
Neutron spectrum	Thermal
Fuel cycle (duration between refueling)	3 years or more

Table 4-1 Reactor Parameters (Continued)

Feature	Parameter
Power per unit	45 MWth
Coolant Temperature (Inlet/Outlet)	300/660°C
Coolant Pressure	6.0 MPa
Functional Containment	TRISO particles Ceramic matrix

Figure 4-1 Reactor System Overview



4.2 REACTOR CORE

This section describes the main reactor core components including the fuel, Reactor Core System, Reactivity Control and Shutdown System, neutron startup sources, and Reactor Internals System.

4.2.1 Reactor Fuel

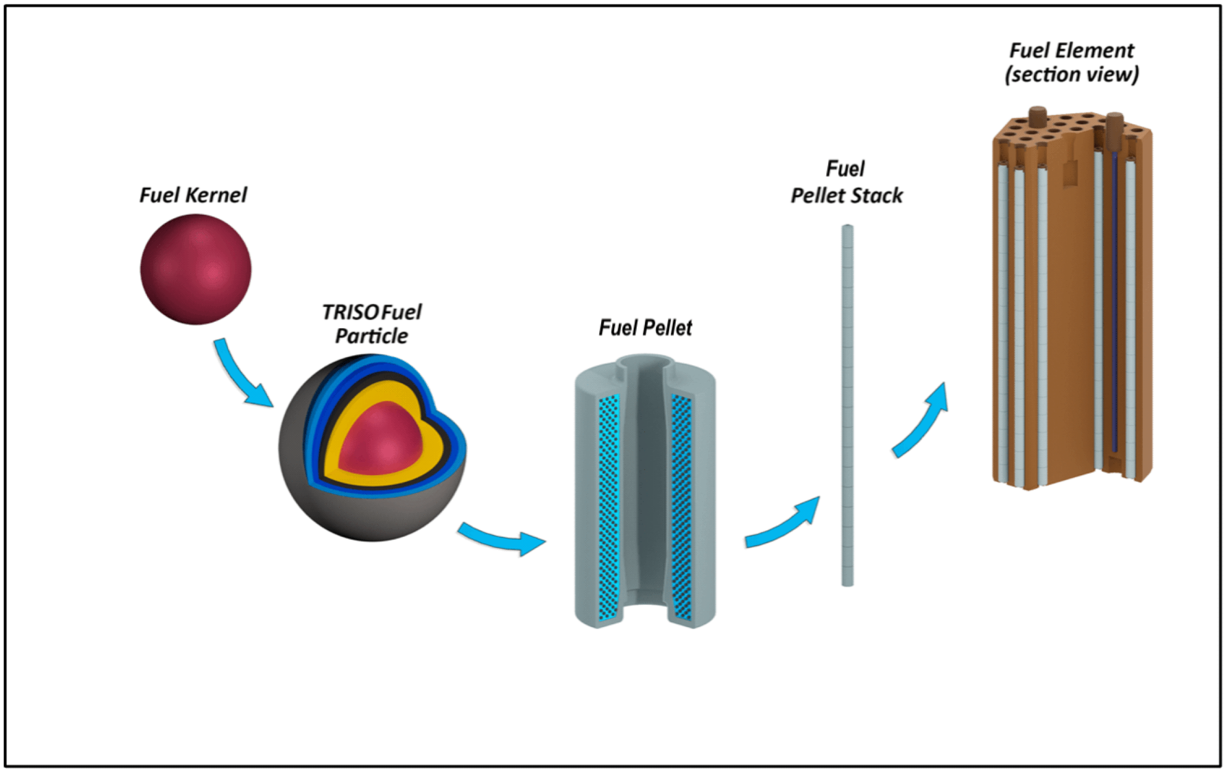
This section provides an overall description of the fuel, its role as part of functional containment, the bases for its design, its operating envelope, its manufacturing, its performance, and its qualification.

4.2.1.1 Fuel Design

The fuel consists of TRISO fuel particles embedded in a ceramic matrix to form an annular cylindrical fuel pellet. These annular fuel pellets are stacked in columns and inserted into prismatic graphite fuel blocks to form fuel elements in the reactor core. The space between the outer wall of the pellets and the graphite block, and the central pellet channel allow the pellets to be cooled from both the inside and outside, with coolant in direct contact with their inner and outer surfaces.

A schematic of the fuel design is provided in Figure 4-2.

Figure 4-2 Fuel Design



4.2.1.1.1 TRISO Fuel Particle Design

The fuel utilizes typical TRISO particles with additional proprietary functional features. Each TRISO particle is composed of a fissile UCO (a mixture of UO_2, UC, and UC_2) kernel containing LEU+ (^{235}U enrichment of ≤9.90 wt%).

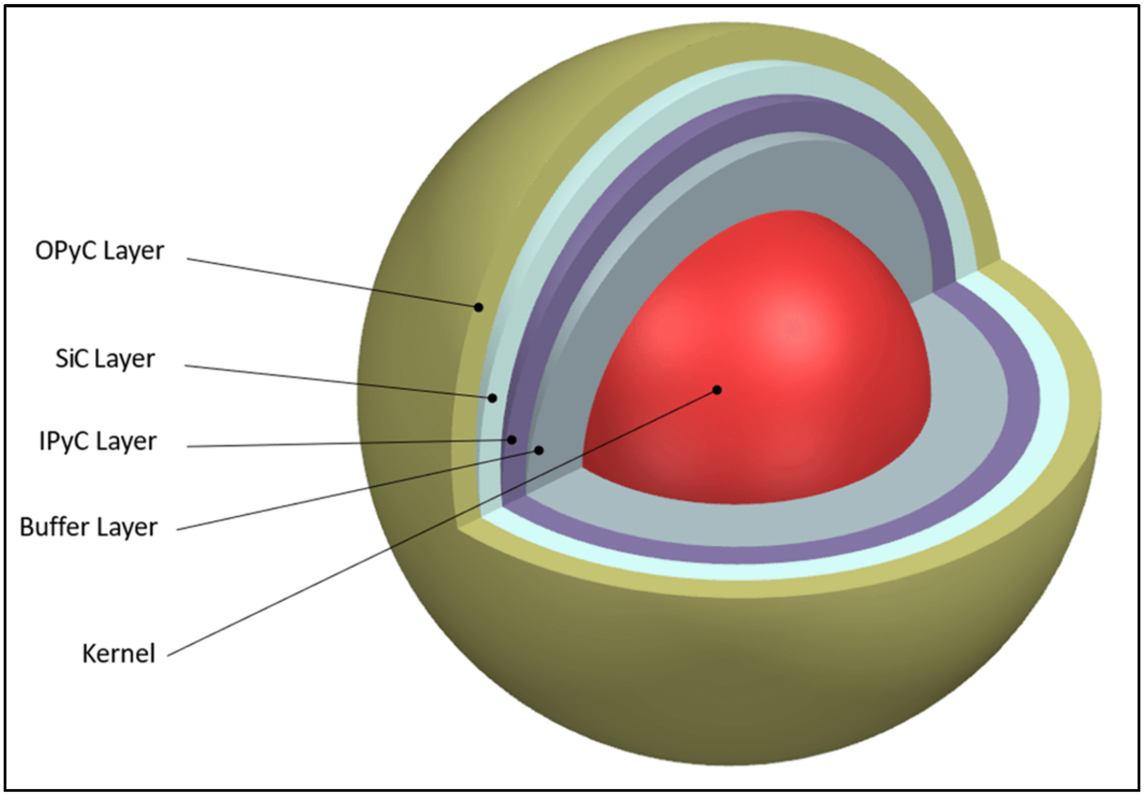
The UCO kernel is surrounded by successive coating layers consisting of a porous carbon buffer layer and three outer coating layers comprising a SiC layer sandwiched between two dense pyrolytic carbon (PyC) layers, the inner PyC (IPyC) and outer PyC (OPyC) layers (Figure 4-3). Additional proprietary functional features were added to the typical TRISO design to improve overall irradiation performance, as described in Section 2.2.1 of the "Fuel Qualification Methodology Topical Report" (FQMTR) (Reference 4-2).

The TRISO kernel contains the fissile material in the form of UCO. UCO kernel chemistry improves on historical UO_2 TRISO fuel design by limiting production of CO through the addition of carbon. Oxygen released from the kernel during fission reacts with the UC_x phases to form uranium oxide, preventing the formation of CO by reaction with the carbon layers surrounding the kernel. Mitigating CO production reduces risk of kernel migration, over-pressurization, and chemical attack on the SiC layer. Nevertheless, in UCO kernels, the oxygen content is kept sufficient to oxidize fission products that could otherwise diffuse through the kernel and IPyC layer as high-mobility carbides. These mobile fission product carbides would attack the SiC layer and potentially result in release from the TRISO particle.

The porous carbon buffer layer, located between the kernel and IPyC layer, mechanically isolates the IPyC layer from kernel swelling, absorbs recoiling fission fragments, and provides a void volume to accommodate fission gases and limit buildup of internal pressure. The SiC layer acts both as a main load-bearing structural layer against internal fission gas pressure and as a main barrier to metallic fission product release. It is sandwiched between the two dense PyC layers. The IPyC and OPyC layers help to maintain the mechanical integrity of the TRISO particle and provide effective barriers to fission gas release.

The exact geometry, properties, and functions of the TRISO fuel particle components are described in Section 2.2.2 of (Reference 4-2).

Figure 4-3 Schematic of a Typical TRISO Fuel Particle



4.2.1.1.2 Ceramic-Matrix Fuel Pellet Design

The fuel pellet is a SiC annular cylinder that contains the TRISO particles. It consists of an inner fueled region surrounded by a fuel-free zone coated with a high-density SiC ODSL, shown in Figure 4-4. The fuel-free zone isolates the TRISO particles from mechanical contact with the ODSL. The ODSL provides an additional retention barrier to the release of fission products. It is designed to limit radionuclide release over the lifetime of the fuel, in case of those becoming present outside of the TRISO particles.

Because the ceramic-matrix TRISO fuel is a relatively new fuel form with limited operational experience regarding its irradiation performance, the fuel pellets will be tested using the fuel qualification methodology described in Section 6 of (Reference 4-2) and summarized in Section 4.2.1.7.

Figure 4-4 Graphical Description of a Ceramic Fuel Pellet

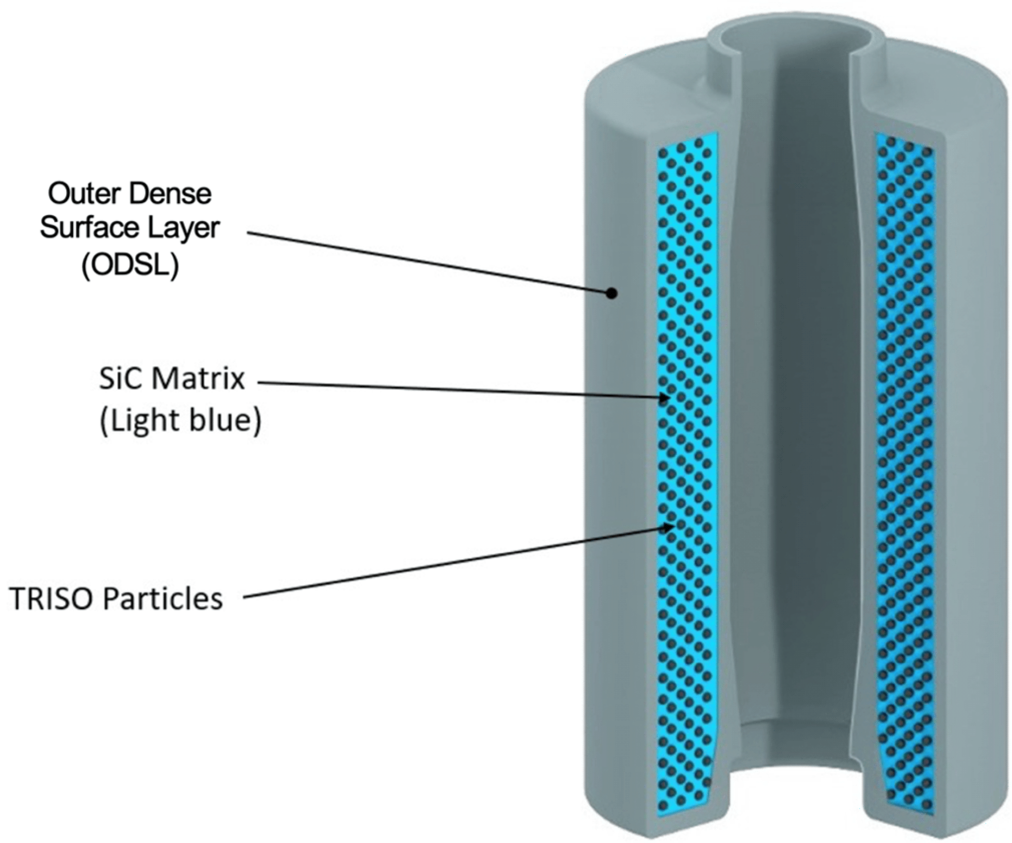


Table 4-2 summarizes the fuel pellet dimensions and properties. The exact geometry, properties, and functions of the ceramic-matrix fuel pellet components are described in Section 2.2.3 of (Reference 4-2).

Table 4-2 Fuel Pellet Dimensions and Properties

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}}a(4)

4.2.1.2 Fuel Design Basis

The Principal Design Criteria (PDC) used to develop design criteria are provided in the approved TRs referenced in Section 3.6.1. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC were identified as applicable to the fuel:

- **PDC-10**, “Reactor design”
- **PDC-16**, “Containment design”

Additionally, the ceramic TRISO fuel is based on the TRISO fuel that has been successfully developed and tested by the U.S. Department of Energy (DOE) Advanced Gas Reactor (AGR) Fuel Development and Qualification Program. The TRISO specifications are based on those developed and tested in the AGR program. Comprehensive irradiation and safety testing campaigns were performed under the first (AGR-1) and second (AGR-2) irradiation experiments of the AGR program (Reference 4-3). The TRISO specifications used are within those developed and tested in the AGR program. Comparison between the used TRISO fuel specification and the AGR-1 and AGR-2 TRISO fuel properties is provided in Section 2.2.2 of (Reference 4-2). The operating conditions used in the U. of I. research reactor are encompassed by the testing conditions of the AGR testing campaign for those specifications.

The performance of the AGR-1 and AGR-2 UCO TRISO fuel is reported in the “*Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance*” Topical Report (Reference 4-4), submitted by the Electric Power Research Institute (EPRI) to the U.S. Nuclear Regulatory Commission (NRC) in 2020 for formal review and issuance of a Safety Evaluation Report (SER) (Reference 4-1).

The EPRI Topical Report provides the technical bases (e.g., particle design, irradiation, and accident testing results) that demonstrate the functional performance of UCO TRISO particles.[1]

1. The AGR-2 irradiation test also included UO_2 fuel, but only UCO fuel performance is addressed in the EPRI Topical Report.

The SER concluded that there is reasonable assurance that TRISO particles produced to the specifications and limited to the performance parameters documented in the EPRI Topical Report will satisfy a portion of the requirements associated with PDC-10, subject to the "Limitations and Conditions" in Section 4.0 of the SER. More specifically, TRISO particles produced to the specifications within the report and limited to the performance parameters in the report will perform in accordance with the AGR data presented in Sections 6 and 7 of the report. Therefore, the data can be used to support safety analyses referencing the unique design features of TRISO particles.

Consequently, adopting a UCO TRISO design similar to the fuel tested by the AGR-1 and AGR-2 irradiation experiments allows the ceramic TRISO fuel to be covered by the conclusions drawn by the SER and by the qualification envelope established by the AGR-1 and AGR-2 irradiation experiments (Section 4.2.1.4).

Since the ceramic fuel matrix has a limited operational experience regarding its irradiation performance, it will be tested through an extensive fuel qualification program. Nevertheless, the EPRI Topical Report sets a pathway for adequate performance of TRISO fuel particles manufactured and tested under specific conditions, which establishes a strong basis for the fuel qualification program. The limitations and conditions identified by the NRC on the applicability of the EPRI Topical Report will be satisfied by the fuel qualification program, as described in Section 1.5.5 of (Reference 4-2) and summarized in Section 4.2.1.7 of this report.

4.2.1.3 Functional Containment

A functional containment is a set of barriers that limit the physical transport and release of radionuclides to the environment across a full range of normal operating and postulated event conditions, ensuring off-site dose limits are not exceeded.

For the U. of I. research reactor, the functional containment is ensured by both the TRISO particle and ODSL, which combine to provide five barriers to fission product release, which demonstrates compliance with PDC-16. This approach provides defense-in-depth to functional containment. As such, the fuel is designed and engineered to demonstrate low defect fractions during manufacturing (Section 4.2.1.5) and limited in-service failure fractions (Section 4.2.1.6).

The UCO kernel retains a large fraction of fission products and constitutes, de facto, the first barrier. The three outer coating layers (IPyC, SiC, and OPyC) provide additional retention. In particular, the SiC layer acts both as main load-bearing structural layer against internal fission gas pressure, which limits in-service failure by over-pressure, and as main retention barrier against metallic fission product release. It is sandwiched between the two dense IPyC and OPyC layers that help maintain the mechanical integrity of the TRISO particle and provide effective barriers to fission gas release.

The fifth barrier is the SiC ODSL. It retains fission products released by TRISO particles into the SiC matrix in which they are embedded. Its thickness is designed to keep fission product release below regulatory limits over the lifetime of the fuel.

By virtue of being immediately adjacent to the helium coolant, the SiC ODSL operates at a lower temperature than the TRISO SiC layer, and as a result radionuclides diffusion through the ODSL will be slower. Thus, the TRISO layers and the SiC ODSL constitute two separate, effective systems that contribute to functional containment. The effectiveness of the TRISO SiC layer as a barrier to fission product release has been demonstrated through extensive testing as described in Section 2.1 of (Reference 4-2). Based on the larger thickness and lower operating temperature of the ODSL, it is expected that it will perform as well or better than the TRISO SiC layer.

Finally, the graphite blocks, the helium pressure boundary in the primary circuit, and the Citadel building are not credited as fission product barriers. The overall functional containment approach is summarized in Section 6.2.

4.2.1.4 Fuel Operating Envelope

Preliminary fuel operating conditions were established for the reactor fueled with LEU+ and operating at 10 (i.e. ~22% of the nominal power) and 45 MWth power, to encompass the range of power conditions anticipated for normal operations. The normal operating conditions include core average and/or peak values for burnup, fast neutron fluence, and temperature, as well as pellet power and TRISO particle power corresponding to the reactor core power. These preliminary results are provided in Table 2.4 of (Reference 4-2); these bound the updated analysis of core performance described in Section 4.5 of this report.

Additionally, Table 2.5 and Table 2.6 of (Reference 4-2) compare the reactor normal operating envelope to AGR-1 and AGR-2 irradiation test parameters. Preliminary assessment of postulated event conditions was also performed, as reported in Section 2.4.2 of (Reference 4-2); this is superseded by the results discussed in Chapter 13 of this report. Neither the normal operating conditions nor the postulated event conditions experienced by fuel pellets in the reactor core challenge the temperature limits of TRISO particles (e.g., AGR-1 and AGR-2 peak temperatures) or of the SiC matrix (e.g., onset of SiC decomposition) and are well below the melting point of the UCO kernel.

4.2.1.5 Fuel Manufacturing and Quality Control

TRISO fuel particles are produced and inspected using fabrication equipment, fabrication procedures, and quality control procedures that closely follow those developed by the AGR program (Reference 4-3). The fabrication process for encapsulation of TRISO particles in SiC to form ceramic-matrix fuel pellets was developed by the DOE Transformational Challenge Reactor (TCR) program (Reference 4-5).

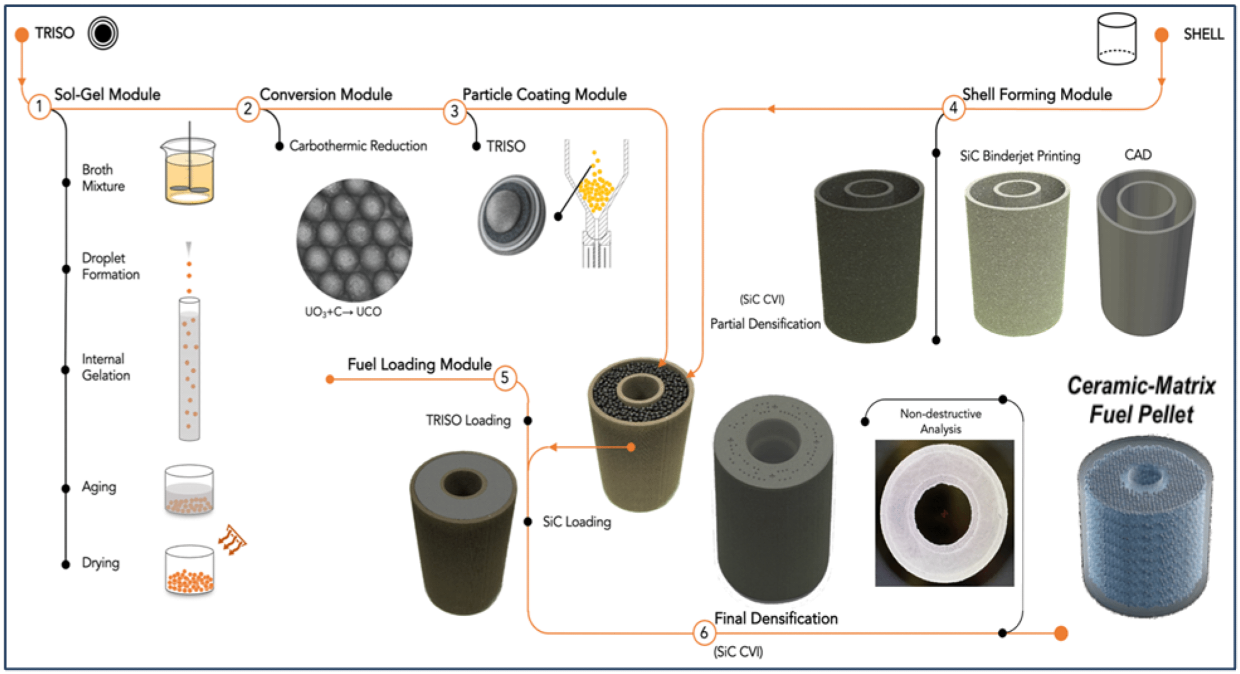
Figure 4-5 provides an overview of the fuel manufacturing process. The left side shows the process for TRISO particle manufacturing. The right side shows the ceramic-matrix pellet manufacturing process, including incorporation of TRISO particles into the pellet.

TRISO particles produced by the AGR program exhibited excellent performance during irradiation and safety testing (Reference 4-4). Ranges of process variables used to manufacture the TRISO fuel are typical of those used to manufacture AGR TRISO particles within specification limits.

TRISO fuel particles are manufactured using a sequence of batch processes. Each of the three major TRISO manufacturing processes (steps 1 – 3) on the left side of Figure 4-5 are conducted within manufacturing modules specifically designed for each process step. The manufacturing process is scaled from pilot scale to commercial scale by adding additional manufacturing modules, operating in parallel.

This section provides an overview of the fuel manufacturing and associated quality control. Additional details are provided in Section 3.1, Section 3.2, Section 3.3, and Section 3.4 of (Reference 4-2).

Figure 4-5 Flow Diagram of Fuel Manufacturing Process



4.2.1.5.1 Manufacturing of TRISO Particles

This process, as implemented by the AGR program, consists of three primary steps which are illustrated as steps 1 – 3 in Figure 4-5:

- Manufacturing of UCO precursor gel spheres using the Sol-Gel module
- Conversion of gel spheres to high-density UCO kernels using the Conversion module
- Coating of UCO kernels with buffer, dense PyC, and SiC layers using the Particle Coating module

Each step in the TRISO manufacturing process includes sampling and quality control measurements (Section 4.2.1.5.3) to ensure that the manufactured TRISO fuel particles meet all product specifications.

As noted in Section 4.2.1.1.1, additional proprietary functional features were added to the typical TRISO design to improve overall irradiation performance. Their manufacturing is described in Section 3.1 of (Reference 4-2).

4.2.1.5.2 Manufacturing of Ceramic-Matrix Fuel Pellets

The ceramic-matrix fuel form is an annular cylindrical pellet composed of TRISO particles encapsulated in a SiC matrix. The SiC pellet is coated with a SiC ODSL designed to retain fission products over the lifetime of the fuel in the reactor core. The fuel manufacturing process is designed to prevent damage to

TRISO particles. The mechanical compaction process used to manufacture carbon-based TRISO compacts is replaced with a three-step fabrication process, illustrated as steps 4 – 6 in Figure 4-5:

- Fabrication of a partially densified SiC shell pre-form using additive manufacturing followed by a partial chemical vapor infiltration (CVI) step
- Loading of the pre-form with TRISO particles and SiC matrix powder
- Final densification of the fuel pellet using CVI

4.2.1.5.3 Quality Control

Intermediate and final fuel products from each batch of TRISO particles are sampled to provide statistical information used to show adherence to the specification. Batches are combined into fuel lots and statistical quality control information is used to represent the properties of each lot.

Sampling methods for large-scale production (> 10^9 particles per core) were developed based on historical experience with sampling methods used for large-scale production of TRISO fuel in combination with modern industrial standards and methods used by the AGR program to calculate confidence intervals. Details about statistical sampling methods are provided in Section 3.3.1 of (Reference 4-2).

TRISO particle quality control procedures were adopted from the AGR program, which refined and formalized these procedures over more than 20 years, building on German experience in large-scale TRISO fuel manufacturing. The TRISO fuel product specification focuses on the key TRISO particle properties (e.g., kernel chemistry, coating layer thickness and density, manufacturing defect fractions) that affect fuel performance. TRISO particle quality control procedures focus on achieving excellent and repeatable fuel performance through adherence to the TRISO fuel product specification. A complete list of key properties of TRISO particles subject to quality control procedures is provided in Section 3.3.2 of (Reference 4-2).

Quality control processes for the ceramic-matrix fuel pellets are currently being developed to ensure the in-service integrity of the pellet and to allow the pellet ODSL to be credited as a retention barrier to fission product release. Critical attributes of the fuel pellets inspected and controlled by quality procedures are:

- Fuel pellet dimensions
- Thickness of the ODSL
- Leak-tightness of the ODSL
- Mechanical integrity of the ODSL
- Uranium loading

Additional details regarding quality controls and inspections will be provided in the Operating License Application (OLA).

4.2.1.5.4 Ceramic-Matrix Fuel Manufacturing Acceptance Criteria

As noted in Section 6.7 of (Reference 4-2), the acceptance criteria for the ceramic-matrix fuel are:

- Defect fractions of the as-fabricated ceramic-matrix TRISO fuel meet limits specified in Table 2.2 of (Reference 4-2)
- The thickness of the as-fabricated ODSL meets limit in Table 2.3 of (Reference 4-2)
- Fuel pellets have zero defective ODSL (i.e., 100% of fuel pellets pass the hermeticity test)

4.2.1.6 Fuel Performance

This section provides an overview of the performance of the TRISO particles and ceramic-matrix fuel pellets, identified failure mechanisms, fission product transport, and fuel performance modeling.

4.2.1.6.1 Fuel Performance Overview

The nuclear fuel form is an annular cylindrical pellet composed of TRISO particles held in a SiC matrix and encapsulated in a high density SiC shell. The fuel is designed and fabricated such that the TRISO particles and the ODSL component of the fuel pellet act as complementary barriers to fission product release.

The performance of TRISO particles with similar characteristics as the TRISO fuel used for the U. of I. research reactor has been extensively tested and shown to provide effective retention of fission products at temperatures (> 1600°C) and times (300 hours) that exceed postulated high-temperature gas-cooled reactor (HTGR) accident conditions (Reference 4-4). The testing and in-pile experience was obtained with TRISO particles embedded in carbon-based matrices. It follows that qualification of the fuel design requires verifying (1) the performance of TRISO particles embedded in the ceramic-matrix fuel pellet, (2) that the geometry of the fuel pellet is maintained during operation, and (3) that the fuel pellet provides an additional barrier to fission product release.

Under neutron irradiation, as the fuel burnup in a TRISO particle increases, the kernel swells outward under the influence of solid and gaseous fission products. Conversely, the buffer retracts inward away from the IPyC layer as it densifies and shrinks. During the process, the kernel and buffer stay bonded, while the buffer tends to detach from the IPyC, creating a gap between the two layers. The buffer-IPyC gap is the largest thermal resistance in the TRISO particle and largely determines the kernel temperature. As irradiation proceeds, the buffer layer is pushed outward by the swelling kernel and the width of the buffer-IPyC gap is determined by the balance between kernel expansion and buffer shrinkage.

The IPyC and OPyC layers exhibit shrinkage early in irradiation followed by swelling as fast neutron fluence increases. The reversal of the strain depends on the density and degree of anisotropy of the PyC layer (controlled by specification) and on the irradiation temperature. Early in irradiation, PyC shrinkage creates tensile stress in the IPyC and OPyC layers, which imparts overall compressive stress onto the more rigid SiC layer. Cracking of the PyC layers can occur if the tensile stress that results from shrinkage overcomes the fracture strength of the PyC layers. This can result in localized tensile stress on the SiC layer and potentially lead to SiC layer failure.

As irradiation progresses, irradiation-induced creep in the PyC layers offsets their shrinkage and relieves some of their tensile stress. Simultaneously, fission gas pressure builds up in the void volume of the TRISO particle internal to the IPyC layer. As internal pressure increases with burnup, the tangential stress in the SiC layer changes from compressive to tensile. This can potentially lead to the mechanical failure of the SiC layer if the tensile stress reaches a value that exceeds the strength of the TRISO SiC layer. The irradiation behavior of the outer coating layers is shown schematically in Figure 4.1 and summarized in Table 4.1 of (Reference 4-2). An illustrative representation of the corresponding stress in the IPyC and SiC layers as a function of fast neutron fluence is shown in Figure 4.2 of (Reference 4-2).

The performance of the fuel pellet during normal operation or under transient scenarios is mainly driven by stress that develops under the influence of irradiation-induced dimensional changes (i.e., swelling) and thermal gradients at high power. The pellet thermal gradient is a function of pellet power density, pellet surface temperature, and pellet matrix thermal conductivity. In the annular fuel pellet, the designed balance in cooling between the inner and outer surfaces minimizes temperature gradients and subsequent thermal stresses. The swelling of SiC increases logarithmically with fast neutron dose until saturation. Saturation occurs at relatively low neutron dose, at which point swelling stops. SiC swelling is also dependent on temperature, exhibiting higher swelling at saturation during irradiation at lower temperatures.

During irradiation, fission heat generation from the embedded TRISO particles creates a thermal gradient between the inner and outer surfaces of the fuel pellet. The stress in the ODSL results from the balance between the differential thermal expansion and swelling rate of the ODSL and matrix. As previously discussed, irradiation-induced swelling in SiC is a function of temperature and fast neutron fluence. As fast neutron fluence accumulates, the volume of the lower temperature near-surface region of the fuel pellet increases at a faster rate relative to the higher temperature of its interior region. The increased low temperature swelling in the inner and outer regions acts to relieve the tensile stress near the pellet surface. As differential swelling proceeds, the stress state of the inner and outer regions transitions from tensile stress to compressive stress. An illustration of this behavior is shown in Figure 4.3 of (Reference 4-2).

4.2.1.6.2 Fuel Failure Mechanisms

Based on historical irradiation experience, the following failure mechanisms of TRISO-coated fuel particles have been identified (Reference 4-4):

- Pressure vessel failure of spherical or aspherical particles resulting in the failure of all three coating layers
- Irradiation-induced cracking of the IPyC layer leading to SiC failure
- Irradiation-induced partial debonding of the IPyC from the SiC leading to SiC failure
- Irradiation-induced cracking of the OPyC layer leading to SiC failure
- Kernel migration towards the SiC layer and its subsequent failure
- Chemical attack of the SiC layer by fission products (e.g., noble metals or lanthanides) or CO leading to its failure
- Thermal decomposition of the SiC layer at high temperatures
- Irradiation-induced buffer fracture leading to cracking of partially or fully attached IPyC

Two types of failure can result from these failure mechanisms: SiC failure and TRISO failure; SiC failure is defined as the loss of fission product retention capability by the SiC layer, but it leaves at least one of the PyC layers intact; TRISO failure, also referred to as "full failure", corresponds to the loss of leak-tightness of all three outer coating layers (i.e., IPyC, SiC, and OPyC), opening a direct pathway to fission product release from the kernel into the SiC matrix.

Additionally, potential OPyC cracking due to interaction with the surrounding matrix is a failure mechanism specific to ceramic-matrix TRISO fuel.

The potential failure mechanisms specific to irradiation of the ceramic-matrix pellet include:

- ODSL degradation caused by reactor operation
- Fuel pellet fracture
- Chemical attack of the ODSL
- Corrosion of the ODSL by impurities in the reactor coolant
- Thermal decomposition of the ODSL or SiC matrix at high temperatures

A fuel pellet failure is defined as the loss of leak-tightness to fission product release by the ODSL. These failure mechanisms are discussed in greater details in Section 4.2 and Section 4.4 of (Reference 4-2).

4.2.1.6.3 Fission Product Transport and Release in the Fuel

The fuel uses SiC in two different forms to prevent fission product release from the fuel pellet: the TRISO SiC layer acts as the primary barrier to fission product release from TRISO particles; the ODSL provides an additional barrier on each fuel pellet to retain fission products released by TRISO particles into the SiC matrix in which they are embedded. This approach provides defense-in-depth to functional containment.

A potential source of fission product release to the helium coolant is uranium contamination, referred to as "tramp" or "dispersed" uranium, outside of the intact TRISO SiC layer. Because of the very high integrity of TRISO particles, fission products from dispersed uranium are expected to be a substantial portion of the radioactivity released from TRISO fuel early during irradiation. Limits are prescribed in the TRISO particle specification to control release by this mechanism, but the SiC ODSL of the fuel pellet should substantially reduce this source.

Another potential mechanism for fission product release to the helium coolant is diffusion of specific radioisotopes from the fuel kernel through the intact TRISO particle coating layers. The transport of mobile fission products through a TRISO particle is a complex process that depends on the microstructure of the kernel and coating layers. It can involve several mechanisms such as lattice diffusion, grain boundary diffusion, pore diffusion, nano-cracking, and vapor transport. These potential transport mechanisms can also be impacted by effects such as irradiation-induced trapping and adsorption, thermal decomposition of the coating layers, or chemical attack of the coating layers by noble metals or rare-earth elements.

The transport of gases and metals in the kernel and coating layers is likely driven by different basic mechanisms. However, the fundamental knowledge of all transport phenomena is limited, and fission product transport in TRISO fuel is modeled by Fickian diffusion using effective diffusion coefficients. The effective diffusivities were derived by fitting experimental data that involve these transport phenomena, which makes them adequate to model transport. The phenomenon of diffusion is dependent on time at temperature. The slow rate of diffusion through the ODSL results in significant fission product decay and in subsequent reduced radiological release.

Additional details about fission product transport in the fuel are provided in Section 4.5 of (Reference 4-2).

4.2.1.6.4 Fuel Performance Modeling

Fuel performance modeling is used to:

- Predict fuel performance during reactor normal operating and transient conditions to support fuel design, fuel fabrication, fuel optimization, and safety evaluation
- Predict fuel performance in material test reactors (MTRs) for the design of irradiation experiments that support fuel qualification
- Assist post-irradiation analysis of fuel behavior in MTRs in support of fuel qualification

The two figures of merit for evaluation of the fuel performance are in-service probability of failure (i.e., loss of leak-tightness to fission product release) and fission product fractional release (i.e., the ratio between the amounts of fission products released from the TRISO particles or the ceramic-matrix pellets and produced in the TRISO fuel kernels and dispersed uranium).

For fuel qualification purposes (Section 4.2.1.7), analysis of the fuel performance will be conducted through a combination of modeling, using validated and approved fuel performance modeling codes, and reliance on testing data. Details are provided in Section 5.1 and Section 5.2 of (Reference 4-2), respectively.

4.2.1.7 Fuel Qualification

The fuel qualification program provides reasonable assurance that the fuel design can operate with a low failure rate and a level of fission product release consistent with the design basis analysis. The fuel qualification methodology of the fuel is based on international and U.S. operating experience with TRISO fuel particles, including the extensive irradiation testing and post-irradiation safety testing of TRISO fuel particles by the DOE AGR Fuel Development and Qualification Program.

The methodology for fuel qualification, as it relates to in-reactor performance of fuel pellets in the reactor core, is described in the FQMTR (Reference 4-2). Its associated results will be provided with the OLA. Similarly, all other aspects related to the qualification of the ceramic-matrix TRISO fuel, that are not within the scope of the FQMTR, will be provided in future regulatory reports, as required prior to receipt of the Operating License.

Activities related to the fuel qualification program, which are discussed in (Reference 4-2), include:

- Development of fuel product specifications for TRISO particles and ceramic-matrix pellets (Section 2.2.2 and Section 2.2.3 of (Reference 4-2)),
- Demonstration of fuel manufacturing and quality control processes capable of consistently meeting specifications (Section 3 of (Reference 4-2))
- Testing and characterization of unirradiated fuel and materials (Section 6.1 of (Reference 4-2))

- Fuel pellet irradiation tests in material test reactors (Section 6.2 of (Reference 4-2))
- High-temperature safety testing of irradiated fuel pellets to measure performance in simulated accident conditions (Section 6.3 of (Reference 4-2))
- Post-irradiation examination of fuel pellets after irradiation testing and high-temperature safety testing to determine fuel performance of irradiated fuel pellets (Section 6.4 of (Reference 4-2))
- Fuel performance modeling calculations in support of fuel qualification (Section 6.5 of (Reference 4-2))

A fuel surveillance program will be implemented at the U. of I. research reactor per ANSI/ANS-15.1, Sections 3.3(5) and 3.7.1(2) (Reference 4-6). The fuel surveillance program will perform online monitoring of the reactor coolant for abnormal fission product activity during startup, startup testing initial operations, and full power equilibrium operation to ensure that fission product release remains within operational limits. In particular, online monitoring of fission gas release from the fuel will serve as an indicator of TRISO particle and ceramic-matrix pellet failure. Anomalous fuel performance resulting in excessive fission gas release will be readily detectable and actionable to ensure no undue risk to public health and safety.

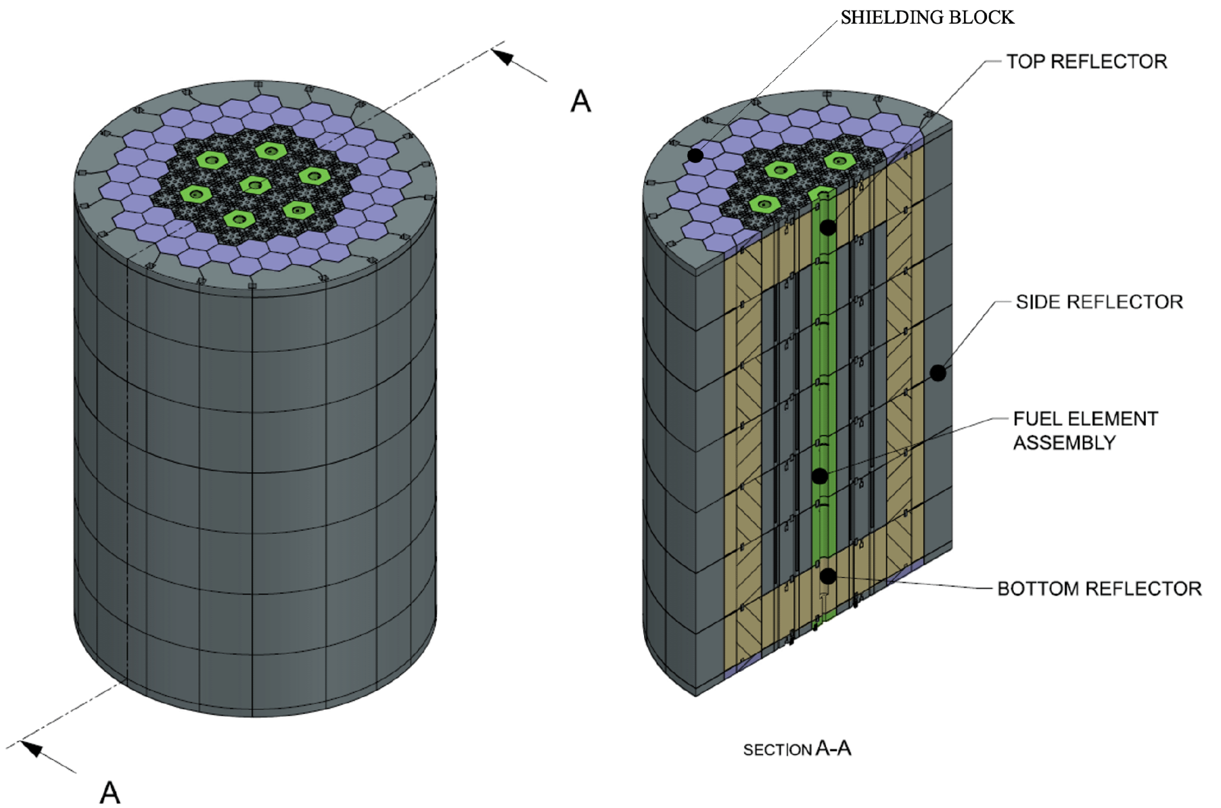
The successful testing of the fuel through the fuel qualification program will be measured against fuel acceptance criteria that are listed in Section 6.7 of (Reference 4-2).

4.2.2 Reactor Core System

4.2.2.1 Description

The reactor core is comprised primarily of prismatic graphite components that are stacked into free-standing columns. The fuel pellets and burnable absorbers are contained within the graphite fuel elements. In addition to the fuel elements, the Reactor Core System contains control block elements, reflector elements, shield blocks, and neutron sources. Figure 4-6 shows the Reactor Core System.

Figure 4-6 Reactor Core System Overview



The active core, shown in Figure 4-7, is the central region of the reactor core, and contains all of the fuel within the reactor. It consists of 30 columns of fuel elements and 7 columns of control block elements, as shown in the core plan view, Figure 4-8. Each of these columns are comprised of five layers of either fuel elements or control block elements. Each column of control block elements contains a control rod channel that spans the length of the active core and permits the insertion of movable control rods for both reactivity control and shutdown. The fuel element columns and control block element columns are arranged in a 1/6th symmetric hexagonal lattice. The fuel elements and control block elements are detailed in Section 4.2.2.1.1 and Section 4.2.2.1.2, respectively.

Figure 4-7 Active Core Overview

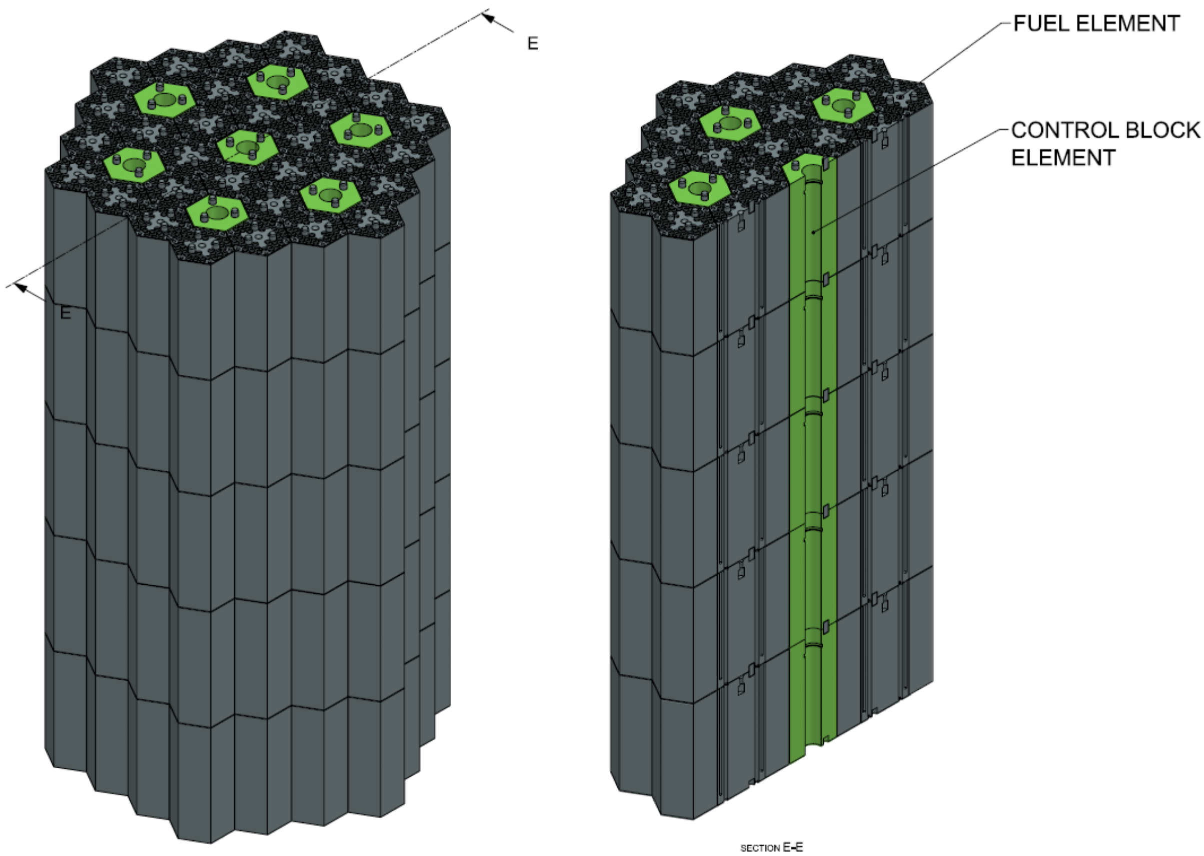
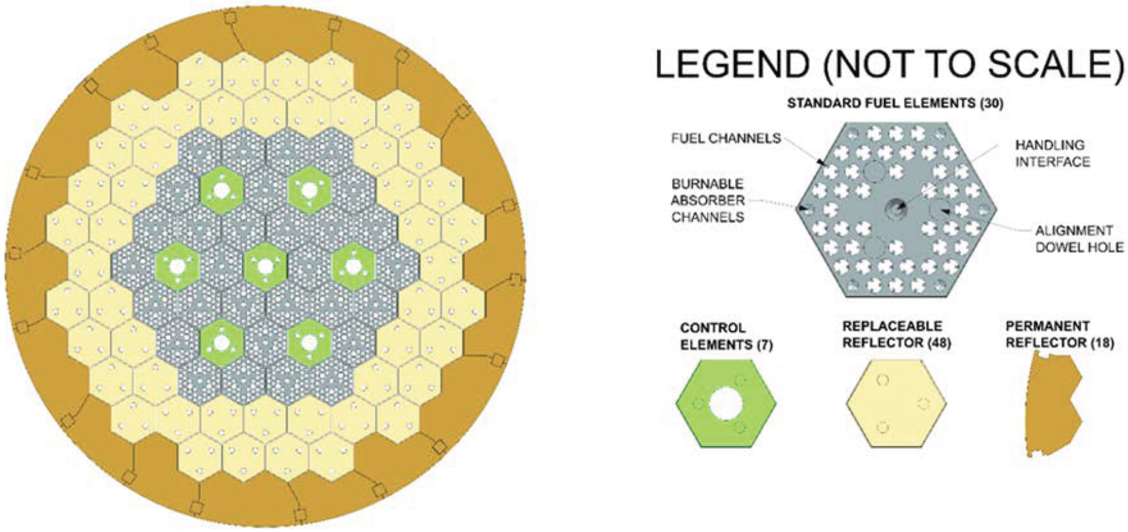


Figure 4-8 Reactor Core Plan View



The top of each fuel element column and control element column consists of an axial reflector element, with a shield block at the very top. The bottom of each column also consists of an axial reflector element, with a shield block at the very bottom. The axial reflector and shield blocks are detailed in Section 4.2.2.1.3 and Section 4.2.2.1.4, respectively.

The active core is surrounded radially by columns of stacked reflector elements with an axial shield block at both the bottom and the top of each column. The outermost reflector ring contains non-hexagonal components, which transform the external profile of the reactor core into a cylinder. The radial reflector elements are detailed in Section 4.2.2.1.3. The core radial reflector is surrounded and restrained by the core lateral restraint structure, described in Section 4.2.5.

At each element-to-element vertical interface in the hexagonal columns, three dowel-socket connectors provide for alignment of blocks and channels, and they also transfer seismic loads. The columns of stacked components are separated by a nominal gap to account for tolerance build ups and differential thermal and irradiation induced expansion. {{

}}$^{a(4)}$

4.2.2.1.1 Fuel Elements

The fuel elements are comprised of prismatic graphite blocks containing channels loaded with fuel pellets and fixed burnable absorbers. Each fuel element contains a handling interface accessible from the top to enable lifting by the Fuel Handling and Storage System (Chapter 9.2). Each fuel element is chamfered at the top and bottom to accommodate refueling, reducing both frictional forces and the likelihood of the edge chipping.

Fuel channels run parallel through the length of the prism with spacing that ensures {{

}}a(4) integrity of the block. The standard fuel element, shown in Figure 4-9, contains channels for the fuel pellets and blind channels (closed at the bottom) for the fixed burnable absorbers, described further in Section 4.5. The fuel pellets are assembled into stacks, which when inserted into the fuel channels, create a flow path on both the inside and outside of the annular pellet stack. Properties of the fuel element are summarized in Table 4-3.

The fuel pellet stacks are vertically supported and laterally located at the bottom of the fuel channels by fixed graphite support fins that allow for uninhibited coolant flow. {{

}}a(4) Features on the top and bottom of each fuel pellet ensure vertical alignment of the pellet stack and prevent relative movement during seismic events or from flow-induced vibrations. {{

}}a(4)

Figure 4-9 Fuel Element

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}}a(4)

Table 4-3 Fuel Element Properties

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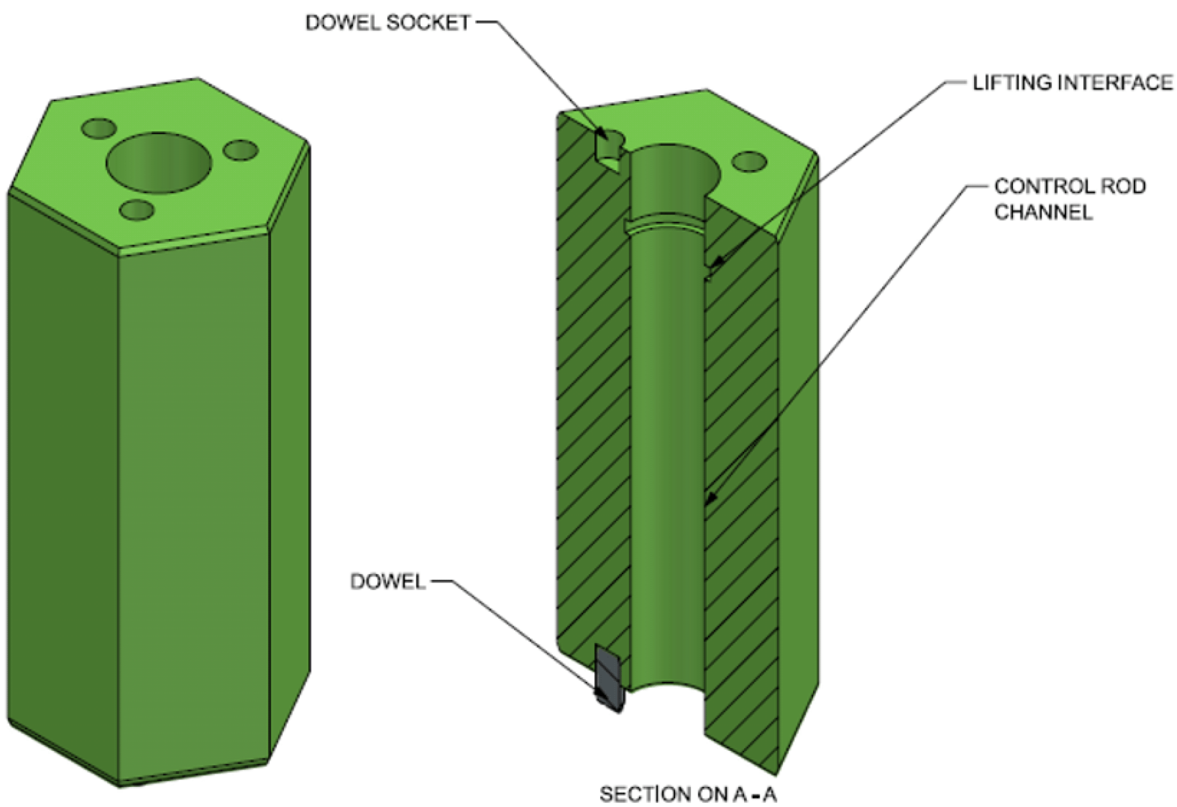
}}a(4)

4.2.2.1.2 Control Block Elements

Each control block element is a hexagonal graphite block containing a single central channel to allow for control rod insertion. A small chamfer at the top face of the channel ensures that insertion of the control rods does not chip the entrance to the channel. Each control block element contains a handling interface inside the channel which is comprised of a machined ledge. The external edges on the top and bottom block faces are chamfered to accommodate refueling. The standard control block element is displayed in Figure 4-10.

Figure 4-10 Control Block Element



4.2.2.1.3 Reflector Elements

The active core is surrounded axially and radially by reflector elements, which conserve neutrons, enhance neutron moderation to the active core, and limit fast neutron fluence to the reactor vessel and reactor internals. The reflector elements are separated into two distinct categories: the hexagonal, prismatic reflector elements, which are above, below and radially surrounding the active core; and the non-hexagonal reflector elements, located radially at the periphery of the core and adjacent to the core barrel.

The reflector elements, which are above and below the active core, are hexagonal prismatic blocks. The reflector elements in line with the fuel element columns contain coolant channels that are aligned with those in the fuel elements. The reflector elements above the control columns contain control rod channels aligned with those in the control elements. In the reflector elements below the control columns, the control rod channel extends only partially into the top of the block; the lower portion of the block contains a reduced-diameter channel to prevent over-insertion of the rods while still allowing helium flow.

The active core is radially surrounded by 48 hexagonal reflector columns, each consisting of seven stacked reflector elements with dowel-socket connections for alignment. Each reflector element includes a lifting interface and chamfered edges at the top and bottom to facilitate periodic replacement throughout the life of the reactor. The replacement frequency of the hexagonal radial reflector elements, as required by structural or dose management considerations, will be included in the OLA.

The 18 reflector columns at the radial periphery of the core have an irregular profile that conforms to the hexagonal radial reflector elements on one side, and to the cylindrical core barrel on the other side. The columns are keyed together to form the outer ring layer, which retains the overall core geometry under all anticipated conditions and minimizes radial bypass flow. {{ }}a(4) Each column consists of seven stacked reflector elements. These non-hexagonal radial reflector elements are designed to remain in the core for the entire life of the reactor.

4.2.2.1.4 Shield Blocks

The axial shield blocks are located at the top and bottom of each column in the reactor core, as shown in Figure 4-6. Each is {{ }}a(4) boron carbide (B_4C). These shield blocks are sized to provide sufficient shielding material to limit neutron damage and activation of the adjacent structures. Each shield block contains a lifting interface that allows for handling during installation, refueling, maintenance and decommissioning. The shield blocks above and below the fuel element columns contain channels that are aligned with the fuel channels in the fuel elements. The shield blocks above each control rod column include a channel to enable control rod insertion, while the shield blocks below control rod channels include a reduced-diameter channel to allow coolant exit flow. The shield blocks above and below the radial reflector columns do not contain any channels.

4.2.2.1.5 Graphite Properties and Performance

High-temperature irradiation of the graphite components of the reactor core causes internal strain via differential irradiation-induced dimensional change of graphite, exclusively shrinkage at the low fluences experienced. Additional effects include increases in the coefficient of thermal expansion, Young's modulus and strength, a decrease in conductivity, and creep. At the beginning-of-life, thermal strains due to the small temperature gradients that exist between the fuel channels and side walls of the blocks dominate the stress field in the components. As the material is irradiated, irradiation-induced creep occurs, which reduces the stress caused by these gradients by deforming the material. In addition, the dimensional change and the change in mechanical properties are dependent on the total fluence and irradiation temperature, and therefore have varying magnitudes throughout the component, which causes additional internal strains. The highest stresses in the reactor core are expected to be at full power operation at the end-of-life (EOL) or during cold conditions at EOL. A thorough stress analysis will be performed to demonstrate that the graphite components will not degrade under the expected conditions.

The graphite components of the reactor core are in direct physical contact with helium. Helium is continuously purified and monitored throughout the life of the reactor. Significant thermal oxidation during normal conditions is therefore unlikely. In addition, with water ingress events essentially ruled out by the secondary molten salt loop, it means that aside from large scale air ingress events, oxidation of graphite components is limited to the oxygen trace impurities that remain after the HPS (Chapter 5.4) has been run.

The graphite components of the reactor core are fabricated from nuclear-grade graphite, as defined and standardized by ASTM D7219 (Reference 4-7). The characteristics of the graphite used are guaranteed to exceed the specifications of ASTM D7219. The quality program from graphite manufacturers currently under commercial consideration, and the acceptance tests to be performed on the graphite upon delivery, will ensure that those characteristics are reliably met. The design basis for the graphite in the core uses the most conservative values from the property ranges specified in the standard, including strength, dynamic elastic modulus, and thermal conductivity, also correcting for irradiation driven phenomena. The specific graphite grade is vendor-dependent and will be selected as a result of the on-going procurement efforts. Currently, {{ }}$^{a(4)}$ specific grades have been down-selected {{ }}$^{a(4)}$ All {{ }}$^{a(4)}$ of these graphite grades readily satisfy ASTM D7219 requirements, and benefit from extensive irradiation data that was collected under stringent quality programs and far exceeding the range of conditions relevant to the research reactor. Detailed characteristics of the grade of graphite that ends up being selected, evidencing state of qualification for the graphite, will be the subject of a topical report ahead of the OLA.

The reactor graphite components are designed as class SN nonmetallic core components in accordance with ASME Section III Div 5 (Reference 4-8), and a compliant graphite qualification program is being developed to fully characterize the properties and performance of the specific selected grade under the operating conditions in the core, including during normal operations and postulated events. To account for any shortfall in experimental data, a combination of data from similar grades, theoretical modeling and increased design conservatism shall be utilized. Required as-manufactured properties include tensile and compressive strength, dynamic and static elastic modulus, Poisson's ratio, coefficient of thermal expansion, density, thermal conductivity, and mean grain size. The required irradiated properties include dimensional change, creep coefficients, tensile strength, elastic modulus, coefficient of thermal expansion, and thermal conductivity. Required oxidized properties include tensile strength, dynamic elastic modulus, and thermal conductivity. For the properties which are dependent on temperature, fluence, and oxidative weight loss, data will be obtained at intervals over the relevant ranges for the research reactor. The peak fast neutron fluence experienced by any of the graphite components in the core over their residence lifetime is well below the shrinkage reversal point of the considered graphite grades, alleviating the need for extensive irradiation data at high fast fluence levels.

4.2.2.2 Functions, Classification and Design Criteria

4.2.2.2.1 Safety Functions

The safety functions of the Reactor Core System are to retain radionuclides within the fuel, maintain geometry to ensure control of reactivity, and facilitate passive removal of decay heat.

4.2.2.2.2 Non-Safety Functions

The non-safety functions of the Reactor Core System are to produce reactor power and to maintain the capabilities to refuel and transition between all operating modes.

4.2.2.2.3 System Classification

The Reactor Core System is safety related, and therefore Seismic Category I.

4.2.2.3 Design Basis

The PDC used to develop design criteria are provided in the approved TRs referenced in Section 3.6.1. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC were identified as applicable to the Reactor Core System:

- **PDC-10,** “Reactor design”
- **PDC-28,** “Reactivity limits”
- **PDC-70(a),** “Citadel design basis”
- **PDC-70(b),** “Reactor vessel and reactor system structural design basis”

4.2.2.4 System Evaluation

The graphite and metallic components that form the reactor core are designed and constructed to meet the intent of ASME Section III Div 5 (Reference 4-8), which is a generally recognized code and endorsed by the US NRC in RG 1.87 (Reference 4-9). The quality assurance program (described in Section 12.9) incorporates the relevant requirements and controls from subsection HAA (metallic) and HAB (graphite). This demonstrates compliance with PDC-1.

The reactor core is protected from natural phenomena by the Citadel building and the reactor vessel. The reactor vessel and core barrel maintain the cylindrical outer geometry of the reactor core. In addition, the minimal gap sizes between the columns of stacked graphite components limits their relative movement, as well as the loading induced during earthquakes and other postulated events. These features demonstrate compliance with PDC-2.

The reactor core is protected from dynamic effects, such as missiles, pipe whipping and discharging fluids, by the Citadel building and the reactor vessel. The reactor core is vertically supported and laterally restrained by the core barrel, which maintains the reactor core's external cylindrical geometry. The stacked columns are axially connected by dowels and are laterally spaced apart to limit loading due to assembly errors, tolerance build-ups and differential thermal and irradiation induced dimensional change. The outer ring of non-hexagonal radial reflectors is interlocked using keys, limiting space for relative movement of the interior hexagonal columns. This configuration ensures that lateral external loadings, such as those from seismic events or pressure differentials, are transferred to the core barrel through the components via compression rather than shear loading of the dowels. Both the graphite and stainless steel used for the core components are qualified for the high temperatures, thermal stresses, neutron fluence and the coolant chemistry they will be exposed to during operation. These features demonstrate compliance with PDC-4.

Large margins to failure are utilized for all safety-related components. Each graphite component is designed to conform to a conservative structural design code. The graphite fuel elements are designed with sufficient clearance to accommodate irradiation induced swelling of the fuel pellets and shrinkage of the graphite in both the axial and radial dimensions. The reactor is designed to limit the peak neutron fast fluence to less than {{ }}$^{a(4)}$ for any graphite component. This limit is informed by the irradiation data that is available for the graphite grades being evaluated for the core components. Additionally, ASME Section III Div 5 Subsection HHA (Reference 4-8) provides a limit of +10% dimensional change, which occurs at fluence levels significantly higher than the adopted limit. These features demonstrate compliance with PDC-10.

The core is designed with a high thermal inertia, which prevents sudden reactivity insertion due to overcooling. The limited space for displacement resulting from the hexagonal shape and tight configuration (lack of available space) of the core components limits the total positive reactivity insertion that could occur from compaction. The burnable absorbers are manufactured from a high temperature ceramic and are installed within the graphite components, so there can be no reactivity insertion due to a change in position or damage to the fixed burnable absorbers. These features demonstrate compliance with PDC-28.

The reactor core provides a passive heat transfer pathway from the fuel to the core barrel during postulated events. The external cylindrical geometry of the reactor core is maintained by the core barrel. The core is compact and comprised of graphite which has high thermal conductivity and thermal inertia. In events where forced cooling is lost or the system is depressurized, the graphite transfers heat from the fuel pellets by thermal radiation and conduction without the need for helium coolant. Heat is then transferred from the graphite to the core barrel, then to the reactor vessel, and finally to the ultimate heat sink. Since the presence or flow of helium coolant is not required, cracking of individual core components cannot lead to significant degradation of the passive heat transfer pathway. The graphite components that form this heat transfer pathway are qualified in accordance with ASME Section III Div 5 (Reference 4-8) for the environmental conditions expected and are designed to maintain their structural integrity to permit removal of transient and decay heat during normal operations and postulated events. These features demonstrate compliance with PDC-70(a).

The external cylindrical geometry of the reactor core is maintained by the core barrel, {{

}}$^{a(4)}$

Additionally, the limited available space and interlocking nature of the hexagonal, prismatic components prevent misalignment of stacked control block elements in the event of dowel failure due to shear loading. The graphite and metallic components that form the control rod channels are qualified in accordance with ASME Section III Div 5 for the environmental conditions expected and are designed to maintain their structural integrity to permit insertion of the required number of control rods during normal operations and postulated events (Reference 4-8). These features demonstrate compliance with PDC-70(b).

4.2.2.5 Testing and Inspection

The reactor core will be included in a testing, inspection and monitoring program that will be submitted at the time of the OLA.

4.2.3 Reactivity Control and Shutdown System

The core reactivity is controlled by a combination of movable control rods, fixed burnable absorbers, and a negative temperature reactivity coefficient. This section describes the Reactivity Control and Shutdown System (RCSS), which provides active reactivity control during normal operation and provides shutdown of the reactor in response to abnormal conditions and postulated events. Both normal reactivity control and shutdown of the reactor are controlled by positioning of the control rods (moveable poisons). The burnable absorbers and negative temperature reactivity coefficient are detailed in Section 4.5.

4.2.3.1 Summary Description

The main function of the RCSS is to control reactor power in all designed operating modes and shut down the reactor safely from any operational condition. During normal operation, the neutron generation rate and reactor power are controlled by adjusting the positioning of the control rods in the reactor core and the reactor is shut down by fully inserting the control rods. In situations where abnormal conditions are encountered, the Reactor Protection System (RPS) (Chapter 7.4) cuts power to the control rod drive units and all control rods drop into the core by gravity to shut down the reactor, control heat generation, and maintain the reactor in a safe state. Rapid insertion of the control rods following a reactor trip is the only safety function of the RCSS.

4.2.3.2 Functions, Classification and Design Criteria

4.2.3.2.1 Safety Functions

The safety function of the RCSS is to rapidly insert the control rods during a reactor trip to shut down the reactor, control heat generation, and maintain the reactor in a safe state.

4.2.3.2.2 Non-Safety Functions

The non-safety function of the RCSS is to adjust the positioning of the control rods to control reactor power (and neutron generation rate) in all designed operating modes.

4.2.3.2.3 System Classification

RCSS equipment which is required to function to perform a reactor trip is safety-related and Seismic Category I. Non-safety-related RCSS equipment within the CRDU housing is Seismic Category I to prevent its failure from impacting safety-related equipment.

4.2.3.3 Design Basis

The PDC used to develop design criteria are provided in the approved TRs referenced in Section 3.6.1. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC were identified as applicable to the RCSS:

- **PDC-23,** "Protection system failure modes"
- **PDC-26,** "Reactivity control systems"
- **PDC-28,** "Reactivity limits"
- **PDC-29,** "Protection against anticipated operational occurrences"

4.2.3.4 System Description

The RCSS consists of the seven CRDUs, seven sets of drive unit instrumentation and control (I&C) cabinetry, and seven control rods. The control rods are used for both reactivity control and shutdown of the reactor.

Each control rod consists of {{ }}$^{a(4)}$ annular borated graphite {{ }}$^{a(4)}$ absorber compacts and is fastened to the CRDU {{ }}$^{a(4)}$ drive chain.

Each CRDU is mounted on a standpipe on the top of the reactor which allows the control rods to be inserted into the control rod channels in the reactor core. One control rod channel is in the center of the core and six are spaced equally in the middle ring of the core, as shown in Figure 4-8. All control rods are used both for reactivity control during normal operation and for shutdown of the reactor. The control rods are designed to be operated from the Main Control Room by the Reactor Control System (RCS) (Chapter 7.3). The RCS limits the control rod withdrawal speed, via stepper motor speed, {{ }}$^{a(4)}$, and includes interlocks which ensure that only a single rod can be withdrawn at a time during at-power operations.

The CRDUs, shown in Figure 4-12, position the control rods vertically within the core. Figure 4-13 details the operating range of the control rods relative to the active core. During normal operation, the control rods are inserted into and withdrawn out of the core using a stepper motor, reduction gearbox, and chain hoist. The stepper motor allows for discrete, trackable, and repeatable positioning of the control rod.

Electrical power to the stepper motor is required to maintain control rod position. During a trip, the power to the stepper motor is removed, resulting in the stepper motor losing the required holding torque. Without the holding torque, the motor and gear train are loose and can rotate, only limited by the resistance provided by inertia, frictional forces, and the motor detent torque (torque present when unpowered). This leads to control rods being inserted into the core under the effects of gravity. {{ }}$^{a(4)}$

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}}a(4)

Additionally, a spring on the fixed end of the chain decelerates the control rod as it approaches the fully inserted position. Due to the nature of the spring system, the control rod will slightly exceed the fully inserted position during a reactor trip before returning to the nominal position. Thus, the control rod channel extends a sufficient distance below the nominal fully inserted position to allow for over-insertion, as shown in Figure 4-13. The rod insertion estimates account for the effects of the motor detent torque; frictional and gravitational forces; inertia of the motor, drivetrain, control rod, and chain; {{ }}a(4) During normal operation, the control rods are never fully withdrawn from the control rod channels, which extend above the active core. This minimizes a potential failure resulting from improper alignment of the control rod with its channel.

The control of the CRDUs within the scope of the RCSS is limited to execution of the movement commands received from the RCS. Although the CRDUs receive power from the electrical power systems (Chapter 8) for normal operation, the RCSS does not require electrical power to fulfill its safety function of control rod insertion by gravity. The RPS provides the trip function which isolates the CRDUs from their power source. The pressurized housings of the CRDUs are part of the Vessel System (VS) and are described in Chapter 4.3. The materials and applicable codes for the main RCSS components are outlined in Table 4-4.

Figure 4-11 Control Rod Detail

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Figure 4-12 Control Rod Drive Unit Detail

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Figure 4-13 Control Rod Positioning Range

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Table 4-4 Summary of the CRDU Main Components, Materials and Applicable Design Code

Component	Safety Classification	Material of Construction	Code of Design/ Construction
Control Rod Canister	Safety-related or Important to Safety	{{ }}$^{a(4)}$	ASME BPVC Section III, Division 5 (Reference 4-8)
Control Rod Chain		{{ }}$^{a(4)}$	
Control Rod Absorber		B_4C Compact	ASME BPVC Section III, Division 5 (Reference 4-8) ASTM C750-24 (Reference 4-16) ASTM C751-20 (Excluding Boron content requirements) (Reference 4-17)
Structural Components		Material to be provided in the OLA	ASME BPVC Section III, Division 5 (Reference 4-8)
CRDU Gears, Shafts, and Chainwheel			
{{ }}$^{a(4)}$		{{ }}$^{a(4)}$	
{{ }}$^{a(4)}$		{{ }}$^{a(4)}$	ASME BPVC Section III, Division 5 (Reference 4-8) {{ }}$^{a(4)}$

4.2.3.5 Operational Overview

During normal operation, each CRDU responds to signals from the RCS to adjust the positioning of the control rods. The CRDU I&C equipment receives these signals from the RCS and provides the necessary control power for performing the required number of steps between the current and desired position. During this process, the CRDU I&C will inform the Main Control Room, through the RCS, of control rod position, or torque measurement such that unexpected values can be identified and the operators notified of potential malfunctions.

During a reactor trip, electrical power is cut from the stepper motor by the RPS, and the control rods insert into the core under gravity. {{ }}$^{a(4)}$ spring ensure the control rod is brought to a stop at the fully inserted position while minimizing shocks and mechanical stresses.

4.2.3.6 System Evaluation

The RCSS meets the design bases as described below:

In compliance with PDC-1, safety-related components in the RCSS are fabricated and tested in accordance with generally recognized codes and standards. A suitable quality assurance program is also established and implemented to ensure satisfactory performance of the system's safety function.

The RCSS is protected from all natural phenomena except earthquakes by the Citadel building and the VS. Safety-related components in the RCSS are designed to Seismic Category I. The linked canister design of the control rods, shown in Figure 4-11 provides mechanical flexibility to articulate any postulated offset between graphite elements during both normal operation and seismic events. The channels are sized such that anticipated geometrical changes in the control rods and in the graphite core geometry between component replacement intervals, combined with postulated offsets, do not prevent rod insertion. This feature ensures that rod insertion is possible in any normal or abnormal operating condition. Additional clearance is included in the control rod channel to further improve insertion reliability in a postulated event. These shutdown design features provide conformance to PDC-2.

The RCSS is designed to be fail-safe, safety-related components are located within the VS and protected from external fires. In the event of an external fire, the RCSS still performs its safety function even if power is lost to the CRDU. Internal fires are mitigated using noncombustible materials and self-lubricating components. These features demonstrate conformance to PDC-3.

Safety-related portions of the RCSS are designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated events. There are no postulated missiles, pipe whipping, or discharging fluids which are anticipated to impact the RCSS. Components of the RCSS are designed to withstand the static and dynamic loads anticipated for normal operation, maintenance, testing, and postulated events. This analysis considers the effects of radiation, temperature, wear, aging, and fatigue on the strength and life of components. Heating in the CRDU stepper motor and drive train is limited by cooling on the outside of the CRDU housing. Heating of the control rods and chain is limited by helium flow through each control rod channel. These features demonstrate conformance to PDC-4.

The reactor trip function of the RCSS is initiated by the RPS trip signal which removes power from the CRDU stepper motors or by a loss of electrical power event. Removal of power from the stepper motor allows the control rods to insert into the core by gravity. These features demonstrate conformance to PDC-23.

The control rods are sized to meet the requirements of PDC-26. The secondary means of fulfilling PDC-26 is the negative temperature coefficient which is inherent to the design of the reactor. Compliance with the requirements in PDC-26 is discussed in Section 4.5.

The RCSS limits control rod withdrawal speed to limit the potential amount and rate of reactivity addition. Control rod withdrawal is limited to a single rod at a time to minimize reactivity addition rates. The maximum control rod withdrawal speed is consistent with the analysis in Chapter 13. These features demonstrate conformance to PDC-28.

The RCSS supports an extremely high probability of accomplishing its safety function during postulated events. Upon receipt of an RPS reactor trip signal or loss of electrical power, the control rods insert into the core by gravity. The rod insertion is passive and therefore able to be performed without electrical power, compressed air, or any other external systems. The reactor trip function is safety related and therefore designed to be single-failure proof. Safe shutdown is still achieved if the highest worth control rod fails to insert. These features demonstrate conformance to PDC-29.

4.2.3.7 Testing and Inspection

Condition monitoring is achieved by means of the instrumentation signals provided by the CRDUs. These signals can be verified against baseline quantities to gauge the condition of each drive unit and control rod.

The CRDU assembly allows for the removal and replacement of all major components. Access to the CRDUs can be achieved from the top of the Citadel building once the concrete plug is removed.

Relevant technical specifications, limiting conditions for operation, and surveillance requirements for the RCSS will be provided in the OLA.

4.2.4 Neutron Startup Source

Neutron sources are inserted into the active core to enable startup and to provide the ex-core flux detectors with a sufficient baseline flux during the initial startup, plant transitions and refueling. These sources provide adequate neutron flux levels to ensure a controlled startup, which requires a source strength high enough for the ex-core flux detectors {{ }}a(4) The neutron sources are Californium-252 or similar neutron emitters contained within a metallic source holder, which contains a handling feature for insertion into the core. The metallic source holder is designed to be compatible with the environmental conditions associated with the reactor core during normal operations and anticipated operational occurrences. The neutron source does not perform safety-related functions.

4.2.5 Reactor Internals System

4.2.5.1 Summary Description

The Reactor Internals System (RIS) consists of the core barrel assembly, neutron shield assembly, the core outlet plenum, and the core support structure insulation. Each of these components and their interfaces are described below.

4.2.5.2 Functions, Classification and Design Criteria

4.2.5.2.1 Safety Functions

The safety functions of the RIS are to support the reactor core, maintain the geometry of the reactor core, to provide radial neutron shielding, and permit heat transfer between the core and reactor vessel.

4.2.5.2.2 Non-Safety Functions

The non-safety function of the RIS is to form a portion of the coolant pathway between the core inlet and outlet and the cross-connection vessel and hot gas duct, respectively.

4.2.5.2.3 System Classification

The RIS is safety related and Seismic Category I.

4.2.5.3 Design Basis

The PDC used to develop design criteria are provided in the approved TRs referenced in Section 3.6.1. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC were identified as applicable to the RIS:

- **PDC-10,** “Reactor design”
- **PDC-34,** “Passive residual heat removal”
- **PDC-70(a),** “Citadel design basis”
- **PDC-70(b),** “Reactor vessel and reactor system structural design basis”

4.2.5.4 Design Description

4.2.5.4.1 Core Barrel Assembly Description

The core barrel assembly is the primary metallic core support structure which is designed to support and restrain the reactor core. The core barrel assembly is a {{ }}[a(4)] structure and is comprised of the core barrel itself, the core support structure and the core lateral restraint. The core support structure is located below the reactor core; and the core lateral restraint is situated in the radial gap between the radial reflectors of the reactor core and the core barrel.

The core support structure (shown in Figure 4-14) is a reinforced circular plate. The outer ring of the core support structure rests on {{ }}[a(4)] supports that are integral with the reactor vessel (Section 4.3). The deadweight loads from the reactor core and RIS are transferred to the reactor vessel through these supports. Radial keys on the lower surface of the support ring interface with slots in the reactor vessel supports to locate and constrain the core barrel assembly to the vessel, while permitting relative thermal expansion. An outlet at the center of the support plate attaches to the hot gas duct and provides the flow path for helium coolant exiting the outlet plenum.

Figure 4-14 Reactor Internals System Detail

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}}[a(4)]

The core lateral restraint structure consists of the core barrel, core former and baffle plates, radial seismic keys and restraints, and top flange. Figure 4-14 shows the core barrel as assembled within the entire RIS. The interface between the RIS and the reactor vessel is shown in Figure 4-15. The barrel is located between the reactor core and the reactor vessel. The annular region between the core barrel and reactor vessel provides the flow pathway for cold helium before it enters the reactor core at the top of the core barrel, as described in Section 4.6. The core former and baffle plates provide an interface from the barrel inner surface to the outer surface of the radial reflectors, thereby restraining and defining the geometry of the reactor core. Any lateral or seismic loads imposed on the reactor core are transmitted through the former and baffle plates to the core barrel. {{ }}[a(4)] The core barrel is restrained by {{ }}[a(4)] pairs of keys and keyways in the top region that interface with the reactor vessel. {{ }}[a(4)] These keyed interfaces limit relative movement and accommodate relative thermal expansion between the components. The materials and applicable codes for the RIS are outlined in Table 4-5.

Figure 4-15 Reactor Internals System Support Detail

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}}[a(4)]

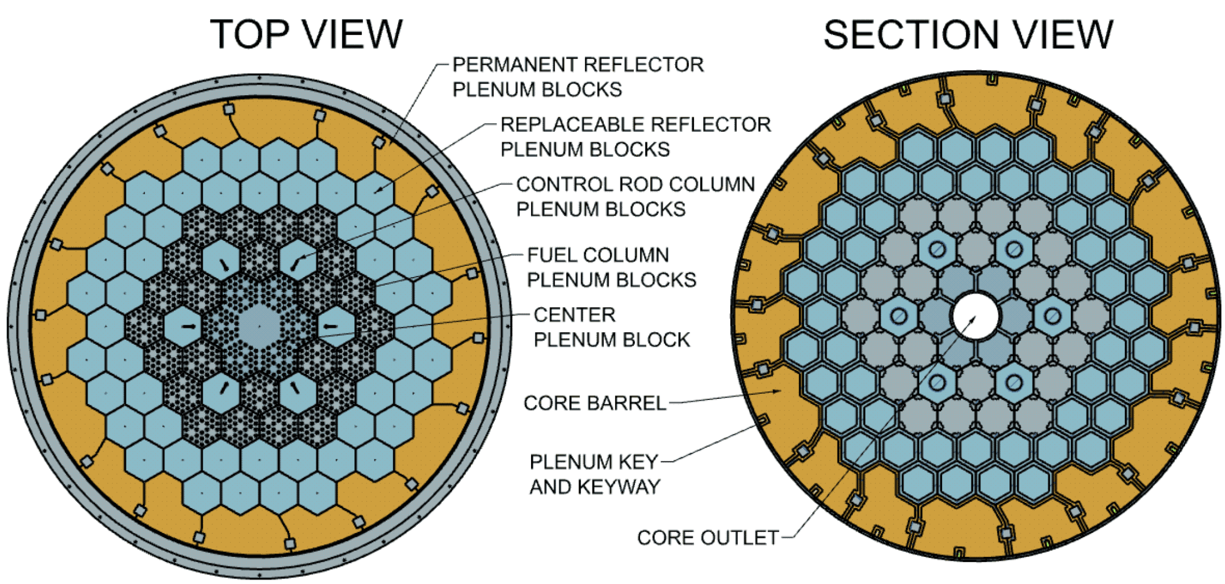
Table 4-5 Summary of Reactor Internals System Components, Materials and Applicable Design Code

Component	Safety Classification	Material of Construction	Code of Design/ Construction
Core Barrel Assembly (including insulation shroud)	Safety-related or Important to Safety	{{ }}[a(4)]	ASME BPVC Section III, Division 5 Sub-section HG (Reference 4-8)
Core Outlet Plenum			
{{ }}[a(4)]			
Neutron Shield Absorber		B_4C{{ }}[a(4)]	ASTM C750-24 (Reference 4-16)
Core Support Structure Insulation		{{ }}[a(4)]	Code basis to be finalized in detailed design and reflected in the OLA

4.2.5.4.2 Core Outlet Plenum Description

The core outlet plenum, shown in Figure 4-16 consists of an arrangement {{ }}[a(4)] blocks which form a pillared cavity below the graphite reactor core. The core outlet plenum supports the reactor core, and it collects and mixes the helium coolant exiting the reactor core through the bottom reflectors and channels it to the core outlet channel at the center. The core outlet plenum is supported vertically by the core support structure and restrained by the core lateral restraint.

Figure 4-16 Outlet Plenum Detail



4.2.5.4.3 Neutron Shield Assembly Description

The radial neutron shield, shown in Figure 4-14 and Figure 4-15, consists {{ }}[a(4)] natural boron carbide B_4C {{ }}[a(4)] to attenuate the neutron flux to the surrounding metallic support structure components. {{ }}[a(4)]

4.2.5.4.4 Core Support Structure Insulation Description

The core support structure insulation consists of a set of layers of low thermal conductivity, {{ }}[a(4)] situated between the core outlet plenum and the core support plate. The inner perimeter of the core outlet formed by the core support structure is also lined with the {{ }}[a(4)] thermal insulation, {{ }}[a(4)]

4.2.5.5 Operational Overview

There are no active systems in the RIS. Under all design basis conditions, the RIS bears the weight of the core, transfers it to the reactor vessel, and maintains the geometry of the core within the acceptable limits. During shutdown and postulated event conditions, the RIS forms a portion of the passive heat transfer pathway to the Reactor Cavity Cooling System (RCCS).

4.2.5.6 System Evaluation

The metallic RIS components are designed in accordance with the requirements of ASME BPVC Section III Division 5, Subsection HG (Reference 4-8) and Regulatory Guide 1.87 Revision 2 (Reference 4-9). A plan for monitoring the integrity of high temperature {{ }}[a(4)] structures over the life of the plant will be included in the OLA. These features demonstrate conformance to PDC-1.

The RIS are protected by all natural phenomena except earthquakes by the Citadel building and the VS pressure boundary. All parts of the RIS are designed to Seismic Category I. These features demonstrate conformance to PDC-2.

Components in the RIS are constructed from non-combustible, fire-resistant materials including high temperature alloys permitted in accordance with the ASME BPVC Section III Division 5 code as well as heat resistant, high-temperature stable {{ }}[a(4)] (Reference 4-8). These features demonstrate conformance to PDC-3.

The RIS is designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated events. There are no postulated missiles or pipe whipping which are anticipated to impact the RIS. Materials in the RIS are selected to withstand appropriate radiation exposure and be compatible with the helium atmosphere at the full range of temperatures. Materials are not anticipated to experience degradation methods such as corrosion or erosion. Components are designed to withstand the static and dynamic loads anticipated for normal operation, maintenance, testing, and postulated events. Final sizing, performance details and failure modes assessment will be provided in the OLA. Components are qualified in accordance with the codes and standards referenced under PDC-1. These features demonstrate conformance to PDC-4.

The RIS is designed to maintain the reactor core geometry, which defines the coolant flow pathway during normal operations. The RIS also defines and maintains the passive heat removal pathway, described in Section 4.6. These features ensure that the fuel condition limits are not challenged and specified acceptable radionuclide release design limits (SARRDL) are not exceeded during normal operations or postulated events. These features demonstrate conformance to PDC-10.

The RIS is designed to support the dissipation of heat from the core to the reactor vessel and ultimately to the RCCS during normal operation and postulated events, at a rate such that SARRDL and the design limits of the reactor vessel are not exceeded. The RIS supports and maintains the geometry required for passive heat removal during postulated events. These features demonstrate conformance to PDC-34.

The RIS is designed to maintain the geometry, of both itself, and the reactor core, within allowable displacements during all modes of operation and postulated events. By maintaining this geometry, the heat transfer pathway from the core to the RCCS is provided to allow removal of the residual and decay heat, and to ensure that the reactor vessel temperatures do not exceed their design limits. By maintaining the geometry of the reactor core, the RIS also ensures the ability to insert the control rods to shut down the reactor. These features demonstrate conformance to PDC-70(a) and PDC-70(b).

4.2.5.7 Testing and Inspection

The RIS is designed to perform its functions without maintenance or replacement throughout the design life of the plant. A detailed plan for addressing aging management will be outlined in the OLA.

The surveillance and in-service inspection of the RIS will be developed under an In-Service and Inspection program in accordance with the applicable ASME Section XI, Division 2 requirements for Reliability and Integrity Management programs, based on the design, fabrication, and applicable degradation mechanisms (Reference 4-10).

Relevant technical specifications, limiting conditions for operation, and surveillance requirements will be provided for the RIS in the OLA.

4.3 REACTOR VESSEL SYSTEM

4.3.1 Summary Description

The Vessel System (VS), shown in Figure 4-17, forms the reactor primary coolant pressure boundary and some supporting structures, systems and components (SSCs). The VS consists of:

- The reactor vessel (RV), including the cylindrical vessel body, bottom head, and top head (also referred to as the reactor vessel lid),

- The intermediate heat exchanger (IHX) vessel, including the cylindrical vessel body and end head (also referred to as the IHX vessel lid),
- The circulator vessel, including a cylindrical vessel body which is sealed at one end with a welded blind flange,
- The cross-connection vessel (CCV), including two 90° elbows and a straight section connecting them,
- The vessel internal and external insulation systems,
- The portions of the molten salt piping between the IHX header and the {{ }}[a(4)] valves, including the hot, cold and drain lines and their associated {{ }}[a(4)] valves. The molten-salt lines are not shown in Figure 4-17, but can be seen in Figure 4-18,
- The helium pressure relief system (HPRS). The pressure relief trains are not shown in Figure 4-17 or Figure 4-18, but are represented schematically in Figure 4-20,
- RV and IHX vessel supports.

The VS is located within the {{ }}[a(4)] concrete Citadel building (described in Section 3.5). The RV and IHX vessel are connected by the CCV, and the circulator vessel is connected to the IHX vessel, thereby forming a continuous pressure boundary. {{ }}[a(4)]

The VS also includes the RV supports, IHX vessel supports, HPRS, and internal and external insulation. Collectively, these components house, support, or interface with the Reactor Core System, RIS, HTS, and RCSS, and form part of the primary helium coolant flow path to the IHX.

Figure 4-17 Vessel Systems Overview

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}}a(4)

Figure 4-18 Section View of IHX Vessel, HPRS and Molten Salt Piping within Scope of Vessel Systems

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}}a(4)

4.3.2 Functions and Classifications

The VS comprises both safety-related (SR) and non-safety-related (NSR) SSCs. For organizational purposes, the VS is described as an SR system based on the highest safety classification of its constituent SSCs; however, not all SSCs within the VS are safety related.

The SR portions of the VS are those SSCs credited with maintaining reactor core geometry and control rod alignment with the core, in support of reactor shutdown and heat removal during normal operation and postulated events. For these reasons, the RV and RV supports are classified as safety-related to maintain required geometry and structural support of the reactor core. The HPRS is also classified as safety-related because it limits primary coolant pressure and thereby prevents damage to the safety-related RV due to overpressure events. These SSCs are designed to maintain their required functions under applicable thermal and pressure transients, including those associated with depressurization and loss-of-coolant events, such that structural integrity and required geometry are preserved for the duration of the event.

Other VS components, including the circulator vessel, CCV, IHX vessel and associated supports, and molten salt piping and valves belonging to the VS, are classified as NSR and do not perform or support fundamental safety functions.

The VS is not credited as the primary means of performing radionuclide retention and confinement functions, which are addressed by other credited SSCs described elsewhere in the PSAR (see Chapter 6).

4.3.3 Design Basis

The design basis incorporates applicable codes, standards, and Principal Design Criteria (PDC) to ensure safety functions are met throughout the reactor lifetime.

4.3.3.1 Codes and Standards

ASME Boiler & Pressure Vessel Code (BPVC), Section III, Division 5 (Reference 4-8), is primarily used to establish the design basis for the helium pressure vessels and HPRS. This is not the only code & standard used in the design, fabrication, inspection, and testing of VS components. Certain VS components are designed in accordance with the requirements of other applicable codes and standards. {{

}}[a(4)] However, as additionally described in Section 5.3, the portions of molten salt piping inside of the IHX vessel belong to the VS and represent a code break transition between the IHX and the salt piping outside the IHX vessel. The specific code break transition and interface requirements will be provided in the OLA. Non-pressure-boundary components, including internal insulation, may be qualified to applicable alternative standards (e.g., ASTM material specifications) once material selection has been finalized. A summary of the Vessel System components and applicable design codes is presented in Table 4-7.

4.3.3.2 Principal Design Criteria

The PDC used to develop design criteria are provided in the approved TRs referenced in Section 3.6.1. In addition to the general PDC discussed in Section 3.6.1.4, the following PDC were identified as applicable for the VS:

- **PDC-10**, “Reactor design”
- **PDC-14**, “Reactor helium pressure boundary”
- **PDC-15**, “Reactor helium pressure boundary design”
- **PDC-30**, “Quality of reactor helium pressure boundary integrity including reducing fluid ingress”
- **PDC-31**, “Fracture prevention of reactor helium pressure boundary”
- **PDC-32**, “Inspection of reactor helium pressure boundary”
- **PDC-45**, “Inspection of structural and equipment cooling systems”
- **PDC-46**, “Testing of structural and equipment cooling systems”
- **PDC-70a**, “Citadel design basis”
- **PDC-70b**, “Reactor vessel and reactor system structural design basis”

4.3.4 Design Description

The VS, shown in Figure 4-17, forms the helium pressure boundary and provides structural support for the Reactor Core System, RIS, Heat Transport System (HTS), and RCSS. {{ }}[a(4)] and interface with the secondary coolant system described in Section 5.3. The {{ }}[a(4)] valves can be actuated to limit helium release through the molten salt lines in the case of an IHX leak, {{ }}[a(4)] The VS contains both SR and NSR portions of the primary helium pressure boundary. The RV structures credited for VS safety functions maintain core geometry, control rod alignment, and the passive heat removal geometry required to support safe shutdown and heat removal. In addition, the VS forms part of the flow path for helium primary coolant, enabling transfer of heat from the reactor core to the secondary loop via the IHX, as described in Section 4.6. The annular region between the core barrel and RV defines a flow pathway for cold helium returning from the IHX via the CCV. The VS is therefore designed such that only cold inlet helium comes into contact with the helium pressure vessels. The major components of the VS are described in this section.

The RV, detailed in Figure 4-19, consists of a cylindrical body which is welded to the bottom ellipsoidal section and connected via bolted flange to the top hemispherical head (also referred to as lid). The RV supports and aligns the core barrel assembly of the RIS, which restrains and locates the reactor core, by means of internal supports in the lower section of the RV and {{ }}[a(4)] radially oriented seismic keys at the top of the RV, as described in Section 4.2.5. The supports and interfacing core barrel components will be designed with due consideration of local stress concentrations arising from normal operating, transient, and dynamic loading conditions, in accordance with the requirements of ASME BPVC Section III, Division 5 (Reference 4-8). These features ensure that the reactor core geometry, control rod guide alignment, and helium flow paths remain within design tolerances under seismic conditions and relative thermal expansion of the RIS and VS. Details of the design of these core support features will be provided in the OLA.

The top head of the RV, depicted in Figure 4-19, has seven control rod standpipes, each capped by a CRDU housing. The standpipes and CRDU housings contain the RCSS (Section 4.2.3) within the pressure boundary, with the standpipes locating the control rods relative to the reactor core. The CRDU housings support the weight and operational loads of the RCSS during normal operation and postulated events. During refueling, one CRDU and its associated control rod are removed to provide access through the standpipe locations for interfacing with the Fuel Handling and Storage System (Section 9.2). Sealing of the refueling interface is provided by {{ }}[a(4)] Accordingly, the RV and associated supports are designed to accommodate the interface sealing loads and applicable accidental handling impact loads associated with refueling but are not required to support the weight {{ }}[a(4)]

Figure 4-19 Reactor Vessel Configuration

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}}a(4)

The IHX vessel, shown in Figure 4-17 and Figure 4-18, houses and supports the IHX, which is a component of the HTS (Section 5.1) that facilitates heat transfer between the helium primary loop and molten salt secondary loop. The IHX vessel is mechanically interfaced with the circulator vessel and connects to the CCV via a flanged connection to create a continuous helium pressure boundary. On the molten salt side, the hot and cold salt lines penetrate the IHX vessel to connect to the IHX. The penetration interfaces are designed to maintain the required pressure boundary function at the interface. {{

}}a(4); the associated secondary coolant system functions outside this boundary are described in Section 5.3.

The circulator vessel, shown in Figure 4-17, contains the helium circulator which drives coolant through the primary loop. The circulator vessel has a forged flange connection to the IHX vessel, minimizing penetrations and maintaining structural integrity under all operating conditions.

The CCV, shown in Figure 4-17, provides a pressure-retaining channel between the IHX vessel and the RV, completing the helium primary coolant loop. In all operational modes, only cold helium returning from the IHX contacts the IHX vessel, CCV, and RV. The hot gas duct, part of the HTS, is concentrically positioned within the CCV and contains the flow of hot helium exiting the reactor core, as described in Section 5.2. The hot helium within the hot gas duct is at a slightly lower pressure than the cold helium in the CCV, ensuring that any leakage path between the helium hot and cold legs will not result in the hot helium impacting the pressure boundary. As discussed in Chapter 13, postulated event analyses indicate that peak RV temperatures would remain within the design limits of ASME BPVC, Section III, Division 5 (Reference 4-8) during any of the analyzed bounding postulated events. This arrangement maintains structural integrity and minimizes thermal stresses to the VS under all operating conditions.

The RV supports and IHX vessel supports, shown in Figure 4-17, maintain alignment and stability of the VS under all operating and seismic conditions. The RV supports consist of support columns anchored on the bottom floor of the reactor cavity within the Citadel building. {{ }}[a(4)] The RV supports also include provisions to ensure stability under seismic loading. The IHX vessel supports, which are anchored to the bottom floor of the IHX cavity in the Citadel building, accommodate thermal expansion of the CCV and IHX vessel using {{ }}[a(4)] Bases used for the initial sizing of the RV supports and IHX vessel supports include evaluation of bending, buckling and compression of their constituent parts. It considered vessel deadweight carried through the supports, and {{ }}[a(4)] associated with support reactions and thermal expansion movements. These coarse estimates were performed to verify that the assumed support configurations are sufficient for basic loading conditions; detailed support analyses will be provided in the OLA.

Internal and external insulation, shown in Figure 4-17, is provided to support thermal performance of the VS. The RV internal insulation {{ }}[a(4)] limits heat transfer from hot convective helium currents to the vessel lid during postulated conditions. It is not credited to support the RV's performance of its safety functions. Supporting details and information will be provided in the OLA. {{ }}[a(4)] the IHX vessel and portions of the CCV minimizes heat loss during normal operation.

The HPRS is a safety-related subsystem of the VS that provides overpressure protection for the credited helium pressure boundary by limiting the maximum pressure of the helium primary coolant and protecting from reactor vessel rupture due to overpressure. {{ }}[a(4)] redundant pressure relief trains, shown schematically in Figure 4-20, connect to the IHX vessel and provide an escape pathway for helium in overpressure conditions. Each relief train consists of discharge piping, block valves, and a pilot-operated safety relief valve in series with a rupture disk. The block valves facilitate maintenance of a single HPRS train and are interlocked to ensure that at least one relief train remains available at all times. {{ }}[a(4)] Upon activation, the HPRS vents helium into the IHX cavity of the Citadel building. The HPRS is designed in compliance with the requirements of the ASME BPVC Section III, Division 5 (Reference 4-8), subsection HB.

Figure 4-20 Helium Pressure Relief System Schematic Diagram

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}}a(4)

The VS design incorporates double-seal arrangements at flanged connections, high-quality welds qualified per ASME BPVC Section III, Division 5 (Reference 4-8), and pressure monitoring on selected penetrations to ensure early detection of boundary degradation. A list of pressure boundary penetrations is presented in Table 4-6. The helium pressure vessels operate at a higher pressure than surrounding systems, preventing ingress of air or molten salt secondary coolant in the event of minor breaches.

Table 4-6 Helium Pressure Boundary Penetration List*

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}}a(4)

*Penetration counts and locations are not final. Updated details will be provided with the OLA.

The VS is designed to maintain structural integrity and limit leakage to a low allowable rate under all credible operating conditions, including postulated events. All helium boundary components of the VS (i.e., the RV, CCV, IHX vessel, circulator vessel, and molten salt components within scope) are designed in accordance with ASME BPVC Section III, Division 5 (Reference 4-8), subsection HB.

The RV supports and IHX vessel supports are designed to satisfy the materials, design, fabrication, examination, testing, inspection, and certification requirements of ASME BPVC Section III, Division 5 (Reference 4-8), subsection HFA. The components will be constructed from materials selected for their strength, high-temperature creep/corrosion resistance, radiation tolerance, and compatibility with high-temperature helium coolant.

A summary of the VS components, classifications, principal materials of construction, and primary code basis, is presented in Table 4-7. Additional considerations include:

- {{ }}a(4) RV and IHX vessel supports {{ }}a(4) accommodate thermal expansion and minimize friction during seismic or thermal transients, with minimal need for maintenance.
- The material for the molten salt piping and valves within the credited helium pressure boundary will be selected with consideration of its compatibility with radiation control, corrosion resistance, and thermal stresses associated with normal operation and postulated events.
- The vessel insulation material will be selected with consideration of its structural and insulative properties, and its tolerance of thermal, physical, chemical and irradiative degradation.

All pressure-retaining components will be designed to withstand neutron fluence and gamma heating over the reactor lifetime. The selected low-alloy steels exhibit stable mechanical properties under irradiation and maintain ductility at operating temperatures. Helium coolant chemistry is controlled by the HPS (Section 5.4) to limit impurities that could cause oxidation or carburization. Performance analysis and evaluation of the materials' satisfaction of design requirements will be provided with the OLA.

Table 4-7 Summary of Vessel Systems Components, Materials and Applicable Design Code

Component	Safety Classification	Material of Construction	Code of Design/ Construction
Reactor vessel (vessel body, bottom head, and top head)	Safety-related	Low-alloy steel (SA-508 Grade 3, Class 1 forgings and SA-533 Type B, Class 1 plate)	ASME BPVC Section III, Division 5 Sub-section HB
Helium pressure relief system (HPRS)			
Intermediate heat exchanger vessel (vessel body and end head)	Non-safety-related		
Circulator vessel (vessel body)			
Cross-connection vessel (90° elbows and straight section)			
Molten salt piping within the IHX vessel		Material to be provided in the OLA	Code basis to be finalized in detailed design and reflected in the OLA
Molten salt piping and {{ }}a(4) valves outside of the IHX vessel but {{ }}a(4)			ASME BPVC Section III, Division 5 / ASME TES-1
Vessel internal and external insulation systems			Code basis to be finalized in detailed design and reflected in the OLA
Reactor vessel supports	Safety-related	Carbon steel (SA-516 Grade 70)	ASME BPVC Section III, Division 5 Sub-section HFA
Intermediate heat exchanger vessel supports	Non-safety-related		
Reactor vessel {{ }}a(4)	Safety-related	Material to be provided in the OLA	Code basis to be finalized in detailed design and reflected in the OLA
IHX vessel {{ }}a(4)	Non-safety-related		

4.3.5 System Evaluation

The components of the VS are designed, fabricated, erected, and tested in accordance with the applicable codes of construction summarized in Table 4-7. Consistent with PDC-1 and PDC-30, these activities will be performed in a manner that complies with the U. of I. NRC-approved Quality Assurance Program (Reference 4-13).

The components of the VS, such as the RV head, the circulator vessel, attached piping and instrumentation connections will have a dynamic response during a design basis earthquake. In addition, systems contained within the VS, such as the RIS, RCSS and HTS, will move relative to the VS components and apply additional loads to the VS. Consistent with PDC-2, seismic analysis will be performed to characterize these loads on the VS components, and the results will be used to refine the VS design to prevent damage of the SR components during a design basis earthquake. Seismic restraints are included in the RV and IHX Vessel supports, which are designed to be compliant with ASME BPVC III Division 5 (Reference 4-8), Subsection HF. The findings of the seismic analysis may lead to the inclusion of further seismic restraint attachments at the top of the RV to resist lateral movement, and increased rigidity of the helium pressure vessels and/or vessel supports. The integrity of the VS will additionally be safeguarded against potential failures of neighboring NSR SSCs in the event of a design basis earthquake. This will be achieved through seismic mounting, physical separation, or the implementation of a barrier to prevent any adverse interactions. Preliminary evaluations show that the seismic performance complies with requirements. Detailed seismic analysis will be included in the OLA.

Consistent with PDC-3, the VS is designed such that credible fire events do not prevent the VS from performing its required safety functions. The VS is located within the reactor cavity of the Citadel building, which is constructed of noncombustible materials and is separated from areas containing combustible loading (see Section 3.5). Materials selected for the VS, including vessel steels and insulation materials, will be selected with consideration for flammability, with preference given to materials that do not contribute to fire propagation. Where VS components interface with NSR SSCs, physical separation and fire-resistant barriers will be provided as necessary to prevent fire-induced damage that could challenge VS structural integrity or pressure boundary performance. As discussed in Section 9.3, fire hazard analyses will be performed, with results provided with the OLA.

Consistent with PDC-4, the VS is designed to accommodate environmental and dynamic effects associated with normal operation, maintenance and postulated events. These effects include pressure and temperature transients, flow-induced vibration, and load combinations arising from interfacing systems and components. By designing the VS in accordance with applicable codes and standards, including ASME BPVC Section III, Division 5 (Reference 4-8), the SSCs are given appropriate design margins to withstand these effects without loss of structural integrity or pressure boundary function. Detailed stress, fatigue, and vibration analyses will be performed to confirm acceptability under all applicable load combinations and will be documented in the OLA.

Consistent with PDC-10, the VS provides a helium pressure boundary that supports the defense-in-depth strategy by limiting the transport and release of radionuclides during normal operation and postulated events. {{

}}[a(4)] The VS is not credited with functional containment, which is described in Chapter 6, but the RV and RV supports support conformance with PDC-10 through maintenance of the required core and control rod channel geometry. The HPRS further supports conformance with PDC-10 by limiting primary coolant pressure and protecting the integrity of the helium pressure vessels. By maintaining the geometry of the core assembly, the VS both sustains heat conduction pathways for the passive removal of decay heat and facilitates control rod insertion for safe reactor shutdown during postulated events, which supports compliance with PDC-70a and 70b, respectively.

Consistent with PDC-14, the VS components are designed to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. The reactor vessel components are fabricated from low-alloy steel that was selected with consideration for its fracture toughness at reactor operating conditions, thus reducing the likelihood of crack propagation. Material selection for the other VS components also consider fracture toughness at operating and design basis conditions. The redundant pressure-relief trains of the safety-related HPRS preclude overpressure and thereby prevent dramatic rupture of the helium pressure vessels, such that applicable design-basis pressure-boundary failure can be bounded as a depressurization event. This treatment is consistent with the postulated event evaluations in Chapter 13, which analyze applicable pressure-boundary failure events as depressurization events and credit safety-related SSCs to achieve a safe shutdown state.

Chemical attack on the credited helium pressure boundary is limited through design and operating procedures, thereby demonstrating further compliance with PDC-14. The internal pressure of the VS is significantly higher than that of the surrounding area and interfacing systems, meaning that any potential leakage would make helium exit the VS rather than permit ingress of external fluids. The purity of the helium primary coolant is controlled by the HPS (Section 5.4), and fluid ingress during conditions where external pressure could exceed helium pressure (e.g., during helium cleanup) is mitigated by a double-seal arrangement at flanged connections. In addition, facility operating procedures minimize non-helium fluid concentrations following conditions where air ingress may occur (e.g., after open maintenance and refueling) by using the primary coolant cleanup system (Section 5.4) to purge the VS and evacuate the helium pressure boundary. Consistent with PDC-30, the VS also incorporates helium activity monitoring and leakage detection through the Unit Monitoring System (see Chapter 7.8), ensuring early detection of pressure boundary degradation. In the event of helium leakage through the IHX, actuation of the {{ }}[a(4)] valves limits energetic helium release through the molten salt lines {{ }}[a(4)] as described in Section 5.3.4.

Consistent with PDC-15 and PDC-31, the use of appropriate design codes ensures that the helium pressure vessels and molten salt components are designed with sufficient margin to maintain the integrity of the pressure boundary during normal operation and postulated events. The VS components are designed to withstand temperatures up to {{ }}[a(4)], which comfortably bounds the helium inlet temperature of 300°C. The annular configuration of the barrel and RV, and the pressure differential between cool and hot legs of the coolant flow-path prevent hot helium from contacting the pressure boundary. To encompass transient conditions exceeding {{ }}[a(4)], the credited helium pressure boundary components are designed in accordance with the construction code, Mandatory Appendix HBB-II of ASME BPVC Section III Division 5 (Reference 4-8) Subsection HB, which addresses elevated-temperature effects such as creep and related time-dependent behavior.

The neutron damage experienced by the RV from the reactor core will be attenuated by the side reflectors of the Reactor Core System (Section 4.2.2) in combination with the neutron shielding provided by the RIS (Section 4.2.5). Vessel fluence calculations will confirm adequate margin relative to the effects of irradiation, with results provided in the OLA. Changes in parameters relevant to pressure boundary integrity (e.g., irradiation, pressure, and temperature) will be addressed through monitoring, in-service inspection, and testing consistent with the construction code. Corrosion and chemical attack on the molten salt components are mitigated through maintenance and monitoring of salt impurity levels via the Thermal Energy Storage System (Section 5.3), use of corrosion-resistant construction materials, and design of the components exposed to molten salt with adequate corrosion allowance. Flow rate and temperature at the molten salt interface are also monitored to ensure that the design conditions of the IHX and associated pressure-boundary components are not exceeded.

The Citadel building is designed to facilitate personnel or remote tooling access to the reactor and IHX cavities, thereby supporting in-service inspection of the VS. Consistent with PDC-32, the uninsulated RV body and bottom sections improve accessibility for in-service inspection. Welds and flanged connections are designed such that they can be accessed remotely for necessary inspections. The external insulation used on the IHX vessel and the CCV is designed to be removable to facilitate in-service inspection where required. The RV and IHX vessel bottom heads include penetrations for personnel and equipment access, providing planned access points that support inspection activities. Provisions will also be incorporated during detailed design of the Citadel building (Chapter 3) and RCCS (Chapter 6) to facilitate access to the VS for inspection and maintenance. Consistent with PDC-32, PDC-45 and PDC-46, {{ }}[a(4)] helium pressure boundary are designed to be readily inspected and tested and to permit appropriate periodic functional testing to assure component integrity and operability. Additional design details, relevant inspection procedures and testing plans will be detailed as part of the OLA.

4.3.6 Technical Specifications

Preliminary operating limits for the VS are established to ensure structural integrity and pressure boundary integrity under all conditions. The RV is designed to meet all codes and standards required for a design pressure of {{ }}[a(4)] MPa and design temperature of {{ }}[a(4)], while the operating pressure and helium inlet temperature are 6.0 MPa and 300°C, respectively. Minimum wall thickness and allowable stress values for the helium pressure vessels comply with ASME BPVC Section III, Division 5 (Reference 4-8) requirements, and the VS will also be designed to withstand loads associated with all operating conditions, including postulated events. For the portions of the molten salt system included in the helium pressure boundary, applicable design pressure, temperature, material, and leak-tightness limits will also be established in accordance with the applicable design codes and standards described in Section 4.3.3.1.

These limits, together with the limits for the molten salt system discussed in Section 5.3, will form the basis for technical specifications to be provided in the OLA.

4.3.7 Testing and Inspection

As discussed in Section 4.3.5, VS design includes access points and {{ }}[a(4)] to facilitate in-service inspection. Periodic inspections will assess the condition of the VS components, including overall structural integrity, corrosion, weld condition, {{

}}[a(4)]

Surveillance programs will monitor for signs of radiation-induced degradation, chemical attack, thermal fatigue, and salt-side degradation mechanisms throughout the reactor lifetime. The VS incorporates helium activity monitoring and leakage detection through the Unit Monitoring System (Chapter 7.8), ensuring early identification of abnormal pressure boundary conditions. Periodic in-service inspections of welds, flanges, and {{ }}[a(4)] ensure structural integrity and leakage within the allowable rate.

Inspection intervals, methods, and acceptance criteria will be submitted with the OLA.

4.4 BIOLOGICAL SHIELD

4.4.1 Summary Description

The biological shield limits radiation exposure to personnel and the public in accessible areas of the facility during all operational and shutdown states. In addition, it protects adjacent SSCs from neutron and photon fluence that would cause significant material degradation or activation.

The biological shielding is primarily provided by a combination of the Citadel building (Chapter 3), the graphite reflectors of the reactor core system (Section 4.2.2), the shielding cartridges of the RIS (Section 4.2.5), and the RCCS (Chapter 6).

4.4.2 Functions and Classifications

The biological shield is designed to maintain direct external radiation dose rates below regulatory limits for both controlled and unrestricted zones, consistent with 10 CFR Part 20 (Reference 4-14) and as low as reasonably achievable (ALARA) principles as outlined in Chapter 11. Additionally, it limits irradiation damage to the metallic core support structures and the reactor vessel, and it provides the means for absorption of neutrons and photons in the shielding structures.

The biological shield is not a single system, however, the components that comprise the shielding belong to safety-related systems; namely, the reactor core system, RIS, RCCS, and Citadel building. While the biological shield does not prevent or mitigate the consequences of a postulated event, it remains intact during and after normal operations and postulated event conditions.

4.4.3 Design Basis

The PDC used to develop design criteria are provided in the approved TRs referenced in Section 3.6.1. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC were identified as applicable to the biological shield:

- **PDC-19**, "Control room or secure remote monitoring facility"
- **PDC-61,** "Fuel storage and handling and radioactivity control"

The biological shielding design is governed by the dose limits established in 10 CFR Part 20 (Reference 4-14), which specify a maximum total effective dose equivalent (TEDE) of 50 mSv per year for occupational radiation workers (§20.1201), and 1 mSv per year for members of the public (§20.1301). In support of ALARA principles defined in Section 11.1.3, the reactor design adopts more conservative dose limit design targets to be provided in the OLA. These targets are not regulatory requirements but are internal administrative and design constraints used to ensure margin and facilitate shielding optimization. Further discussion on the radiation protection approach is provided in Chapter 11.

4.4.4 Design Description

4.4.4.1 Radiation Source Terms

During reactor operation at the nominal rated power of 45 MWth, the radiation source terms include:

- Fission source neutrons and photons
- Fuel decay source neutrons and photons
- Secondary photons from particle interactions, such as neutron capture (n, photon)
- Decay of radionuclides distributed throughout the facility

More detailed discussion is provided in Section 11.1.

4.4.4.2 Design Approach

The shielding is designed based on neutron and photon attenuation requirements to meet various dose, material activation and radiation damage targets. Material selection and dimensions are derived from shielding analysis and activation studies.

The concrete Citadel building serves as the main biological shield. Hydrogen content in the concrete contributes to thermalizing neutrons, which are subsequently absorbed within the concrete. Additionally, bulk concrete attenuates photon intensity. While not credited towards the 10 CFR Part 20 (Reference 4-14) dose limits, water in the RCCS also contributes a shielding effect before neutrons and photons reach the Citadel concrete and therefore contributes to the ALARA approach.

The Citadel building is compartmentalized into distinct radiation zones to control exposure and support operational access. Figure 4-21 and Figure 4-22 illustrates the Citadel building zoning, with red representing high-radiation zones, green representing medium-radiation zones and white representing low-radiation zones, with dose rate limits and access restrictions to be provided in the OLA. The zones include:

- Reactor, CRDU, and IHX cavities, which are inaccessible during power operation.
- Equipment room, {{ }}a(4) helium services room, battery room and access corridors, which are accessible during power operation and are shielded to maintain dose rates below design targets, although access to these areas during operation will be controlled and time-monitored for each worker.

Access to these zones is strictly controlled for both security and radiation protection, with locked bulkhead doors and monitored entry points.

Figure 4-21 Radiation Zones and Illustration of Concrete in the Citadel Building, Elevation View

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Figure 4-22 Radiation Zones and Illustration of Concrete in the Citadel Building, Section View at -1 m

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}}a(4)

The reactor core and RIS together significantly limit neutron leakage from the core. The top, bottom, and side graphite reflectors thermalize fast neutrons leaking from the fueled core, significantly increasing the probability of those being captured. The internal shielding, which consists of the radial and axial neutron shield within the core barrel of the RIS, surrounds the reflector and absorbs neutrons in the boron carbide (B_4C) material. These components also contribute to attenuation of photon radiation. The core reflector and internal shielding are shown in Figure 4-23.

By reducing neutron leakage, the reflector and neutron shield limit fast neutron fluence to the reactor vessel wall well below embrittlement limits, and reduce activation of the core barrel, reactor vessel and other ex-core structures. By reducing activation of such large structures, their associated waste class is potentially reduced, contributing to ALARA in the design for decommissioning.

Figure 4-23 Reactor Core Reflector and Internal Shielding

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}}a(4)

During maintenance and/or refueling operations, additional movable shielding may be used to limit worker doses. Generally, movable shield solutions will be made of high-density materials and/or hydrogen-dense materials, depending on the use case.

Penetrations through the reactor core, RIS, and Citadel building are designed with measures such as sealing or indirect paths where possible to minimize neutron streaming as a source of radiation.

4.4.4.3 Geometry, Materials and Configuration

The dimensions and material composition of the credited biological shielding layers are provided in Table 4-8. The shielding materials used in the Citadel building and RIS are selected to provide effective attenuation of neutron and photon radiation, while maintaining structural integrity and compatibility with thermal and environmental conditions.

Table 4-8 Biological Shielding Layer Geometry, Material and Function Summary

Layer	Approximate Thickness	Material	Function
Graphite reflectors	{{ }}a(4) {{ }}a(4)	Nuclear-grade graphite	Reflect neutrons Thermalize fast neutrons
Radial and axial neutron shield	{{ }}a(4)	B_4C {{ }}a(4)	Absorb neutrons
Citadel building concrete	{{ }}a(4)	{{ }}a(4) concrete	Main biological shield; absorb neutrons, absorb photons

Detailed shield geometry and material specifications, including Citadel building walls, roof slab, and penetrations, will be included in the OLA. These will include additional local shielding needed to support potential access requirements.

4.4.4.4 Environmental Considerations

The layered biological shielding design supports environmental protection by limiting neutron and photon leakage from the Citadel building, thereby limiting irradiation and activation of the surrounding soil and groundwater. Pre-operational soil assays shall be performed to detect trace elements affecting long-term activity levels. This site data on soil composition will be used to calculate potential activation of soil and groundwater for evaluation against applicable NRC limits, to be provided in the OLA.

4.4.5 System Evaluation

Shielding calculations are performed primarily using the Standardized Computer Analyses Licensing Evaluation (SCALE) code suite, version 6.3 (Reference 4-15). SCALE is generally validated in the following domains: reactor physics, nuclear criticality, spent nuclear fuel and radiation shielding.

Full usage details and coupling methodologies of the various codes and sequences within SCALE will be provided in the OLA. The analyses account for the various source terms, material properties, geometry, and potential streaming paths. Dose rates for both full power and shutdown conditions will be provided in the OLA.

Dose rate results are evaluated in worker dose assessments. The final combination of dose rates and worker access times will show compliance with PDC-19, PDC-61 and 10 CFR Part 20 (Reference 4-14). Unrestricted access areas open to the public will comply with the design bases. Material damage and activation will be assessed against engineering codes and engineering requirements. Activation of ex-core materials, including soil and groundwater, will be assessed against applicable NRC limits. Detailed results will be provided in the OLA.

4.4.6 Testing and Inspection

The biological shield is designed to maintain its structural and radiological performance over the full operational life of the reactor facility. This includes mechanical integrity under thermal and seismic loads, resistance to radiation-induced degradation, and control of activation in structural materials.

The biological shielding is subject to technical specifications and surveillance requirements to ensure continued compliance with dose limits and structural integrity over the facility's operational life, to be provided in the OLA.

4.5 NUCLEAR DESIGN

This section describes the nuclear design of the U. of I. research reactor, including the design bases and the analytical methods used to perform the nuclear design. The reactor is designed to generate a maximum of 45 MWth while achieving significant operational and safety margins. It is fueled using annular ceramic-matrix fuel pellets, leveraging a uranium enrichment of ≤9.90 wt%. Analytical results are presented for operation at the full rated power with a single refueling strategy, which captures the most limiting conditions for the reactor and fuel for normal power operation.

4.5.1 Design Basis

The Principal Design Criteria used to develop design criteria are provided in the approved TRs referenced in Section 3.6.1. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC were identified as applicable to the nuclear design of the reactor:

- **PDC-10**, "Reactor design"
- **PDC-11,** "Reactor inherent protection"
- **PDC-12,** "Suppression of reactor power oscillations"
- **PDC-26,** "Reactivity control systems"
- **PDC-28,** "Reactivity limits"

4.5.2 Normal Operating Conditions

4.5.2.1 Overview of Core Nuclear Design

The reactor core is a prismatic high-temperature gas-cooled core that produces heat by the fission of enriched uranium fuel contained in the TRISO-based fuel pellets placed within the graphite fuel blocks. The fuel blocks and control blocks comprise the active region of the core, and are surrounded radially and axially by graphite reflector elements. Details of the core layout and components are provided in Section 4.2.2. Neutron moderation is provided both by the prismatic graphite blocks of the active core and by the surrounding graphite reflector elements. The core is under-moderated during all operating conditions and is ensured to have a compact configuration using a hexagonal lattice with minimal gap sizes, which precludes more reactive core configurations. At the end of an operating cycle, all fuel elements are replaced with fresh fuel elements, and therefore only one core configuration will be used during the life of the reactor, albeit control rods will be used for several cycles. A summary of key nuclear design parameters is provided in Table 4-9.

Core reactivity is controlled by a combination of fixed burnable absorbers (BA) placed directly in the graphite blocks, movable control rods, and negative reactivity feedback. As described in Section 4.2.3, the RCSS consists of seven control rods that are moved independently and one at a time into and out of the reactor core for controlling the reactor reactivity during normal operation, while all rods insert fully under gravity during a reactor trip under normal operations and all postulated events. The RCSS also ensures a shutdown margin of >1% Δk/k after accounting for all uncertainties, demonstrating compliance with PDC-26 (1), (3), and (4). The control rods are inserted into channels in the control block element columns, which do not contain any fuel.

The BA configuration is designed to offset most of the excess reactivity during the cycle, while balancing the impact of residual B-10 on the cycle length. By using the BA to manage excess reactivity, control rods remain near the top of the active core throughout the cycle, which avoids high power peaking near the bottom of the core. This design approach maintains compliance with PDC-10, ensuring that sufficient power and temperature margins are maintained in the fuel so that SARRDL are not exceeded during normal operation and postulated events. In addition, limiting excess reactivity using BA ensures that the shutdown margin of the control rods is sufficient to demonstrate compliance with PDC-26 (1), (3), and (4). {{

}}a(4)

In addition to the BA and control rods, core reactivity is also passively controlled via inherent reactivity feedback, demonstrating compliance with PDC-26 (2). Small power variations can be performed by modulating the speed of the helium circulator, where reducing the speed increases core temperatures and in turn decreases core reactivity without moving the control rods. The core can also be entirely shut down by tripping the helium circulator, which stops the flow of primary coolant, and the negative temperature reactivity coefficient immediately reduces the core reactivity and shuts it down until it passively cools. The combined temperature coefficient is negative over the normal operating range and for all postulated event conditions.

Reactor startup and power ascension is achieved by withdrawal of control rods and adjustment of the helium circulator speed and molten salt pump speed. The operating modes are described in Chapter 14, and the associated detailed analysis of reactor performance will be provided in the OLA.

Table 4-9 Nuclear Design Parameters for the Reactor Core

Parameter	Value
Rated Power (MWth)	45
{{ }}a(4)	{{ }}a(4)
{{ }}a(4)	{{ }}a(4)
{{ }}a(4)	{{ }}a(4)
{{ }}a(4)	{{ }}a(4)
{{ }}a(4)	{{ }}a(4)
{{ }}a(4)	{{ }}a(4)
{{ }}a(4)	{{ }}a(4)
{{ }}a(4)	{{ }}a(4)

4.5.2.2 Analytical Methods

Steady-state neutronics analysis was performed using the continuous-energy Monte Carlo neutron transport code Serpent version 2.2.0 with the ENDF/B-VIII.0 nuclear cross-section data library. Serpent's particle fuel modeling capability for HTGRs was utilized to model the TRISO particles explicitly. An analytical thermal-hydraulics solver was coupled to Serpent to update the core temperature distribution at every depletion step using a tightly coupled iterative process. Each iteration involved (1) a criticality search where control rods were adjusted to the critical position followed by (2) calculations of fuel and moderator temperature distributions with the thermal-hydraulics solver, using the power distribution resulting from control rods at the critical position. The compositions of both the fuel and control rods were depleted. A Python-based module was used to couple the thermal-hydraulics solver with Serpent.

The analysis described throughout this section was performed for the full-power condition with equilibrium xenon, and with all seven control rods moved in unison. Analysis of the control rods at different heights relative to one another, as reflected by the operating strategy to move each control rod individually, will be provided in the OLA.

The method for validation and verification of these codes including the method for determining uncertainty factors will be provided with the OLA.

4.5.2.3 Power Distribution

The core power distribution is characterized using the following parameters:

- Normalized Axial Power, calculated as the ratio between the radially averaged power at a given axial core position divided by the core-average power.
- Normalized Maximum Power, given as the ratio between the maximum value of localized power divided by the core-average power.

4.5.2.4 Reactivity Coefficients

The fuel temperature (Doppler), moderator temperature (graphite blocks), and reflector temperature coefficients were calculated to evaluate the inherent safety performance of the reactor. The fuel temperature reactivity coefficient is the change in reactivity due to a change in temperature of the fuel pellets. The moderator temperature reactivity coefficient is the change in reactivity due to the change in the graphite fuel blocks temperature. The reflector temperature reactivity coefficient is the change in reactivity due to the change in temperature of the top, bottom, and radial graphite reflectors. The reactor does not have a coolant reactivity coefficient, since the helium primary coolant is gaseous and almost completely transparent to neutrons.

The reactivity coefficients depend on power, temperature, fuel burnup, and control rod position. The coefficients were calculated at several intervals throughout the cycle, based on the full-power condition with critical control rod position. The material densities were not perturbed in calculation of the reactivity coefficients, which is a conservative approximation since thermal expansion of the fuel, moderator, and reflector components would enhance negative reactivity feedback.

4.5.2.5 Shutdown Margin

The shutdown margin is evaluated as the difference between the required shutdown worth and the available worth of the RCSS. Analysis was performed to estimate the shutdown margins for various core states. These are evaluated against the required shutdown margin of 1% $\Delta k/k$ (i.e., 1000 pcm), which supports safe operation, testing and maintenance activities of the reactor facility.

The calculation of required shutdown worth incorporates the reactivity worths of the temperature defect, xenon decay, excess reactivity, and one ejected rod, as well as margin for uncertainties demonstrating compliance with PDC 26 (1), (3), and (4). The temperature defect is the difference in reactivity between stable, full-power operation and cold zero power conditions (assumed 27°C), due to the difference in core temperatures. The xenon decay worth is the difference in reactivity between the full-power condition with equilibrium xenon concentration, versus the zero-power condition in which xenon decays to zero. The excess reactivity is the reactivity of the core with all control rods removed. The reactivity worth of an ejected rod is incorporated to satisfy the requirement of PDC-26 that the reactivity control system shall provide “appropriate margin for malfunction,” such as an ejection or inadvertent withdrawal of a control rod, where the most reactive rod is assumed to malfunction. The nominal worth value of each of these components is calculated throughout the cycle.

Margin for uncertainties is incorporated by applying an assumed uncertainty factor to each nominal worth component to scale it up, thereby increasing the estimated required shutdown worth associated with each condition described by PDC-26. These assumed uncertainty factors, listed in Table 4-10, are conservative values typical of preliminary design and will be refined as detailed uncertainty quantification is completed.

Table 4-10 Assumed Uncertainties of Reactivity Worth Components for Shutdown Margin Analysis

Shutdown Worth Component	Assumed Uncertainty Factor
Excess reactivity	30%
Reactivity worth of ejected rod	30%
Temperature defect	25%
Xenon worth	25%

At any time in the cycle, the most limiting scenario for shutdown margin occurs during the maintenance or refueling condition corresponding to PDC-26 (4), which is assumed to occur at cold zero power core temperatures of 27°C, without xenon, and with the highest reactivity worth control rod removed fully out of the core. During the maintenance mode, the core is depressurized as described in Chapter 14, so there is no physical mechanism for a rod ejection to occur. Additionally, interlocks and administrative controls are in place on the RCSS so that inadvertent withdrawal cannot occur. The design and operating procedures essentially guarantee that a control rod malfunction cannot occur during this mode, so only a single control rod is assumed to be deliberately removed for maintenance or refueling access. For all other conditions described by PDC-26, a single control rod removal due to malfunction is assumed in the calculation of shutdown margin.

4.5.2.6 Kinetics Parameters

The prompt neutron lifetime and the effective delayed neutron fraction were obtained using the calculation methodology described in Section 4.5.2.2 and provided as inputs to the transient analysis described in Chapter 13. In addition, core power distribution, reactivity coefficients, xenon and samarium worth, and control rod worth are also provided as inputs to the transient analysis.

4.5.3 Nuclear Design Evaluation

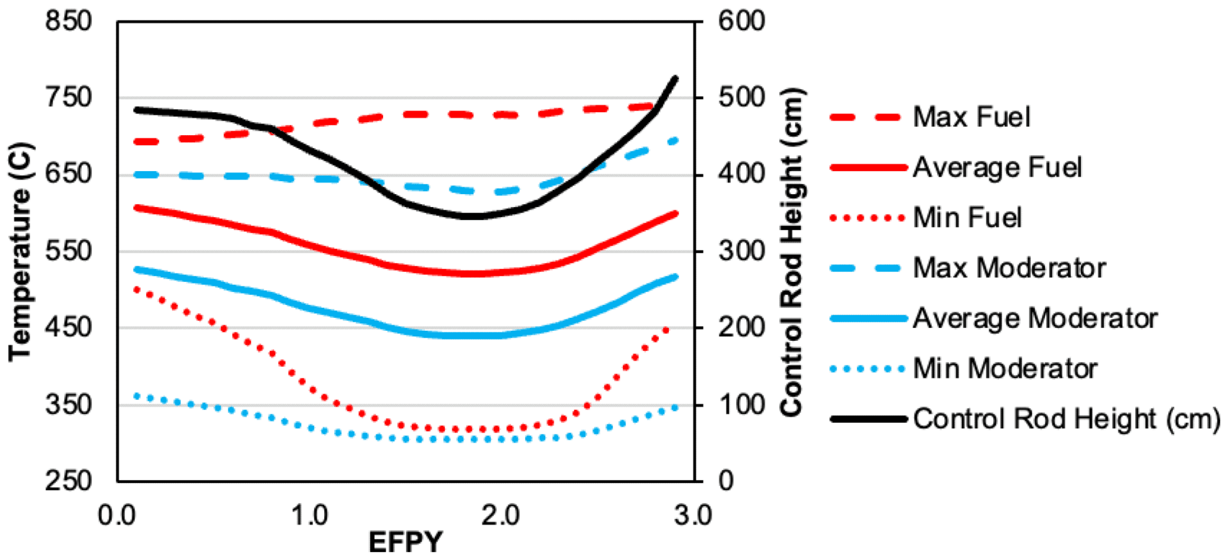
This section provides an evaluation of the nuclear design and describes how the nuclear design bases in Section 4.5.1 are met. In addition, this section also discusses nuclear design analyses that are provided as input to other parts of the design.

4.5.3.1 Temperature and Power Distribution

Using the coupled calculation methodology described in Section 4.5.2.2, the core power distribution and resulting fuel and moderator temperatures were calculated for the full-power condition with control rods positioned at their critical position for each depletion step. The critical control rod positions and the minimum, maximum, and average fuel and moderator temperatures throughout the cycle are shown in Figure 4-24.

As the cycle progresses and control rods are inserted, power shifts towards the bottom of the core, and the peak fuel temperature increases. The maximum control rod insertion depth and maximum fuel temperature occur near the middle of the cycle, when the excess reactivity is largest due to depletion of the BA. The maximum fuel temperature is < 750°C, well within the established temperature limits of TRISO particles and of the SiC matrix, as described in (Reference 4-2).

Figure 4-24 Critical Control Rod Position and Core Temperatures



The radially averaged axial power trend in the core at beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) is displayed in Figure 4-25. The trends are influenced by control rod insertion and the axial coolant temperature gradient. At BOC (0 effective full power years (EFPY)), the control rods are only slightly inserted since the burnable absorbers are the primary contributor to reactivity control, and the power is produced primarily at the top of the core. As the burnable absorber gradually depletes, the control rods are inserted further, and power is shifted towards the bottom half of the core at MOC (1.5 EFPY). At EOC (2.9 EFPY), the burnable absorbers are depleted and the control rods are fully withdrawn to account for the reduction in fissile material, resulting in a power profile that is similar to BOC. Throughout the cycle, the radial distribution of power at any given axial position is relatively uniform.

Figure 4-25 Radially Averaged Axial Power Trend

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Figure 4-26 shows the ratio of the maximum power anywhere in the core to the core average power density, throughout the cycle. The maximum local power directly relates to the thermal stress in the fuel pellet as discussed in Section 4.2.1. In compliance with PDC-10, the power and temperature results calculated with the coupled methodology ensure that SARRDL are not exceeded during normal operation and postulated events.

Figure 4-26 Local Power Peaking Factor Throughout Cycle

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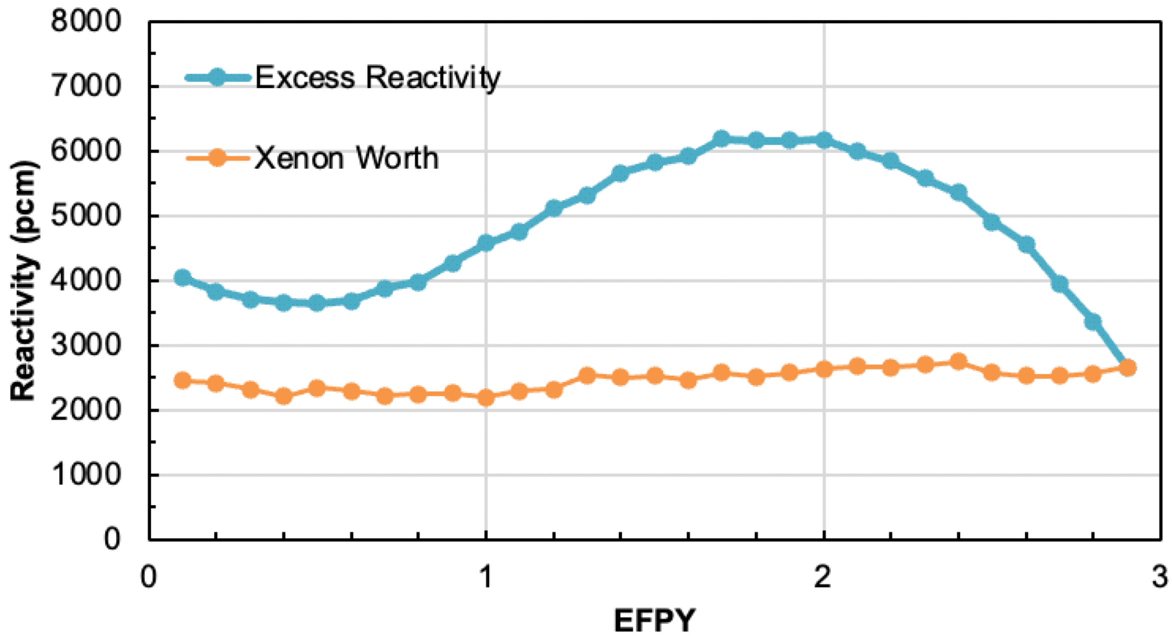
Neutron flux distributions will be verified during startup using ex-core detectors. These measurements will be compared against core design calculations to ensure that the core is operating as designed.

4.5.3.2 Reactivity Control

Figure 4-27 shows the excess reactivity and xenon worth over time for constant power operation. The excess reactivity in this figure is the core reactivity with control rods fully withdrawn and assuming no xenon buildup. The xenon worth is the reactivity worth of the equilibrium xenon concentration at the full nominal power. These values were calculated at each time step using a perturbation from the calculation methodology described in Section 4.5.2.2, so that the power history with full-power conditions and critical control rod position was reflected. The difference between the two curves shown in Figure 4-27 is the amount of reactivity that must be offset by insertion of the control rods. {{

}}a(4) The xenon worth is mostly stable throughout the cycle.

Figure 4-27 Excess Reactivity and Xenon Worth



The seven control rods of the RCSS are used for active reactivity control during operation, as well as to ensure that the required shutdown margins are met. Consistent with PDC-28, the RCSS has appropriate limits on the potential amount and rate of reactivity insertion to ensure the effects of postulated reactivity events can neither damage the safety-related elements of the primary coolant boundary or disturb the core and internals in a way that would impair the ability to cool the core.

Figure 4-28 shows the integral reactivity worth of all seven control rods at full-power conditions, as a function of rod insertion depth into the core. As B-10 in the BA depletes, neutron capture in the control rods increases due to less relative capture in the BA and the overall softer neutron spectrum. Consequently, the integral worth of the control rods increases from BOC to EOC.

Figure 4-28 Integral Reactivity Rod Worth

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As burnup increases over the cycle, U-235 depletes and Pu-239 slowly builds up, causing the effective delayed neutron fraction β_{eff} to decrease slightly, as shown in Figure 4-29. The discontinuities in the curve reflect statistical uncertainties of the calculation procedure described in Section 4.5.2.2.

Figure 4-29 Effective Delayed Neutron Fraction

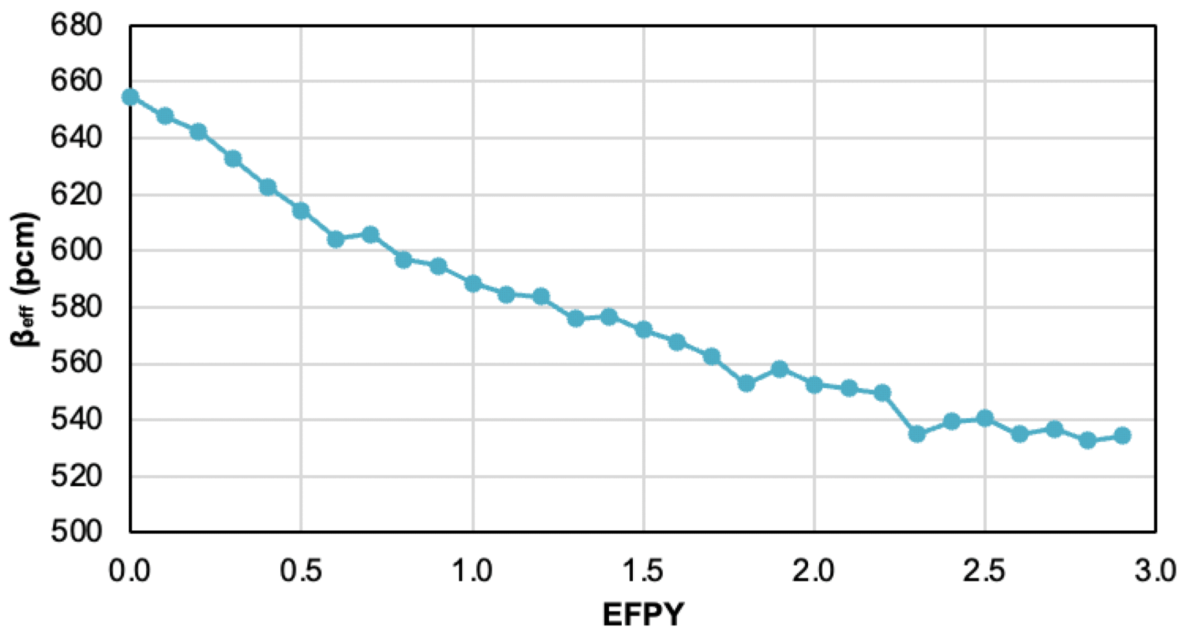

Figure 4-30 shows the worth of the single most reactive control rod, calculated as the reactivity difference from its full-power critical condition to fully withdrawn position. The peak value in the cycle is {{ }}a(4) including statistical uncertainty, which effectively guarantees that the reactor cannot be prompt critical regardless of the hypothetical speed of removal of the control rod.

Figure 4-30 Maximum Single Control Rod Worth
(error bars reflect 1σ statistical uncertainty)

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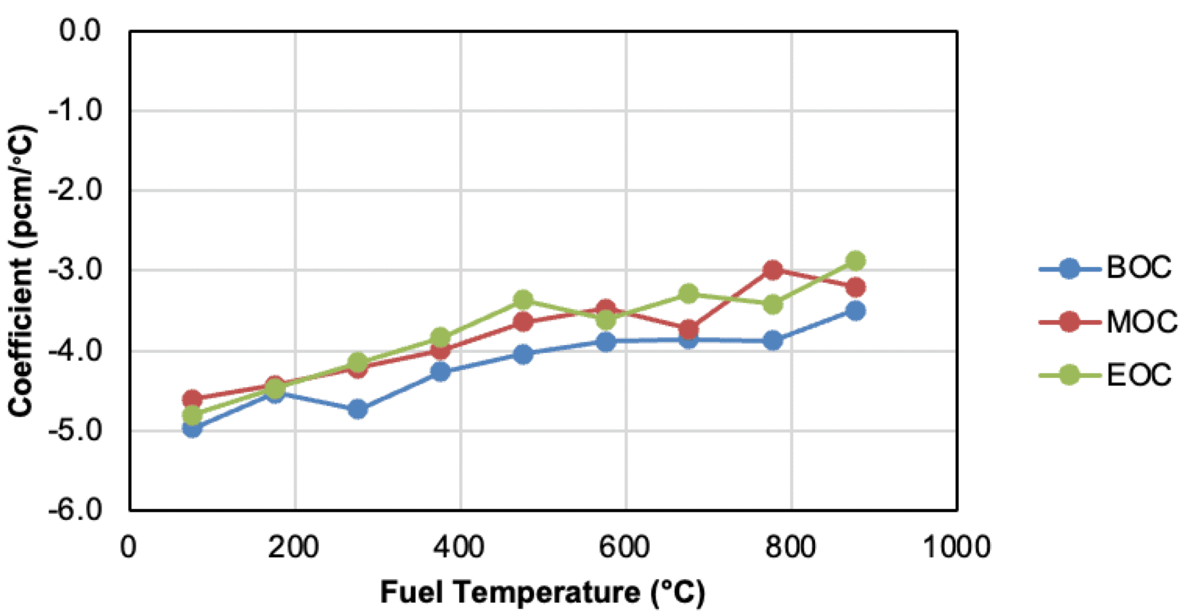
}}a(4)

4.5.3.3 Reactivity Coefficients

The following reactivity coefficients were calculated throughout the cycle: fuel temperature coefficient (FTC), moderator temperature coefficient (MTC), and reflector temperature coefficient (RTC). In compliance with PDC-11, the combined reactivity feedback effect is negative under all anticipated conditions.

Figure 4-31 shows the FTC at BOC, MOC, and EOC over the range of normal operating temperatures. The FTC is strongly negative over the entire cycle length, with only slight variation due to burnup. Fuel temperature reactivity feedback is largely driven by the Doppler broadening of capture resonances in U-238, of which the concentration does not change significantly during the fuel cycle.

Figure 4-31 Fuel Temperature Reactivity Coefficient



The MTC, shown in Figure 4-32, is strongly dependent on the burnup condition because of the effect of the BA. B-10 in the BA contributes negative reactivity feedback. Depletion of B-10 leads to more significant capture in graphite, which contributes positive feedback and causes the negative MTC magnitude to become smaller. While the MTC curve is relatively flat at BOC with temperature, towards EOC it becomes peaked around 500°C such that the MTC value may be slightly positive, based on uncertainties. However, the small positive MTC value is offset by the strongly negative FTC value, and the fuel temperature will increase before the moderator temperature as the heat originates from the fuel.

Figure 4-32 Moderator Temperature Reactivity Coefficient

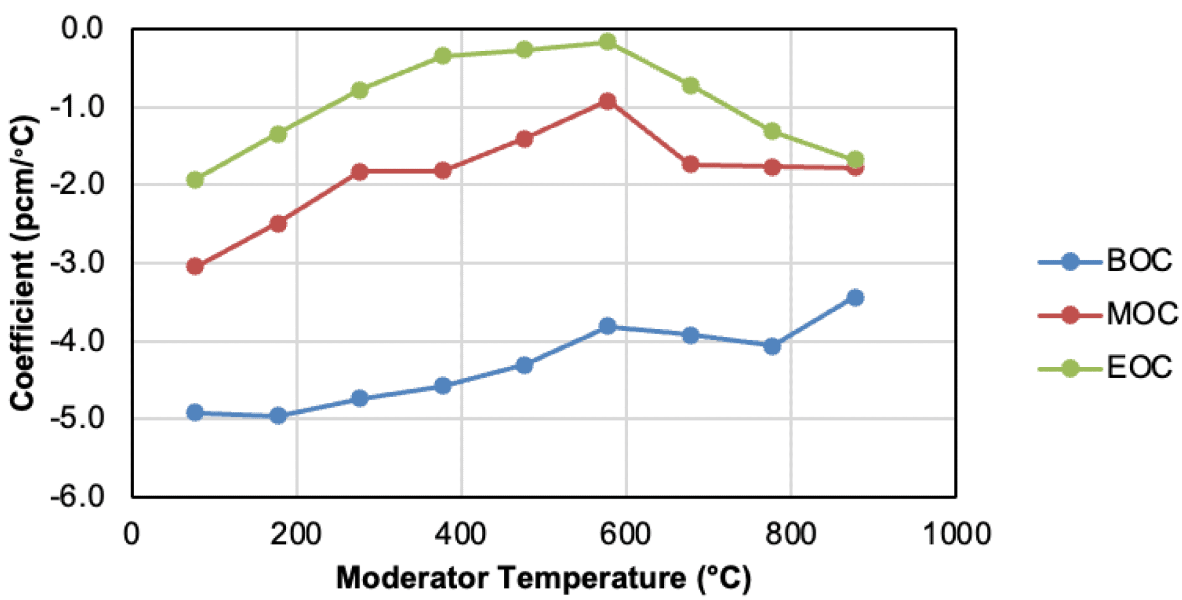
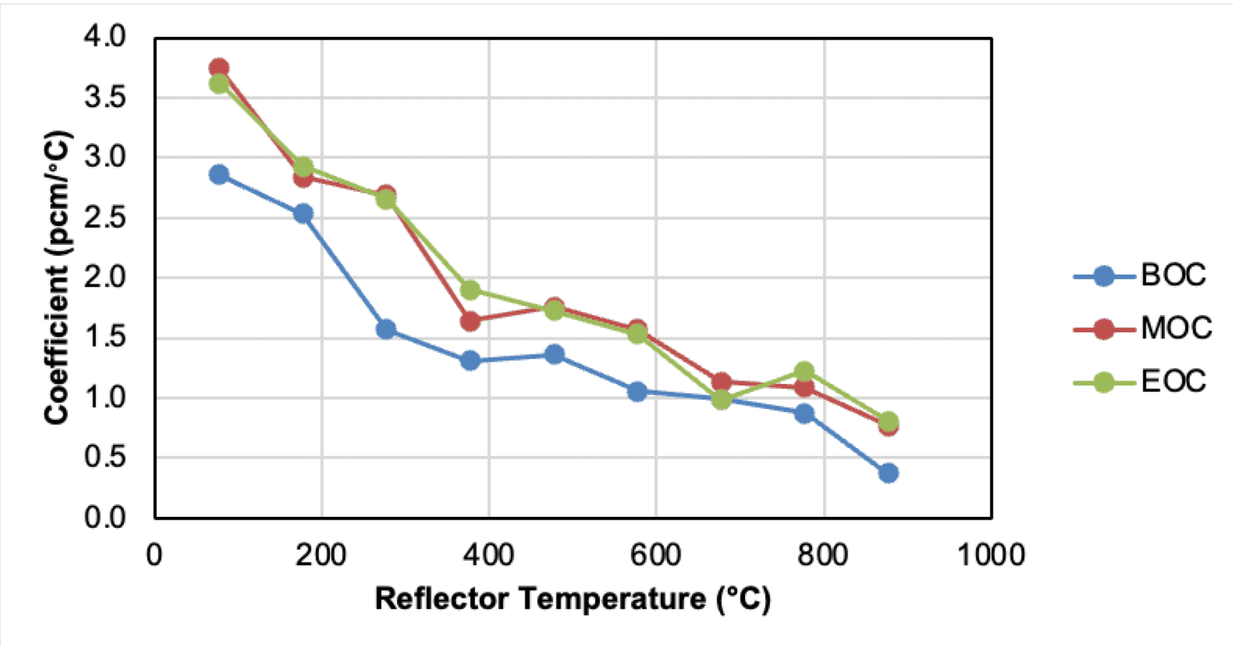


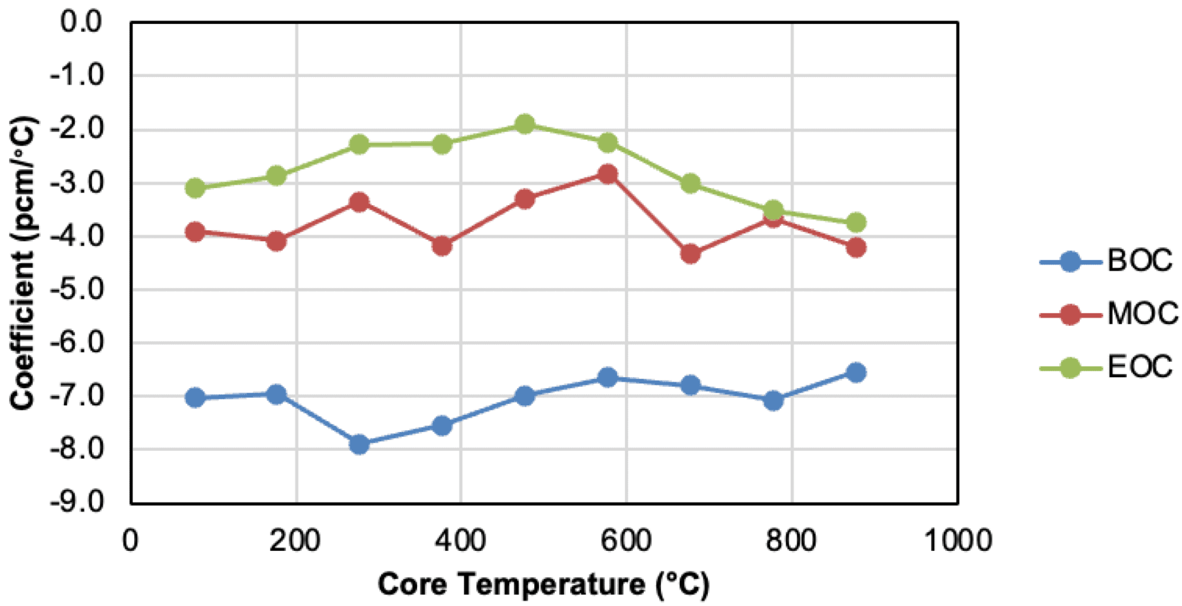
Figure 4-33 shows that the RTC is positive throughout the fuel cycle, but its magnitude decreases at higher temperatures. The positive RTC is the result of spectrum hardening at the periphery of the core due to increased reflector temperature, which reduces both neutron leakage and parasitic capture in the reflector graphite. These effects contribute positive reactivity feedback. However, the reflector temperature change is considerably delayed compared to fuel and moderator temperature feedbacks, and the positive RTC value is strongly outweighed by the combined negative FTC and MTC. Additionally, the calculated RTC value does not account for thermal expansion of the reflector, which will contribute negative reactivity feedback.

Figure 4-33 Reflector Temperature Reactivity Coefficient



The combined temperature coefficient (CTC) is the sum of FTC, MTC, and RTC. Figure 4-34 shows that the CTC remains strongly negative over all conditions, satisfying PDC-11. The CTC becomes less negative with burnup, largely driven by the MTC behavior shown in Figure 4-32, and is relatively insensitive to temperature.

Figure 4-34 Combined Temperature Reactivity Coefficient



4.5.3.4 Shutdown Margin

The credited means for complying with the PDC-26 requirements for shutdown and control of the rate of reactivity change are summarized in Table 4-11. As described in Section 4.2.3, the reactor is shutdown by fully inserting the control rods, which is ensured by tripping the safety-related RPS, establishing compliance with conditions (1), (3), and (4). In compliance with condition (2), any reactivity changes resulting from planned power level adjustment, including xenon effects, are passively compensated for by the inherent reactivity feedback provided by the nuclear design, shown in Section 4.5.3.3. Despite the thermal inertia of the graphite core, the annular fuel is directly cooled by the helium primary coolant and, as a result, it provides a quick thermal response. As fuel temperature changes, it contributes negative reactivity at a rate sufficient to ensure that SARRDL and the pressure boundary limits are not exceeded.

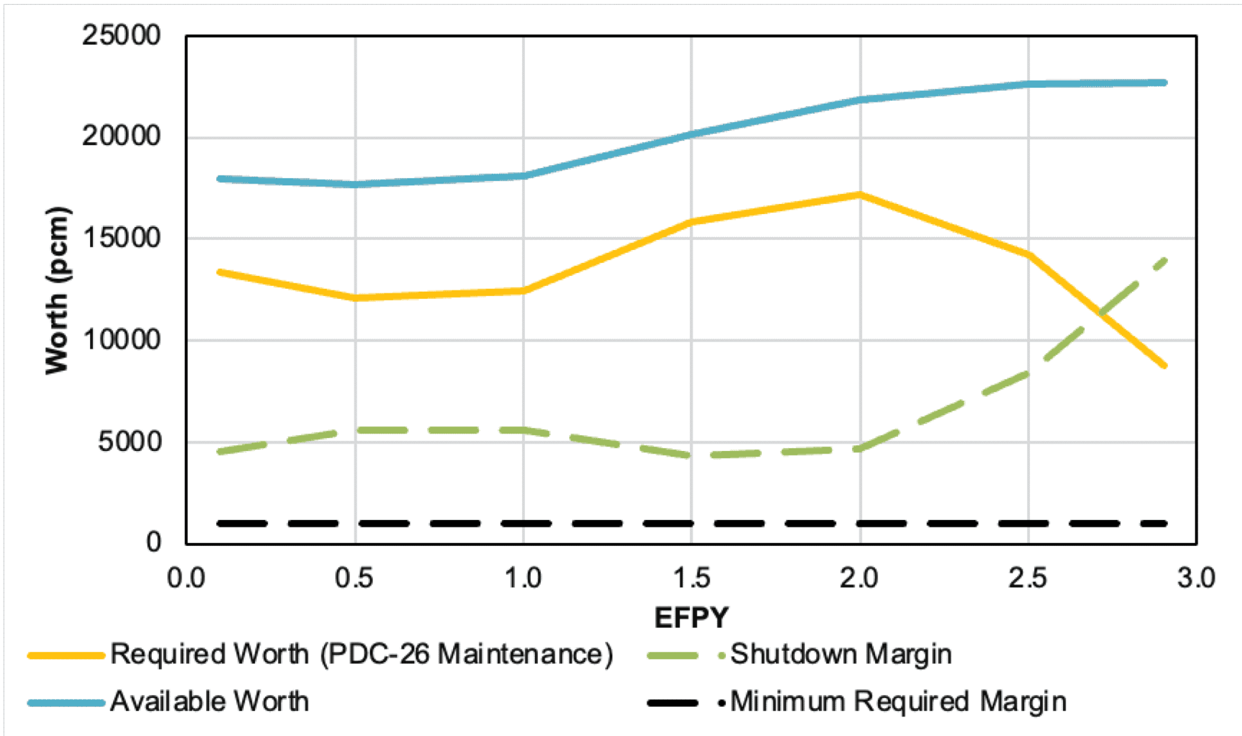
This effect is evident in the Pressurized Loss of Forced Cooling (PLOFC) event analyzed in Section 13.2.4, which considers a spurious helium circulator trip that rapidly stops the flow of helium primary coolant. For the first 20+ hours, no RPS trip point is triggered, and control rods do not get inserted. Nevertheless, the resulting increased fuel and graphite temperatures contribute negative reactivity feedback, bringing the reactor to a shutdown state near immediately after the initiation of the circulator trip. Additionally, the RCS control of helium circulator speed during normal operations provides a defense-in-depth means of controlling reactivity beyond the design of the fuel and core; corresponding analysis of the transient response will be provided in the OLA.

Table 4-11 PDC-26 Compliance

PDC-26 Conditions	**Credited Means for Compliance**
The reactivity control systems or means shall provide:	
(1) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the specified acceptable system radionuclide release design limits (SARRDL) and the reactor helium pressure boundary design limits are not exceeded, and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences.	Control Rods (RPS Trip)
(2) A means which is independent and diverse from the other(s), shall be capable of controlling the rate of reactivity changes resulting from planned, normal power changes to assure that the SARRDL and the reactor helium pressure boundary design limits are not exceeded.	Inherent Reactivity Feedback
(3) A means of inserting negative reactivity at a sufficient rate and amount to assure, with appropriate margin for malfunctions, that the capability to cool the core is maintained and a means of shutting down the reactor and maintaining, at a minimum, a safe shutdown condition following a postulated accident.	Control Rods (RPS Trip)
(4) A means for holding the reactor shutdown under conditions which allow for interventions such as inspection and repair shall be provided.	Control Rods (RPS Trip), Circulator Secured, Plant Conditions Established for Intervention

At any time during the cycle, the most limiting shutdown condition occurs during maintenance and refueling operations described by PDC-26 (4), when the core is at room temperature without xenon, and with the most reactive control rod deliberately removed for maintenance access. Figure 4-35 shows the calculated shutdown worth, including assumed uncertainties, as required for the limiting maintenance condition; the available worth of the six control rods (excluding the most reactive rod); and the resulting estimated shutdown margin throughout the cycle length. The estimated shutdown margin with conservatively large, assumed uncertainties remains well above the required minimum value of 1000 pcm.

Figure 4-35 Estimated Shutdown Margin



The shutdown margin values at BOC, MOC, and EOC are summarized in Table 4-12 for the limiting condition described by PDC-26 (4).

Table 4-12 Shutdown Margin Values

Parameter	BOC	MOC	EOC
Required Shutdown Margin	1,000	1,000	1,000
Actual Shutdown Margin (pcm)	4,500	4,300	13,900
Required Worth for Shutdown (pcm)	13,400	15,800	8,800
Shutdown Worth of Control Rods (pcm)	17,900	20,100	22,700

4.5.3.5 Nuclear Stability

The reactor is highly resistant to power oscillations owing to its low power density, large thermal inertia, compact size, and strong negative reactivity feedback. In compliance with PDC-12, the core is designed to ensure that any power oscillations have a low frequency that can be comfortably managed by the control systems. Detailed analysis will be included in the OLA.

4.5.3.6 Nuclear Design Analysis Inputs to Other Sections

Reactor kinetics parameters, including the neutron generation time and effective delayed neutron fraction, are calculated throughout the cycle length using the methodology described in Section 4.5.2.2. Table 4-13 shows the kinetic parameters for safety analysis, including prompt neutron lifetime, effective delayed neutron fraction (β_{eff}), and reactivity worth of xenon at BOC, MOC, and EOC. Along with power distributions, reactivity coefficients, xenon worth, and control rod worth, the kinetics parameters are provided as inputs to the transient analysis described in Chapter 13.

Table 4-13 Kinetics Parameters

Parameter	BOC	MOC	EOC
Prompt neutron lifetime (μs)	668	736	830
Neutron generation time (μs)	647	717	832
Effective delayed neutron fraction, β_{eff} (pcm)	648	572	534
Xenon worth (pcm)	2456	2527	2656

4.5.4 Operating Limits

The reactor core is designed such that the excess reactivity, negative reactivity feedback, and control rod worth are balanced throughout the cycle and across all operating conditions, including maintenance and refueling, such that the core always remains in a controllable state. Normal operational parameters are consistent with the performance described in this chapter and remain within the RPS trip limits described in Section 7.4.3. The RPS will initiate a reactor trip if the measured variables are exceeded. Administrative limits for the movement of each individual control rod will be provided in the OLA.

The reactor core is designed to physically prevent inadvertent addition of positive reactivity. As discussed in Section 4.2.2, the core configuration is fixed and compact, such that no additional fuel can be inserted. The high thermal inertia of the primarily graphite core prevents sudden overcooling. Interlocks are used so that only a single control rod can be withdrawn at a time and so that inserted control rods are deenergized during maintenance and refueling. Administrative controls to prevent the insertion of positive reactivity will be described in the OLA.

Relevant technical specifications that control important design features, limiting conditions for operation, and surveillance programs for the reactor core will be included in the OLA.

4.6 THERMAL-HYDRAULIC DESIGN

4.6.1 Description

The reactor generates thermal power primarily in the fuel. Additionally, some heat is deposited in the core graphite, reflector graphite, neutron shielding, and control rods, as the result of gamma and neutron interaction with those components. The heat is removed from the core by the downward flow of pressurized helium primary coolant. Helium enters through the inlet plenum region above the core and exits through the outlet plenum below the core.

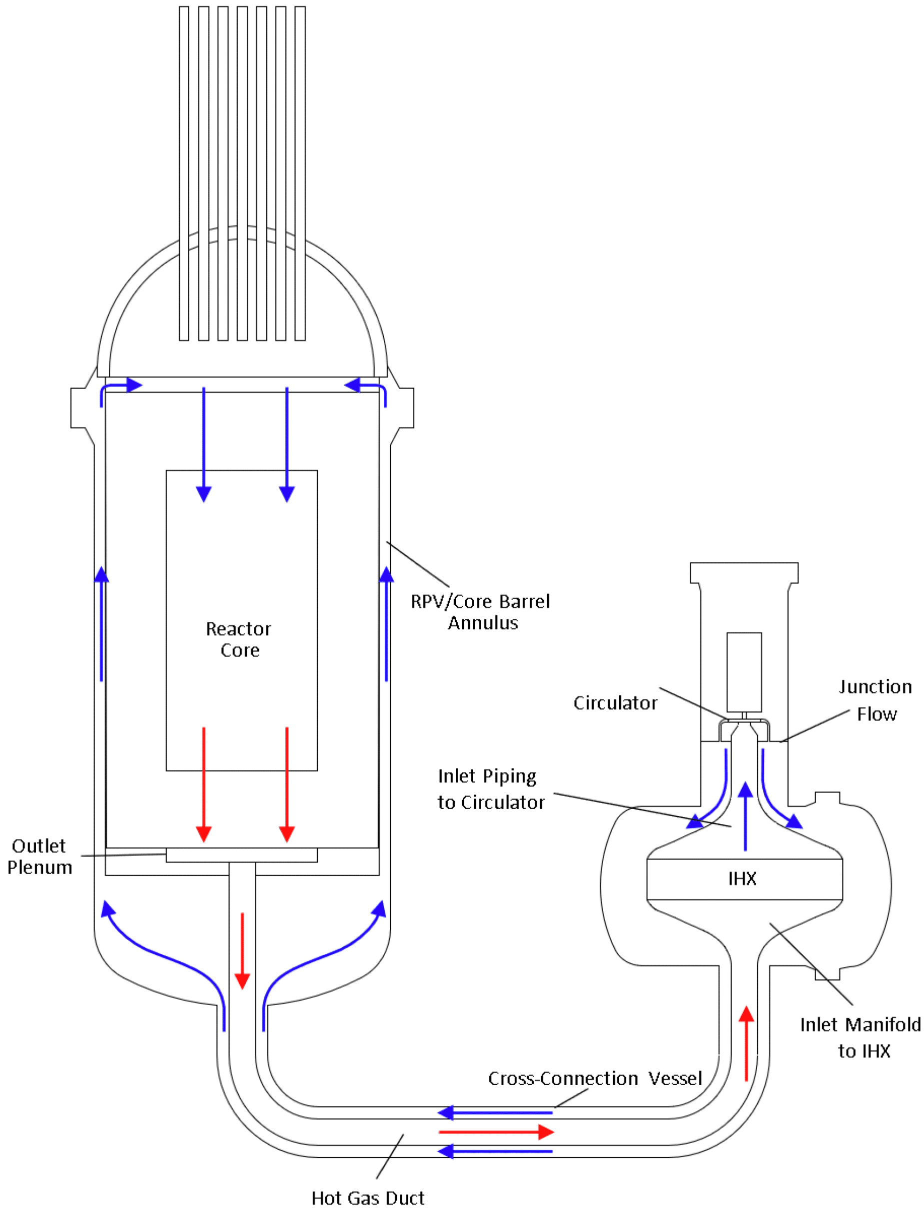
The bulk of the coolant entering the core upper plenum flows through the fuel channels in the fuel elements. The coolant then flows through the coolant channels in the bottom reflector, before being collected in the lower plenum and channeled to the hot gas duct. Besides the flow in the fuel channels, some coolant flow goes through the control rod channels and removes the heat generated in the control rods and the control blocks. There is also some bypass flow through the gaps between the various graphite columns forming the core as well as between the outer reflector layer and the radial shield and barrel elements.

The core thermal-hydraulic design, in conjunction with the nuclear and mechanical designs, ensures the requirements for maximum time-average fuel temperature, fuel pellet thermal stresses, core component stresses, core stability, and core pressure drop are met. To accomplish these objectives, the core is designed to provide nearly uniform flow to the fuel channels in the fuel blocks, provide flow for cooling of the control rods, minimize core bypass flow, and minimize core pressure drop. The features of the core assembly and individual core components as they relate to these objectives are discussed below.

4.6.1.1 Forced-Flow Cooling

The normal forced-flow path of helium primary coolant is shown in Figure 4-36. The cold helium exiting the helium circulator flows through the cross-connection vessel before entering the reactor pressure vessel. It flows up in the annulus space between the RV and the core barrel into the upper plenum, then downward through coolant channels in the top shield and top reflector, and downward through the fuel channels in the active core. As the flowing primary coolant is in direct contact with the annular fuel pellets, heat generated within the fuel pellets is transferred to the coolant via forced convection on both the inside and outside surface of the pellets. The primary coolant also removes the heat deposited in the graphite block via thermal radiation from the fuel pellets and from gamma and neutron interactions with those blocks. The primary coolant exits the reactor core through the coolant channels in the bottom reflector and lower shielding blocks. From there, it flows into the bottom plenum, where it is mixed to limit hot streaks before entering the hot gas duct assembly, which runs inside the cross-connection vessel. The helium flows through the hot gas duct to the inlet (helium hot leg) of the IHX. Finally, the flow from the exit (helium cold leg) of the IHX is channeled to the inlet of the helium circulator, which circulates the primary coolant back into the cross-connection vessel.

Figure 4-36 Schematic of Helium Forced Cooling Pathway



The helium flow paths in the various components it goes through are sized to limit the pressure drop to below the design limit of the helium circulator. The axial alignment of the cooling channels in the reactor core region, between the successive axial layers forming the core, is ensured by the graphite dowels described in Section 4.2.2. The overall core geometry and arrangement is assured by the core barrel, which provides the cylindrical outer boundary and limits the relative displacement of the reactor core components.

In the thermal-hydraulic analysis pertaining to the normal operating conditions, the average flow rate through the core region is determined based on the total reactor power generated, aiming at retaining a constant power to flow ratio throughout the range of normal operations. The annular fuel pellets and channel in the graphite blocks are sized such that the resulting flow split between the inner and outer fuel channels results in nearly equal inner and outer fuel surface temperatures under a wide range of reactor power and flow conditions. As described in Section 4.5, an analytical thermal hydraulics solver was coupled to the neutronics analysis to calculate primary coolant, moderator, and fuel temperatures alongside the core power distribution, throughout the cycle. Table 4-14 provides a summary of the thermal-hydraulic design and performance parameters.

Table 4-14 Summary of Thermal-Hydraulic Parameters During Normal Operation

Parameter	Unit	Value
Reactor power	MW	45.0
Power density	MW/m^3	{{ }}$^{a(4)}$
Primary coolant pressure	MPa	6.0
Number of fuel channels	-	{{ }}$^{a(4)}$
Average power per fuel channel	kW	{{ }}$^{a(4)}$
Primary coolant inlet temperature	°C	300.0
Helium heat capacity	J/kg-K	5,195.0
Average primary coolant temperature rise	°C	360.0
Total primary coolant flow rate	kg/s	24.1
Average fuel channel flow rate[1]	kg/s	{{ }}$^{a(4)}$
Average inner channel flow rate[1]	kg/s	{{ }}$^{a(4)}$
Average outer channel flow rate[1]	kg/s	{{ }}$^{a(4)}$
Peak fuel temperature[1]	°C	{{ }}$^{a(4)}$
Peak fuel temperature radial gradient	°C	{{ }}$^{a(4)}$

Notes:

1. Value does not account for bypass flow.

4.6.1.2 Passive Decay Heat Removal

The geometry and size of the Reactor Core System and its low power density allow passive transfer of the decay heat generated in the core to the RCCS when forced-flow cooling is absent. During postulated events where heat removal via forced flow of helium primary coolant is no longer available, the decay heat generated in the fuel pellets is transferred to the graphite components via conductive and radiative heat transfer through the stagnant helium. During primary coolant depressurization events, the same function is performed through radiative heat transfer alone. The increase in fuel temperatures during depressurized loss of forced cooling and pressurized loss of forced cooling is dampened by the large thermal mass of the graphite components, low power density of the core, and strong negative reactivity coefficients. The high thermal conductivity, compact configuration, and large surface area-to-volume ratio of the core, together with the uninsulated RV, provide for the removal of decay heat from the reactor core at a rate that is sufficient to maintain fuel temperatures well within the qualification envelope described in Section 4.2.1. Heat is transferred from the core via the RV to the reactor cavity and RCCS via a combination of natural convection, conduction and thermal radiation, and the heat is ultimately removed by the RCCS. This passive heat transfer pathway is illustrated in Figure 4-37.

During normal operations, the RCCS provides forced flow of water through an array of pipes that surround the reactor cavity. Cold water flows down through the outer pipes, located against the Citadel building wall, and up through the inner pipes, removing heat from the reactor cavity. In passive mode, the water inside the RCCS pipes is stagnant and will heat to the point of boiling; the resultant steam is released through an atmospheric vent. The system can operate passively for more than 72 hours without needing to be refilled with water. The RCCS is detailed in Section 6.3, and analysis of thermal-hydraulic performance during postulated events will be included in the OLA.

Figure 4-37 Passive Heat Removal Path During Conduction Cooldown

{{

}}[a(4)]

4.6.2 Design Basis

The PDC used to develop design criteria are provided in the approved TRs referenced in Section 3.6.1. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC were identified as applicable to the reactor thermal hydraulic design:

- **PDC-10,** “Reactor design”
- **PDC-12,** “Suppression of reactor power oscillations”
- **PDC-16,** “Containment design”
- **PDC-34,** “Passive residual heat removal”

4.6.3 System Evaluation

In compliance with PDC-10, the reactor core, reactor internals and heat removal systems associated with the thermal-hydraulic design of the reactor system have appropriate margins to ensure that SARRDL are not exceeded during any condition. Due to the annular design of the fuel pellets, heat transfer occurs on both the inner and outer surface of the annulus. {{

}}[a(4)] Additionally, the TRISO particles are well within their acceptable temperature limit, effectively resulting in an extremely small probability of failure. The high thermal conductivity of the graphite components ensures that changes in helium primary coolant flow due to maldistribution (such as a blocked channel), core deformation, or component cracking do not lead to excessive fuel temperatures. In the event that a fuel channel is blocked, heat would be transferred to the graphite via radiative and conductive heat transfer, and removed either by forced convection of the primary coolant in the adjacent fuel channels, or by conduction to the adjacent graphite components in situations where forced flow is lost. The large thermal mass of the core graphite components ensures that short-duration postulated events do not lead to rapid increases in fuel temperatures. Relevant thermal-hydraulic parameters, including the peak fuel temperature and peak fuel temperature gradient, are summarized in Table 4-14. These features and analyses demonstrate compliance with PDC-10 with respect to thermal-hydraulic design. Detailed thermal-hydraulic performance analyses will be included in the OLA.

In compliance with PDC-12, the thermal-hydraulic design of the reactor system ensures that instability phenomena that could lead to exceeding SARRDL are prohibited. The helium primary coolant itself does not contribute to core reactivity and is maintained in its gaseous phase during all postulated operating conditions and events. Additionally, the low power density, high thermal inertia of the graphite core and strongly negative reactivity coefficients ensure that the reactor always approaches its stable state asymptotically, ensuring that no power oscillations occur. Additionally, the pace at which bulk graphite temperature can change is much slower than the response time of the RCSS. This demonstrates compliance with PDC-12 with respect to the thermal-hydraulic design. Analysis results demonstrating the inherent stability of the reactor will be provided with the OLA.

The functional containment provided by the ceramic matrix TRISO fuel controls the release of radioactivity to the environment, as described in Section 6.2. In compliance with PDC-16, the thermal-hydraulic design ensures that the fuel temperatures are maintained within limits during long-duration postulated events. The high surface area to volume ratio of the reactor core combined with the high thermal inertia and conductivity of the graphite ensure adequate heat removal from the fuel under all postulated event conditions.

The reactor core defines and maintains a forced-flow path of helium primary coolant during normal operations as described in Section 4.6.1.1. During postulated events where forced-flow cooling is lost, the reactor core relies on the inherent characteristics of the graphite core to passively conduct and radiate heat to the RCCS, including during helium depressurization events, as described in Section 4.6.1.2. Passive heat removal from the core during postulated events is evaluated in the analysis described in Chapter 13. These features demonstrate compliance with PDC-34 with respect to thermal-hydraulic design.

4.6.4 Testing and Inspection

Reactor primary coolant temperatures, flow, and core power are to be periodically monitored during operations to ensure these parameters are within specified limits. Testing and inspection plans will be provided with the OLA.

4.7 REFERENCES

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USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 5 - REACTOR COOLANT SYSTEMS

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
CCV	cross connection vessel
HGD	Hot Gas Duct
HPS	Helium Purification System
HRS	Helium Recovery System
HTS	Heat Transport Systems
IHX	Intermediate Heat Exchanger
MHA	maximum hypothetical accident
PCHE	printed circuit heat exchanger
PCMS	Primary Coolant Make-up System
PDC	Principal Design Criteria
PSAR	Preliminary Safety Analysis Report
RCCS	Reactor Cavity Cooling System
RCS	Reactor Control System
SARRDLs	specified acceptable radionuclide release design limits
SSCs	Structures, Systems and Components
TESS	Thermal Energy Storage System
TRs	Topical Reports
UMSs	Unit Monitoring Systems
VS	Vessel Systems
VSD	variable speed drive

CHAPTER 5 REACTOR COOLANT SYSTEMS

This chapter provides the design bases, descriptions, and functional analyses of the reactor coolant systems.

5.1 SUMMARY DESCRIPTION

The Heat Transport Systems (HTS) use helium as the primary heat transfer medium to move heat from the reactor core to the Intermediate Heat Exchanger (IHX). The heat is transferred through the IHX from the helium to the secondary heat transport medium, molten salt. The HTS form a coolant loop comprised of the Hot Gas Duct (HGD), the IHX and the helium circulator. The helium circulator provides forced convective flow through the HTS. The HTS' components are entirely housed within the pressure boundary formed by the Vessel Systems (VS). The reactor core assemblies and core internals also contribute to guiding the primary coolant and completing the HTS' loop.

The HTS interface with the secondary loop via the IHX. Within the secondary loop, the Thermal Energy Storage System (TESS) primarily consists of the molten salt pumps, molten salt piping, cold and hot molten salt storage tanks, molten salt itself and the associated instrumentation and control equipment. The cold molten salt returns from the steam generation building and accumulates in the cold salt storage tank. During research reactor operation, the cold molten salt is pumped through the molten salt piping, through IHX, and fed into the hot salt storage tank. As the salt goes through the IHX, heat is transferred from the primary coolant to the salt. The salt from the hot salt storage tank can then be pumped into the steam generation building where the heat is utilized and the salt returned to the cold salt storage tank at its nominal cold temperature.

The HTS additionally include systems to purify and replace the helium primary coolant. The Helium Purification System (HPS) removes impurities and radionuclides from the primary coolant during operation. The HPS allows for controlled depressurization of the pressure boundary for refuel and maintenance. The Primary Coolant Make-up System (PCMS) supplies fresh helium to the reactor both at startup, and during normal operation. Helium lost through leakage is replenished and pressure is adjusted as needed to accommodate temperature changes.

The HTS interface with the Reactor Internals System, Reactor Core System, Water Services Systems, Reactor Protection Systems, Electrical Power Systems, Vessel Systems, Thermal Energy Storage Systems, Reactor Control Systems, Instrumentation & Controls System, and the Citadel and Unit Auxiliary buildings. Further description and design basis for the HTS, TESS, HPS, and PCMS are described for each subsystem in this chapter.

5.2 PRIMARY COOLANT SYSTEM

5.2.1 Summary Description

The HTS, forming the primary coolant loop, are responsible for transporting heat from the reactor core to the secondary coolant loop, which uses molten salt as coolant. The HTS consist of a single, closed-cycle primary loop containing the HGD, IHX, and helium circulator.

5.2.2 Functions, Classification and Design Criteria

5.2.2.1 Safety Functions

The HTS do not perform any safety functions. Cooling of the fuel and reactor core during and following postulated events, such as loss of off-site power, does not rely on the HTS and are discussed in Chapter 6.

5.2.2.2 Non-Safety Functions

The HTS' non-safety functions are the following:

- Provide the means of transporting energy generated by the reactor core that is deposited into the primary coolant.
- Provide the means to exchange energy from the primary coolant loop to the secondary coolant loop. This is achieved by transferring the energy from the primary coolant to the secondary coolant through the IHX.
- Provide the means to transport energy from the primary coolant to the secondary coolant for normal power generation conditions as well as during both startup and shutdown transitions.
- Control the circulation of the primary coolant to match the heat generation for normal power generation as well as during both startup and shutdown transitions.
- Transfer decay heat from the reactor core to the secondary coolant under normal reactor shutdown conditions.

5.2.2.3 System Classification

The HTS do not perform any safety functions. Therefore, the HTS are classified as non-safety related.

5.2.3 Design Basis

The Principal Design Criteria (PDC) used to develop design criteria are provided in the approved Topical Reports (TRs) referenced in Section 3.6.1. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC are applicable to the HTS:

- PDC-10, "Reactor design"
- PDC-12, "Suppression of reactor power oscillations"
- PDC-14, "Reactor helium pressure boundary"
- PDC-15, "Reactor helium pressure boundary design"
- PDC-26, "Reactivity control systems"
- PDC-30, "Quality of reactor helium pressure boundary integrity included reducing fluid ingress"
- PDC-31, "Fracture prevention of reactor helium pressure boundary"
- PDC-44, "Structural and equipment cooling"
- PDC-45, "Inspection of structural and equipment cooling systems"
- PDC-46, "Testing of structural and equipment cooling systems"
- 10 CFR 20.1406, "Minimization of Contamination" (Reference 5-1)

Note regarding PDC-14, PDC-15, PDC-30 and PDC-31: {{ }}$^{a(4)}$, the IHX is in practice a helium pressure retaining component. It is therefore a self-imposed requirement that the engineering design of the IHX shall follow guidance from these design criteria. This is discussed further in Section 5.2.6.

5.2.4 Design Description

The HTS consist of a single closed-cycle primary loop, containing the HGD in series with the IHX and the helium circulator. To ensure that the HTS can fulfill their functions, the system components are housed entirely within the pressure boundary provided by the VS and discussed in Section 4.3.

Together with the VS and reactor core system, the HTS form the entire primary coolant flow path, as shown in Figure 5-1, where:

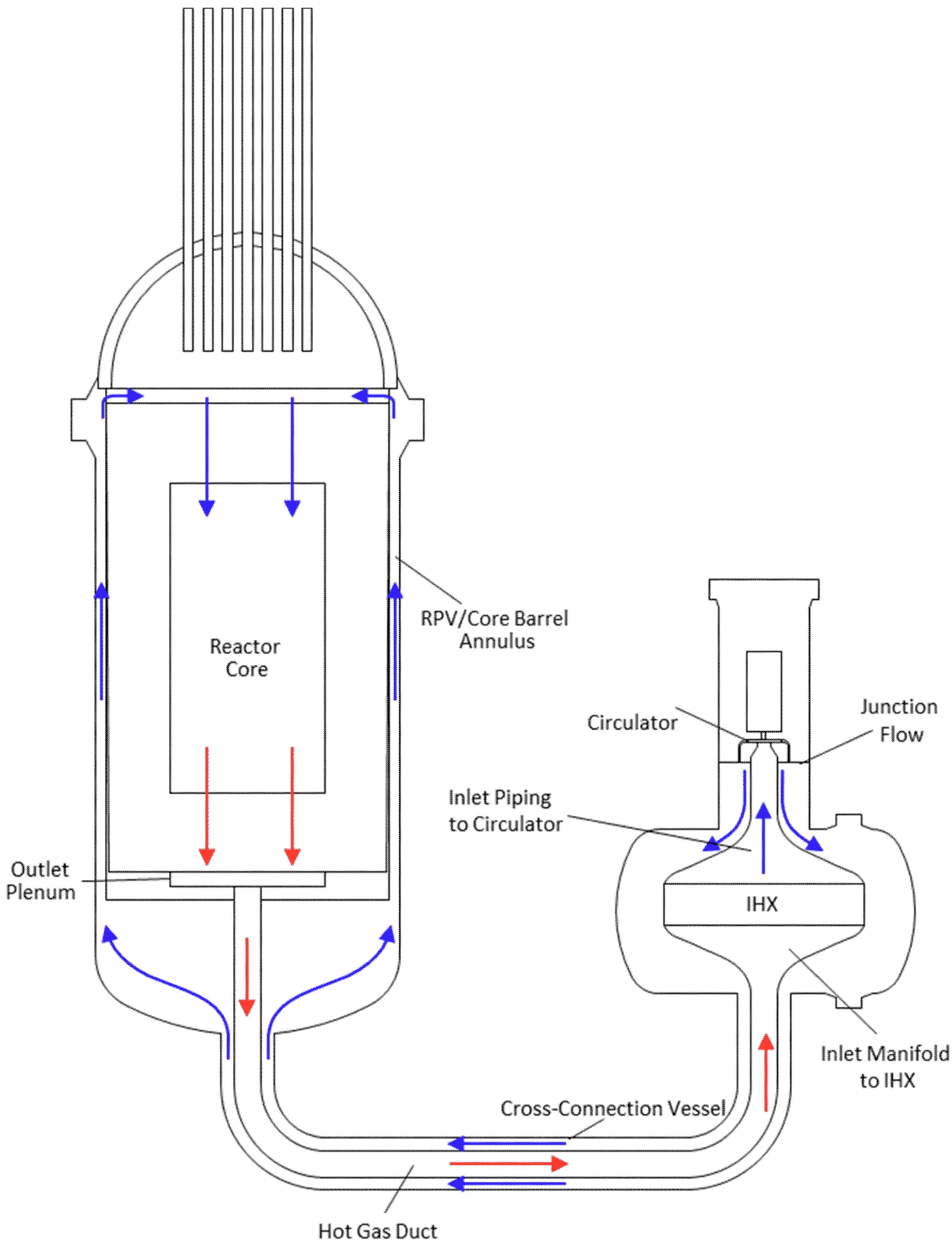
- Hot primary coolant exits the reactor core through the outlet plenum, flows into the HGD and through to the IHX, where heat is transferred to the secondary coolant.
- Cold primary coolant exiting the IHX enters the helium circulator through its impeller region.
- Upon exiting the helium circulator, the cold primary coolant flows through the space between the IHX and the IHX vessel, and into the annular space between the HGD and the cross-connection vessel (CCV).
- Upon reaching the reactor vessel region, the cold primary coolant flows up through the annulus between the core barrel and the reactor pressure vessel.
- Finally, the primary coolant reaches the top of the core region and flows back down through the core region where it heats up as it goes down.

Additional discussion of the reactor core and VS is provided in Chapter 4. Nominal specifications for the primary loop are provided in Table 5-1.

Table 5-1 Primary Loop Specifications

Parameter	Nominal Value	Unit
Thermal power	45	MW
Primary loop pressure	6	MPa
Primary loop cold leg temperature	300	°C
Primary loop hot leg temperature	660	°C
Primary mass flow rate	24.1	kg/s
Primary heat transfer medium	Helium	N/A
Number of primary loops per unit	1	N/A
Number of hot legs (HGD) per loop	1	N/A
Number of cold legs (CCV annulus) per loop	1	N/A
Primary coolant volume	{{ }}$^{a(4)}$	m
Primary coolant mass	{{ }}$^{a(4)}$	kg

Figure 5-1 Primary Loop Flow Diagram



5.2.4.1 Primary Coolant

The primary coolant is pressurized helium. Helium is transparent to neutrons and has no impact on core reactivity through means of neutron moderation or parasitic absorption. It serves no shielding function and the coolant itself is not credited as a functional containment barrier. The coolant is not required for cooling in postulated events, unlike traditional light-water reactors. Finally, due to its low-density gaseous state, there is no need to consider elevations or pressure differentials associated with coolant head.

Helium alone is chemically inert and poses no risk of corrosion. However, corrosion can still occur if sufficient amounts of impurities accumulate in the helium. The HPS, discussed in Section 5.4, is responsible for cleaning the helium to maintain sufficiently low impurity levels to preclude corrosion within the primary loop under normal conditions. Limits on maximum allowable helium impurity levels during operation will be provided in the Operating License Application.

A very small fraction of radionuclide inventory may be released from the fuel over the course of normal operations. Activation of impurities possibly contained in the helium and graphite blocks also results in production of radionuclides outside of the fuel. This will lead to a small amount of circulating activity within the primary coolant. The HPS, discussed in Section 5.4, is responsible for purifying the helium to ensure that low circulating activity is maintained to ensure the specified acceptable radionuclide release design limits (SARRDLs) are satisfied.

In addition to the HPS keeping the circulating activity within the SARRDLs, leakage of helium (and associated circulating activity) through the pressure boundary during normal operations is controlled through appropriate sealing technology on all flanged vessel connections. This is discussed further in Section 4.3. Makeup supply of helium to offset potential helium leakage during normal operations is discussed in Section 5.5. More detailed assessment of radionuclide transport will be provided in the Operating License Application.

The sampling system, discussed in Section 5.4, provides a means of sampling helium from the primary coolant loop to ensure that the impurity and radionuclide concentrations are within acceptable limits. The acceptable limits and necessary sampling frequency and methodology will be detailed in the Operating License Application.

5.2.4.2 Hot Gas Duct

The HGD is a passive piping component, designed in accordance with ASME BPVC section VIII (Reference 5-2) with additional consideration for temperature creep. It {{

}}$^{a(4)}$ connects the core outlet plenum and core support plate to the IHX helium inlet header. It is coaxially located within the CCV to form an annular set of pipes. Hot helium flows inside the HGD, while cold helium flows in the annular section between the HGD and CCV. {{

}}$^{a(4)}$ These features allow for radial and axial thermal expansion of the HGD within the CCV.

The HGD is {{ }}[a(4)] insulated to minimize heat transfer between the cold and hot helium flows, ensuring the primary and secondary coolant temperatures remain within the desired ranges and maintaining balance-of-plant stability. {{

}}[a(4)]

Limited bypass flow between the hot and cold helium is anticipated but does not adversely impact safe operation. This results in a very slight increase of the core temperatures but does not have any notable impact on the core performance characteristics or safety margins.

5.2.4.3 Intermediate Heat Exchanger

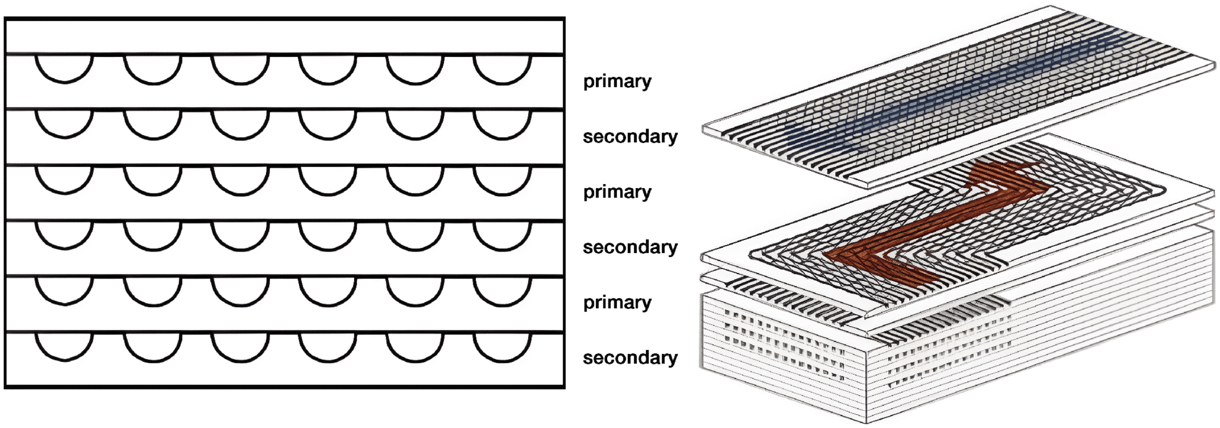
The IHX is a printed circuit heat exchanger (PCHE) and is a passive component that is responsible for transferring heat between the primary and secondary coolants. It is designed in accordance with ASME BPVC section VIII (Reference 5-2) with additional consideration for temperature creep. {{ }}[a(4)]

The IHX is comprised of a collection of heat transfer cores welded into a single monolithic block. The inlet and outlet headers for the secondary and primary coolants, which are fully seal welded to the monolithic block, connect with the piping from the primary and secondary coolant loops. The secondary coolant used is a molten salt, further discussed in Section 5.3. The IHX receives hot helium from the HGD and sends cold helium to the circulator. Hot and cold molten salt lines penetrate and interface with the IHX vessel, while the IHX is mounted on supports installed within the IHX vessel.

As illustrated in Figure 5-2, each heat transfer core is a collection of plates with microchannels etched onto their surfaces. The plates are stacked together and diffusion bonded to form a solid block of flow channels. The primary and secondary sides are separated by the plate walls. {{

}}[a(4)] Should a leak occur between plates, the helium is at a higher pressure than the salt and helium will leak into the salt side. The radiological consequences of a helium-to-salt leak are minimized through operationally maintaining primary coolant activity below the SARRDLs, as discussed in Section 5.2.4.1.

Figure 5-2 IHX PCHE Heat Transfer Core Channel Illustration



Since the IHX is fully surrounded by cold helium flowing back from the circulator towards the core, it is {{ }}$^{a(4)}$ insulated to limit undesirable heat transfer from the hot IHX regions to the surrounding cold helium. {{ }}$^{a(4)}$ The molten salt is drained from the IHX to a drainage tank if the temperature drops below the design setpoint, before reaching its crystallization onset temperature. Decay heat removal when the molten salt is drained is discussed in Chapter 6.

Heat transfer surface area in the IHX is designed with both a fouling factor and corrosion allowance, driven by the molten salt, to account for operational degradation during the life of the IHX. Layout and support of the IHX considers the helium flow distribution through the cores, management of thermal expansion {{ }}$^{a(4)}$

Given the presence of a small amount of impurities and radionuclides in the primary coolant, some elements (primarily metallic fusion products) may plate out in the cold regions of the IHX helium channels. Trace permeation of the tritium contained in the primary coolant through the IHX plate walls into the secondary coolant may also occur. As with potential helium leakage, this is mitigated by maintaining the primary coolant activity below the SARRDLs. Detailed assessment of these mechanisms as a part of radionuclide transport and how it affects contamination control and worker dose will be detailed in the Operating License Application.

5.2.4.4 Helium Circulator

The helium circulator is an electrically driven, radial flow blower that establishes the required helium pressure rise in the cycle to overcome the system's flow resistance up to the maximum nominal mass flow rate. {{ }}$^{a(4)}$

Rotating components are designed to operate subcritical, meaning it does not have any natural harmonic vibration frequencies for which excessive vibration would become a design consideration in the operating speed range. This allows for unrestricted operation over the full operating range of the facility. The impeller is mounted directly onto the shaft of the motor rotor, {{

}}a(4) The impeller is surrounded by a missile containment shield to prevent damage to other components, including the helium pressure boundary, in the event of catastrophic impeller failure.

The helium circulator speed is controlled through a variable speed drive (VSD), allowing effective operation over the entire range of expected helium mass flow rates needed to support normal operations, startup, shutdown decay heat removal and/or other plant transitions. The VSD is controlled by signals coming from the Reactor Control System (RCS), as detailed in Chapter 7. The circulator also has internal instrumentation and sensors to continuously monitor its condition to ensure safe and reliable operation within the established design parameters.

{{

}}a(4)

5.2.4.5 Summary of Sensors

A summary of sensors is provided in Table 5-2. Information obtained from these sensors and parameters will be used by the RCS and overall instrumentation and control (I&C) systems to control the reactor so as to keep the primary and secondary coolants' temperatures, pressures and flow rates within their design limits.

Table 5-2 Summary of Heat Transport System Sensors

Measurement	Purpose	Location	Control
Circulator helium inlet pressure	Pressure drop and rise calculation for IHX and circulator performance point	{{ }}a(4)	{{ }}a(4)
Circulator helium outlet pressure	Pressure rise calculation for circulator performance point	{{ }}a(4)	{{ }}a(4)
Circulator helium inlet temperature	Equipment protection	{{ }}a(4)	{{ }}a(4)
Circulator motor stator winding temperature	Equipment protection	{{ }}a(4)	{{ }}a(4)
Circulator speed	Limit generation of missile impacts from rotating equipment; Equipment protection - circulator	{{ }}a(4)	{{ }}a(4)
IHX helium inlet temperature	Reactor outlet temperature – IHX performance; Equipment protection – molten salt maximum & minimum temperature	{{ }}a(4)	{{ }}a(4)
IHX molten salt inlet temperature	Monitor minimum molten salt temperature	{{ }}a(4)	{{ }}a(4)
IHX molten salt outlet temperature	Monitor maximum molten salt temperature	{{ }}a(4)	{{ }}a(4)

5.2.5 Operational Overview

The HTS utilize the circulator to move the primary coolant through the HGD and IHX before returning it to the reactor core. During startup operations, the circulator is brought up to speed before taking the reactor critical. The helium circulator and its VSD are controlled by the RCS.

During power operations, reactor power is adjusted by changing the primary coolant flow rate, control rod position and secondary coolant flow rate. The primary coolant flow rate is controlled by adjusting the speed of the helium circulator. The relationship between speed and mass flow is determined from the steady-state performance data and will be detailed in the Operating License Application.

During system shutdown, the HTS provide the normal method of decay heat removal {{

}}a(4) Circulator speed is actively controlled to maintain the desired core and coolant temperatures during shutdown.

During and following postulated events, the HTS is not credited for cooling the reactor but may be utilized if available.

5.2.6 System Evaluation

PDC-2

The HTS is non-safety related, therefore only components which could potentially fail during an earthquake and potentially impact the function of safety related Structures, Systems and Components (SSCs) are designed to meet Seismic Category I requirements. In addition to protective barriers against internally generated missiles, fracture speeds of rotating components of the helium circulator impeller are physically limited by the capabilities of the system. These features, in conjunction with the seismic design of the reactor, demonstrate conformance with PDC-2.

PDC-3

The HTS is designed and located in the Citadel building so as to minimize the probability and effect of fires and explosions. All materials used as part of the HTS are low combustible materials. These design features, in conjunction with the fire protection system (Section 9.3), provide assurance that the HTS demonstrates conformance with PDC-3.

PDC-4

The design of rotating equipment in the helium circulator includes missile containment features to mitigate any consequences of internally generated missiles that could result in the consequential failure of the helium pressure boundary. In addition to the physical barriers against missile generation, the maximum speed of the circulator is such that fragmentation of the impeller, being the main source of missile generation, is precluded. These features demonstrate conformance with PDC-4.

PDC-10

The HTS is designed to circulate the primary coolant within the pressure boundary, which includes a pathway for coolant flow through the HPS. This purification system maintains a low circulating activity that is below the SARRDLs. Means of sampling the primary coolant to measure radionuclide content are provided to ensure satisfaction of the SARRDLs. This demonstrates conformance with PDC-10 during normal operations. The HTS does not perform any safety function, and active circulation of the primary coolant to the HPS during postulated events is not credited.

PDC-12

The helium circulator controls the flow rate of primary coolant in the primary loop, thus influencing the temperature of the reactor, including the fuel, during normal operations. Design of its subcomponents ensures safe, reliable and controllable operation within the established design parameters, allowing adjustment of flow conditions as necessary to prevent power oscillations. These features limit the operational reactor power oscillations such that the SARRDLs are not exceeded and demonstrate conformance to PDC-12.

PDC-14

{{ }}[a(4)] the IHX is in practice a helium pressure retaining component and as such is designed, fabricated and tested in accordance with the ASME BPVC VIII code (Reference 5-2), with additional consideration for temperature creep. The IHX components are fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The IHX materials will be qualified for use and have sufficiently high fracture toughness to reduce the likelihood of crack propagation. {{

}}[a(4)] These features demonstrate conformance to PDC-14.

PDC-15

{{ }}[a(4)] the IHX is in practice a helium pressure retaining component and as such is designed, fabricated and tested in accordance with the ASME BPVC VIII code (Reference 5-2), with additional consideration for temperature creep. The HTS' pressure and temperature are monitored during normal operations to ensure that design conditions of the IHX are not exceeded. The IHX does not perform any safety function and while not required to mitigate postulated events, the design pressure and temperature of the IHX are sufficient to withstand all overpressure conditions discussed in Section 4.3. These features demonstrate conformance to PDC-15.

PDC-26

In addition to the discussion of PDC-12, the circulator design ensures that when it is de-energized, the impeller speed will ramp down at a rate sufficient to ensure core temperatures rise quickly enough to satisfy negative reactivity insertion requirements. This ramp down rate also satisfies requirements on the VS's temperatures. These features demonstrate conformance to PDC-26.

PDC-30

{{ }}[a(4)] the IHX is in practice a helium pressure retaining component and as such is designed, fabricated and tested in accordance with the ASME BPVC VIII code (Reference 5-2), with additional consideration for temperature creep. The primary loop pressure is higher than the secondary loop pressure and potential leakage, although unlikely, will be from the primary side of the IHX to the secondary side. Appropriate monitoring is included in the secondary loop to detect primary coolant leakage into the secondary loop, as discussed in Section 5-3. These features demonstrate conformance to PDC-30.

PDC-31

Refer also to the discussion addressing PDC-15 for design conditions considered in the IHX. {{ }}a(4) the IHX is in practice a helium pressure retaining component and as such is designed to prevent failure due to material creep, fatigue, thermal, mechanical, and hydraulic stresses through adherence to the ASME BPVC VIII code (Reference 5-2), with additional consideration for temperature creep.

Corrosion and chemical attack on the primary side of the IHX is precluded through the use of high purity helium, discussed in Section 5.2.4.1. Corrosion on the secondary side from the molten salt coolant is accounted for in the design and the channel wall thickness is sized to support the IHX lifetime requirement.

These features demonstrate conformance to PDC-31.

PDC-44

The non-safety related HTS provide the means by which normal operational reactor power and decay heat during shutdown are removed. This heat is transferred to the secondary loop coolant, which serves as the heat sink, provided that temperatures in the molten salt are above the design setpoint. If active secondary cooling is not available, decay heat is removed by means discussed in Chapter 6. These features demonstrate conformance to PDC-44 for the HTS' non-safety functions.

PDC-45

The HTS provides means for periodic inspection of applicable components, with details to be provided in the Operating License Application. {{

}}a(4) the HTS demonstrate conformance with PDC-45.

PDC-46

The HTS' components permit appropriate periodic functional testing to assure component integrity and operability to perform non-safety functions, with details to be provided in the Operating License Application. These features demonstrate conformance to PDC-46 for the HTS' non-safety functions.

10 CFR 20.1406

The HTS are designed, to the extent practical, to minimize facility and environmental contamination and to facilitate eventual decommissioning, consistent with 10 CFR 20.1406 (Reference 5-1). {{

}}a(4)

5.2.7 Testing and Inspection

While the IHX microchannels separate the high-pressure helium from the lower-pressure molten salt, they are inherently not inspectable with current techniques. However, the IHX is not credited as part of the pressure boundary and is located within an inspectable pressure boundary created by the IHX vessel (Section 4.3) and the secondary coolant {{ }}a(4) (Section 5.3).

Descriptions of any tests and inspections of the HTS will be provided with the application for the operating license.

5.3 SECONDARY COOLANT SYSTEM

5.3.1 Summary Description

The secondary coolant loop receives heat from the HTS through the IHX and transfers it through molten salt storage tanks and eventually to connected systems for electrical power generation, process heat, or both (cogeneration) during normal operation. Solar salt (40% KNO_3 + 60% $NaNO_3$ by weight) is used as the heat transfer fluid. This is the same salt and associated technology as that used in operating Concentrated Solar Plants.

The molten salt heat transfer fluid is contained primarily within the TESS and, in much smaller quantities, within the HTS IHX. {{

}}a(4) the TESS primarily consists of molten salt cold and hot storage tanks, molten salt pumps, molten salt piping, molten salt itself, heat tracing, and the associated instrumentation and control equipment.

Figure 5-3 Molten Salt System Boundary Interfaces and Process Flow Diagram

{{

}}a(4)

5.3.2 Functions, Classification and Design Criteria

5.3.2.1 Safety Functions

The TESS is not credited for performing any safety functions or mitigation of postulated events.

5.3.2.2 Non-Safety Functions

The non-safety functions of the TESS include:

- Transfer the thermal energy from the primary coolant in the IHX to the energy conversion systems connected to the TESS.
- Receive and return molten salt between the VS and energy conversion systems.
- Provide thermal energy storage capability.
- Separate primary coolant in the reactor from energy conversion systems.

5.3.2.3 System Classification

The TESS {{ }}$^{a(4)}$ are classified as non-safety related. None of the secondary loop components are credited during postulated events. Components which could potentially fail during an earthquake and impact the functionality of safety related SSCs are designed to Seismic Category I to prevent such a condition.

5.3.3 Design Basis

Although the TESS {{ }}$^{a(4)}$ are classified as non-safety related, the secondary loop components have been designed to remain consistent with the following PDC where applicable, in addition to the generally applicable PDC discussed in Section 3.6.1.4:

- PDC-10, “Reactor design”
- PDC-14, “Reactor helium pressure boundary”
- PDC-15, “Reactor helium pressure boundary design”
- PDC-30, “Quality of reactor helium pressure boundary integrity including reducing fluid ingress”
- PDC-31, “Fracture prevention of reactor helium pressure boundary”
- PDC-32, “Inspection of reactor helium pressure boundary”
- PDC-45, “Inspection of structural and equipment cooling systems”
- PDC-46, “Testing of structural and equipment cooling systems”
- PDC-64, “Monitoring radioactivity releases”
- 10 CFR 20.1406, “Minimization of Contamination” (Reference 5-1)

5.3.4 Design Description

To fulfill its essential functions, the secondary loop TESS includes storage tanks, transfer piping, and salt pumps. Supporting equipment for the system includes a drain subsystem, heat tracing, and associated instrumentation. A simplified process flow diagram of the TESS is provided in Figure 5-3, and the key design parameters are provided in Table 5-3.

Table 5-3 Key Design Parameters of the Molten Salt TESS

Parameter	Nominal Value	Unit
Thermal Duty	45	MWth
Secondary Loop Salt Flow Rate	{{ }}[a(4)]	kg/s
Secondary Loop Pipe Operating Pressure	{{ }}[a(4)]	kPa
Secondary Storage Vessel Operating Pressure	Near Ambient	N/A
Secondary Loop Hot Leg Temperature	{{ }}[a(4)]	°C
Secondary Loop Cold Leg Temperature	{{ }}[a(4)]	°C
Primary Loop Pressure[1]	6,000	kPa

{{ }}[a(4)]

The TESS is designed in accordance with ASME-TES-1 (Reference 5-3). However, the interface between the IHX itself and the IHX vessel represents a code break between the IHX design code (ASME BPVC Section VIII, described in Section 5.2) (Reference 5-2) and the codes applied {{ }}[a(4)] outside of the vessel (ASME-TES-1 {{ }}[a(4)] The specific code break transition and interface requirements will be provided in the Operating License Application. Constructions materials demonstrate compatibility with radiation control, corrosion resistance, and thermal stresses associated with normal operation and postulated events. Performance analysis and evaluation of these components' satisfaction of PDC will be provided with the Operating License Application.

Electric heat tracing, insulation, salt condition heaters, and a monitoring system ensure that the salt within the piping and storage vessels is maintained above solidification temperatures during all operating conditions.

Molten salt pumps {{ }}[a(4)] provide the necessary flow of salt to the connected components.

The molten salt piping to the IHX includes a drain vessel, which is able to contain the molten salt from the IHX and associated piping when draining is required (e.g., during maintenance, emergency shutdown, etc.).

The RCS uses measurements {{ }}[a(4)] to support reactor operations. The hot salt line includes a thermowell where a thermostat is housed, on a section of piping outside of the IHX cavity.

The Unit Monitoring Systems (UMSs) monitors the salt condition of the molten salt piping system. The hot salt line includes an interface outside of the Citadel building where a radiation monitor measures the activity in the hot salt.

Control and shut-off valves provide control and isolation functions for the TESS to supply salt at the required flow rates. As an RCS function described in Section 7.3, in the event that leakage occurs from the higher-pressure primary helium coolant to the lower-pressure secondary molten salt lines, the UMSs activity monitoring systems will detect the activity, with the RCS interfacing with the {{ }}[a(4)] valves to mitigate the release of activated primary coolant through the TESS pipelines and eventually to the molten salt storage tanks. Although these {{ }}[a(4)] valves are not necessary for radionuclide retention and meeting the SARRDLs, they are intended to limit hypothetical energetic helium release through the salt lines and potential propagation through the TESS.

To minimize salt chemical reactivity and corrosion on the equipment, low molten salt impurity levels are maintained through the use of high-purity salt and operation at temperatures below the threshold for thermal decomposition. The TESS provides sampling points where representative molten salt samples are collected for analysis.

The TESS is designed to isolate the IHX on the molten salt side from the rest of the TESS system, following a reactor shutdown for maintenance or refueling, in accordance with ASME TES-1 minimum maintenance requirements (Reference 5-3).

5.3.5 Operational Overview

The TESS provides the capability to collect, transfer and supply heat between the reactor and the connected energy conversion systems. It conveys the heat produced in the reactor through circulating molten salt driven by independent pumps in the molten salt storage tanks.

During normal operation, cold molten salt is pumped from the cold salt tank to the IHX where it is heated {{ }}[a(4)] as heat is exchanged from the primary to the secondary coolant systems. The hot salt is then transferred from the IHX to the hot salt tank for storage. When heat is needed by one of the energy conversion systems connected to the TESS, hot molten salt is pumped from the hot molten salt tank to those systems {{ }}[a(4)] (heat exchange to provide process heat, to generate steam, or dumped), after which it is returned to the cold molten salt tank.

The molten salt hot leg coming from the IHX includes a molten salt drain tank that allows drainage of the salt contained in the IHX and piping section leading to it, when required. Control and shut-off valves regulate the flow of salt and provide isolation functions, {{ }}[a(4)] If a condition causes cold salt tank temperatures to rise or cold molten salt flow is lost, the reactor will trip on high reactor inlet temperature as described in Chapter 7 to stop reactor operations. Appropriate instrumentation, controls, and sampling points monitor and maintain desired system operational states, including temperature, flow rate, chemical reactivity, and radiation levels.

5.3.6 System Evaluation

The TESS {{ }}[a(4)] are non-safety related and are not credited for mitigating the consequences of postulated events. Therefore, the secondary coolant loop components are not required to maintain fuel temperatures within allowable limits or to support the fuel's functional containment and safety functions. During postulated events, including pressurized and depressurized loss of forced cooling events, the secondary loop is taken offline, and heat removal is achieved passively by the Reactor Cavity Cooling System (RCCS). The path for passive heat removal is described in Chapter 4.6, the RCCS design and its heat removal functionality are presented in Chapter 6, and the heat removal analysis under limiting postulated events are provided in Chapter 13. Beyond the functional containment of radionuclides provided by the fuel, as discussed in Chapter 4, operation of the HTS containing the primary coolant ensures that radionuclide concentrations remain below the SARRDLs. This is discussed in Section 5.2.4.1.

Accordingly, radionuclide transfer to the molten salt, primarily through tritium permeation across the IHX, will also remain within these limits. While the molten salt coolant could become contaminated by the higher-pressure helium and its impurities in the event of minor primary coolant leakage to the secondary coolant, helium is chemically inert with salt, and the resulting contamination would also remain within the SARRDLs.

Substantial leakage is unlikely due to the diffusion-bonded solid steel plates separating helium and salt in the IHX. {{ }}[a(4)] Therefore, the transfer of primary coolant to the secondary coolant system is an unlikely propagation pathway for radionuclide release to the environment. If rupture were to occur, the radiological consequences are bounded by the limiting depressurized loss of forced cooling event described in Section 13.1.3. The consequences of salt ingress to the primary loop are not expected to be consequential; salt ingress is also discussed in Section 13.1.3.

<u>PDC-1</u>

The TESS {{ }}[a(4)] are designed, fabricated, erected, and tested, as applicable, in accordance with relevant codes and standards, including the provisions of {{ }}[a(4)] ASME-TES-1 (Reference 5-3), {{ }}[a(4)] and ASME B31.3 (Reference 5-5) to assure satisfactory performance of important to safety functions and associated quality requirements. This demonstrates conformance with the requirements of PDC-1.

<u>PDC-2</u>

The TESS {{ }}[a(4)] do not perform any safety functions. Therefore, only components which could potentially fail during an earthquake and impact the functionality of safety related SSCs are designed to Seismic Category I. This demonstrates conformance with the requirements of PDC-2.

PDC-3

The TESS {{ }}a(4) are designed and located to minimize the probability and effect of fires and explosions through the use of low combustible materials, where possible, and physical separation. These design features, in conjunction with the fire protection system discussed in Section 9.3, demonstrate compliance with PDC-3.

PDC-4

As discussed in Section 5.2, internal missiles generated from components near the molten salt lines and helium pressure boundary, such as the helium circulator, are prevented by shielding features and the physical limitations of the systems. Hypothetical high-energy events from helium leaking into the TESS are mitigated by the isolation valves. The TESS is also operated at a significantly lower pressure range than the primary coolant and is designed, fabricated, erected, and tested, as applicable, in accordance with relevant codes and standards to limit the potential for failure from dynamic forces. These features demonstrate compliance with PDC-4.

PDC-10

As operation of the primary coolant ensures radionuclide concentrations remain below the SARRDLs, radionuclide transfer to the molten salt through IHX permeation and minor primary coolant leakage will also remain within the SARRDLs. Associated with the {{ }}a(4) valves and radiation monitoring of salt activity in the TESS hot salt line, these features demonstrate conformance with the requirements of PDC-10.

PDC-14

The molten salt components {{ }}a(4) are fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture. The piping extending from the IHX headers {{ }}a(4) are constructed, as applicable, in accordance with relevant codes and standards, including the provisions of ASME-TES-1 (Reference 5-3), {{ }}a(4) This demonstrates conformance with the requirements of PDC-14.

Additionally described in Section 5.2, the IHX materials will be qualified for use and have sufficiently high fracture toughness to reduce the likelihood of crack propagation. Should failure occur, it is more likely to result in delamination between plates of the IHX, which does not result in secondary and primary coolant mixing. The IHX is designed, fabricated and tested in accordance with rules of the ASME BPVC VIII code (Reference 5-5).

PDC-15

While not required to mitigate postulated events, the flow rate and temperature of the TESS {{ }}a(4) are sufficiently monitored during normal operations to ensure that design conditions of the IHX and associated pressure boundary components are not exceeded. The molten salt TESS, IHX, {{ }}a(4)

{{ }}[a(4)] are designed, fabricated, erected, and tested, as applicable, in accordance with relevant codes and standards, including the provisions of ASME BPVC VIII (Reference 5-2), ASME-TES-1 (Reference 5-3), ASME BPVC III, Division 5 (Reference 5-4), and ASME B31.3 (Reference 5-5). This demonstrates conformance with the requirements of PDC-15.

PDC-30

The molten salt piping extending from the IHX headers {{ }}[a(4)] are constructed, as applicable, in accordance with relevant codes and standards, including the provisions of ASME-TES-1 (Reference 5-3), {{ }}[a(4)] These features, including UMSs activity monitoring and actuation of the {{ }}[a(4)] valves in the case of helium leakage through the IHX, demonstrate conformance with the requirements of PDC-30.

PDC-31

As discussed in Section 5.2, the molten salt components within the helium pressure boundary are designed in accordance with relevant codes and standards, {{ }}[a(4)] as applicable, with sufficient margin to ensure nonbrittle behavior and to minimize the probability of rapidly propagating fracture under operating, maintenance, testing, and postulated event conditions.

Corrosion and chemical attack on the molten salt secondary loop are mitigated through maintenance and monitoring of salt impurity levels, employing corrosion-resistant materials of construction, and design of channel walls with appropriate thickness. These design features demonstrate conformance with the requirements of PDC-31.

PDC-32

{{ }}[a(4)] Those components can readily be inspected and tested, with relevant procedures detailed as part of the Operating License Application.

PDC-45

The secondary loop components permit appropriate periodic inspection of applicable components to assure the integrity and capability of important to safety components forming the pressure boundary; this capability is also described in PDC-32. Details of the inspection plans will be provided as part of the Operating License Application. This demonstrates conformance with the requirements of PDC-45.

PDC-46

The secondary loop components permit appropriate periodic functional testing to assure component integrity and operability to perform non-safety functions. Details of the testing plans will be provided as part of the Operating License Application. This demonstrates conformance with the requirements of PDC-46.

PDC-64

Radiation monitoring of salt activity in the TESS hot salt line is provided through the UMSs. This demonstrates the TESS's conformance with the requirements of PDC-64.

10 CFR 20.1406

The TESS is designed, to the extent practical, to minimize facility and environmental contamination and to facilitate eventual decommissioning, consistent with 10 CFR 20.1406 (Reference 5-1).

5.3.7 Testing and Inspection

Further discussion of the technical specifications, including the testing and inspection plans, will be provided in the Operating License Application.

5.4 HELIUM PURIFICATION SYSTEM

5.4.1 Summary Description

The HPS removes particulate, chemical, and radioactive impurities from the primary helium coolant to maintain required purity and performance. It processes helium during normal operation and depressurization for maintenance, returning purified helium to the system. The HPS supports reactor efficiency, protects equipment, and helps control radiological releases by ensuring a clean and stable helium coolant inventory.

5.4.2 System Functions

5.4.2.1 Safety Functions

No safety functions are credited to the system. The system provides only defense in depth by maintaining helium purity and controlling radiological releases from the purification equipment.

5.4.2.2 Non-safety Functions

The primary non-safety function of the HPS is to continuously bleed helium from the primary coolant loop, process it to eliminate particulates and chemical or radioactive impurities, and return purified helium to the loop, effectively maintaining a level or circulating activity in the HTS below the SARRDLs.

Additionally, the HPS processes primary coolant during reactor depressurization prior to refueling or maintenance, preparing it for transfer to the Helium Recovery System (HRS).

These non-safety functions are supported by the following responsibilities:

- Processing helium withdrawn from the primary coolant loop.
- Purifying the primary coolant.
- Removing chemical, particulate, and radioactive impurities from the helium contained in the HTS.
- Maintaining required primary coolant conditions.
- Maintaining pressure boundary integrity to ensure retention of the primary coolant inventory.
- Preserving the system's ability to perform purification functions.
- Controlling radiation associated with HPS equipment.

5.4.2.3 System Classifications

The HPS is classified as non-safety related.

5.4.3 Design Bases

Although the HPS is classified as non-safety related, the system and its components contained within the helium pressure boundary have been designed to remain consistent, where applicable, with the following PDC, in addition to the general PDC discussed in Section 3.6.1.4:

- PDC-10, "Reactor design"
- PDC-13, "Instrumentation and Control"
- PDC-14, "Reactor helium pressure boundary"
- PDC-15, "Reactor helium pressure boundary design"
- PDC-30, "Quality of reactor helium pressure boundary integrity including reducing fluid ingress"
- PDC-32, "Inspection of reactor helium pressure boundary"
- PDC-60, "Control of releases of radioactive materials to the environment"
- PDC-64, "Monitoring radioactivity releases"
- 10 CFR 20.1406, "Minimization of Contamination" (Reference 5-1)

5.4.4 Helium Purification System Design Description

The HPS draws helium from and returns helium to the HTS through supply and return lines equipped with double isolation valves that are attached to the IHX vessel. The entire HPS is designed to maintain pressure boundary integrity and define a clear interface between the HPS and the primary coolant pressure boundary. Most of the HPS equipment is located in a dedicated room within the Citadel building to allow maintenance access during reactor operation while protecting it from external hazards. Purification components are sized to accommodate the highest anticipated radionuclide inventory associated with normal operation. Monitoring is provided to detect leakage or abnormal conditions, supporting timely isolation of the HPS from the HTS, and periodic maintenance and consumables replacement are planned to maintain purification. Details will be provided in the Operating License Application.

The schematic of the HPS is shown in the Figure 5-4. Helium entering the HPS passes through a pressure-reducing interface upstream of the purification equipment. Downstream of this interface, the HPS includes components used to process and circulate helium for impurity removal and return to the HTS. As a result, credible pipe rupture or component failure within the HPS does not lead to high-energy events. Such failures cannot cause pipe whips or adverse interactions with nearby safety related systems.

Figure 5-4 Helium Purification System Schematic Illustration

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}}a(4)

The HPS consists of several modules, as shown in Figure 5-4. Module-1 consists of a high temperature absorber and a high temperature filter. Helium from the cold side of the IHX vessel first passes through the high temperature absorber, {{ }}a(4) containing a charcoal-filled removable cartridge with a dust chamber that removes condensable metallic fission products from the primary coolant side stream. The helium then flows through the high temperature filter, which removes carbon dust and particulates using replaceable filter cartridges with removable dust storage capacity.

Module-1 is designed, fabricated, and tested in accordance with ASME BPVC III, Division 5 (Reference 5-4).

Module-2 contains five components arranged in series, including a pre-heater, oxidizer, recuperator, helium cooler, and molecular sieve dryers. The pre-heater raises the helium temperature to support oxidation in a CuO-based oxidizer, where CO, H_2, tritium, and residual oxygen are converted to CO_2 and water. The helium is subsequently cooled through a recuperator and water-cooled heat exchanger to condense moisture, and oxidized impurities are removed in parallel molecular sieve dryer beds, enabling continuous operation with alternating regeneration.

Module-3 is composed of the low temperature heat exchanger, liquid nitrogen heat exchanger and low temperature absorption bed module. It removes noble gases and any trace contaminants by liquifying these impurities at cryogenic temperature. The impurities, such as CH_4, Kr and Xe plus N_2, and Ar if present, are removed by adsorption in a charcoal bed. This module will be regenerated by the regeneration module.

Module-4 consists of a helium compressor which provides the motive force to circulate helium through the purification train and back to the HTS. The HPS is designed to withstand normal cyclic loads of the reactor and to continue operation during normal operating transients, with automatic trip and valve closure occurring in case of postulated events, such as loss of forced cooling.

The regeneration module, not shown in detail in Figure 5-5, regenerates saturated adsorbent beds within the HPS. The module operates intermittently and functions only when the low temperature absorber or dryer requires regeneration.

During regeneration, impurities removed from the low temperature absorber are processed and managed within the HPS. Means are provided to restore, recharge and/or replace the capacities of the components that remove and accumulate radionuclides. Means are provided to package this waste in appropriately sealed packaging. Packaged waste is sent to the liquid and solid radioactive waste systems, based on their nature.

The HPS is operated and monitored in such a way that the radionuclide inventories in each of its components are maintained below design limits that ensure the radiological consequences of potential releases are bound by those of the maximum hypothetical accident (MHA).

Figure 5-5 Helium Purification System Modules

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}}a(4)

During shutdown and maintenance conditions, the HPS provides a processing path to remove impurities for the helium being routed to the HRS for storage. The HRS consists of storage vessels, piping, valves, and a compressor that temporarily stores helium and transfers it to and from the HTS, enabling depressurization and pressurization of the reactor during startup, shutdown, refueling, and other maintenance operations. This interface enables recovery and reuse of helium {{ }}a(4) when necessary.

The HPS maintains the purity and inventory of the primary helium coolant, protects plant components from contamination, and ensures radiological release control from the purification equipment. To support these functions, the HPS is provided with instrumentation and controls to monitor and maintain key operating parameters, including pressure and temperature indicators, to support normal operation. Pressure indicators provide indication at high-pressure helium storage, compressor inlet and discharge, and supports control to prevent depressurization of the reactor vessel and HPS boundary. Differential pressure instrumentation monitors filter loading to support timely replacement. Temperature controls maintain operating conditions in the preheater, cooler, low-temperature absorber, to ensure purification performance and protect equipment.

In addition to its purification function during normal plant operation, the HPS interfaces with the primary coolant cleanup system, a dedicated maintenance system that conditions the primary coolant. The primary coolant cleanup system is designed to purge the reactor following refueling and open maintenance. This system operates only during plant maintenance modes and is not used during normal operation. Helium used during purging is not recovered and is routed to the reactor Heating, Ventilation and Air Conditioning System (Chapter 9.1).

The sampling system for gaseous media monitors the quality of the circulating primary coolant to ensure that chemical and radiological impurity levels remain below the SARRDLs. Sampling points are located at key process locations in the HPS, where helium can be sampled into sealed containers for external laboratory analysis.

The HPS is designed in accordance with applicable industry codes and standards. Process piping within the HPS is designed to meet the requirements of ASME B31.3 (Reference 5-5) and pressure-retaining valves are designed in accordance with ASME B16.34 (Reference 5-8), valves, flanged, threaded, and welding end. HPS components that form part of the primary helium pressure boundary are designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 5 (Reference 5-4), consistent with the requirements applied to the reactor vessel and primary coolant pressure boundary.

5.4.5 System Evaluation

PDC-1

The HPS pressure vessels are designed, fabricated and tested in accordance with ASME BPVC III, Division 5 (Reference 5-4) requirements, ensuring adequate structural integrity and pressure-retaining capability for their intended service life. These design features demonstrate conformance with PDC-1.

PDC-2

HPS components located near safety related SSCs are either seismically mounted, physically separated, or protected by engineered barriers to prevent adverse seismic interactions and ensure safety related SSCs remain functional during and following seismic events. These provisions demonstrate conformance with PDC-2.

PDC-3

The HPS is constructed using non-combustible or fire-resistant materials, and its placement relative to potential fire or explosion sources is evaluated to minimize fire hazards and maintain system integrity under postulated fire conditions. Collectively, these measures satisfy the fire protection objectives of PDC-3.

PDC-4

The low-pressure design of the HPS (excluding supply tanks and piping to the pressure-reducing regulator) eliminates pipe-whip hazards, and any helium releases during system failures do not pose adverse interactions with nearby safety related systems. Accordingly, the HPS design meets the interaction and dynamic effects considerations of PDC-4.

PDC-10

The HPS is designed to remove radioactive impurities from the primary helium coolant, thereby supporting maintaining the primary coolant activity within SARRDLs during normal operation. Although the HPS is classified as NSR, its components are designed and fabricated in accordance with applicable codes and standards including the provisions of ASME BPVC III, Division 5 (Reference 5-4) thereby minimizing failure probability. In addition, monitoring equipment is integrated within the HPS to detect abnormal conditions and initiate appropriate actions, including isolation of affected HPS modules.

In the event of a postulated failure, radionuclides are retained within system components such as adsorption beds and filters, and would not be released as gaseous effluents, thereby maintaining worker and public exposures within SARRDLs. If helium were to leak through the HPS lines, the radionuclide content would remain well below the SARRDLs and would not exceed postulated events assessed in Chapter 13. These design features ensure that the HPS does not result in radioactive releases exceeding SARRDLs, consistent with the intent of PDC-10.

PDC-13

The HPS includes instrumentation to monitor key process variables, including pressure, temperature, and differential pressure to ensure operating conditions remain within limits. These signals provide operator indication and, where applicable, support automatic control and isolation actions to limit helium loss and prevent depressurization of the helium pressure boundary, thereby supporting safe operation. These are consistent with the instrumentation and control intent of PDC-13.

PDC-14

The HPS, as part of the primary helium pressure boundary, is designed, fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code, Section III (Reference 5-5), consistent with the requirements applied to the reactor vessel and primary coolant pressure boundary, to minimize the probability of leakage, failure, and gross rupture. This design approach supports pressure boundary integrity is consistent with the requirements of PDC-14.

PDC-15

HPS components that form part of the primary helium pressure boundary, including piping, valves, and storage components, are designed in accordance with applicable ASME standards, including ASME Boiler and Pressure Vessel Code Section III, Division 5 (Reference 5-4), with piping and valves designed in accordance with ASME B31.3 (Reference 5-5) and ASME B16.34 (Reference 5-8), as applicable. Design limits are established in excess of conditions anticipated during normal operation and postulated events to ensure allowable limits are not exceeded during operation, maintenance or testing. This design approach establishes appropriate margins and is consistent with the design limit requirements of PDC-15.

PDC-30

The HPS components part of the helium pressure boundary are manufactured and tested in accordance with applicable quality standards and are subject to pre-operation inspection and testing, as well as periodic inspection during scheduled outages, to support identification and evaluation of degradation or leakage and maintain pressure boundary integrity. These inspection and quality assurance provisions support continued pressure boundary integrity and are consistent with the intent of PDC-30.

PDC-32

The HPS design includes instrumentation, inspection access, sampling ports, and monitoring features that support inspection, monitoring, and functional testing of key areas needed to assess structural integrity and maintain leak-tight performance of the portions of the HPS part of the helium pressure boundary. These design features demonstrate conformance with PDC-32.

PDC-60

The HPS is designed to minimize tritium releases to the environment by maintaining pressure boundary integrity and providing isolation from the HTS. The HPS draws helium from and returns helium to the HTS through supply and return lines equipped with double isolation valves attached to the IHX vessel, enabling isolation of the system. The defined interface between the HPS and the primary coolant pressure boundary, together with system monitoring and maintenance, supports control of tritium-bearing gases and ensures that gaseous and liquid effluents remain within the regulatory limits of 10 CFR 20. These design features demonstrate conformance with PDC-60.

PDC-64

Radiation monitoring is provided in the HPS to support the identification and assessment of potential radioactive material releases that may occur as a result of system failures, including evaluation of tritium levels. These monitoring capabilities provide the means to detect and assess radiological conditions consistent with the objectives of PDC-64.

10 CFR 20.1406

The HPS contributes to maintaining low circulating activity in the primary coolant, within the SARRDLs, thus minimizing contaminants within the primary coolant. This further minimizes the potential leakage of contaminants into the secondary coolant or Citadel building air volumes. This design is consistent with the contamination minimization intent of 10 CFR 20.1406.

5.4.6 Testing and Inspections

The HPS, including the primary coolant cleanup system and sampling system ports for gaseous media, are designed to support testing, inspection and maintenance. The HPS is designed to support testing, inspection, and maintenance activities throughout commissioning and operation. Instrumentation penetrations and connection points are provided to allow attachment of test equipment as needed to verify system performance. Testing, qualification, and commissioning activities for HPS equipment, including components that form part of the reactor helium pressure boundary, will be conducted in accordance with applicable regulatory and code requirements, with detailed procedures to be defined in the Operating License Application. Pre- and in-service inspections, testing, and monitoring and alarm functions implemented in the UMSs (see Section 7.8), where practicable, are used to detect and identify the location of coolant leakage. The HPS design permits inspection, monitoring, and functional testing of important areas and features to assess structural and leak-tight integrity. The system and components incorporate the features necessary to implement surveillance and in-service inspection functions. Differential pressure sensors are provided across the high-temperature filter and the helium compressor to continuously monitor the condition of these components. Additional diagnostic monitoring techniques may be employed for detecting leaks, vibration, and other degraded conditions. Further details will be included in the Operating License Application.

Inspection, testing, and maintenance activities for the HPS will be planned and performed such that personnel radiation exposure, and any release of radioactivity remain within applicable requirements, including 10 CFR Part 20 (Reference 5-1), and below the SARRDLs. These activities will be performed in a manner that preserves the ability to safely shut down the reactor. Maintenance and testing activities will include controls intended to prevent loss of primary coolant inventory or uncontrolled release of primary coolant due to malfunctions or leaks. Operating limits, design parameters, and surveillance requirements applicable to the HPS, including those that may be proposed as technical specifications, will be provided in the Operating License Application.

The HPS also provides for the acquisition of grab samples for laboratory analysis to monitor primary coolant chemistry and radionuclide content and to confirm performance of the purification function.

Consistent with the white paper, “Proposed Contents of PSAR using NUREG-1537 Guidance for the Micro Modular Reactor” (Reference 5-6) and the associated NRC feedback (Reference 5-7), this document describes the system configuration, functional design features, interfaces, and monitoring and isolation capabilities of the HPS at high level. Impurity concentration limits, detailed testing and inspection procedures, surveillance frequencies, administrative controls, and the associated operating limits and design parameters that may be proposed as technical specifications for the PCCS, HPS, and Sampling System are deferred to the Operating License Application and associated technical specifications.

5.5 PRIMARY COOLANT MAKE-UP SYSTEM

5.5.1 Summary Description

The PCMS supplies fresh helium to the reactor unit both during initial plant startup and during normal operation to replace helium losses from leakage or maintenance activities. The system consists of high-pressure, high-purity, helium cylinders connected through pressure-regulating and control valves that regulate the supply pressure before delivery to the primary coolant contained within the VS. Flexible hoses and associated piping distribute the regulated helium to the reactor until the set system pressure is reached and maintained.

5.5.2 System Functions

5.5.2.1 Safety Functions

No safety functions are credited to the system.

5.5.2.2 Non-safety Functions

The primary non-safety function of the PCMS is to provide clean helium to the reactor unit during start-up of operations and during operations to offset normal losses of helium from the pressure boundary.

These non-safety functions are supported by the following responsibilities:

- Maintaining the primary coolant inventory by supplying helium to compensate for leakage or losses.
- Controlling helium make-up addition based on primary coolant pressure and inventory signals from reactor instrumentation.
- Providing and maintaining a controlled flow path between helium make-up cylinders and the primary system.
- Maintaining an on-site inventory of high-pressure helium, in cylinders, for primary coolant make-up.

5.5.2.3 System Classifications

The PCMS is classified as non-safety related.

5.5.3 Design Bases

Although the PCMS is classified as NSR, the system and its components have been designed to remain consistent, where applicable, with the following PDC, in addition to the general PDC discussed in Section 3.6.1.4:

- PDC-14, “Reactor helium pressure boundary”
- PDC-15, “Reactor helium pressure boundary design”
- PDC-30, “Quality of reactor helium pressure boundary integrity including reducing fluid ingress”
- PDC-31, “Fracture prevention of helium pressure boundary”
- PDC-32, “Inspection of reactor helium pressure boundary”

5.5.4 PCMS Design Description

The PCMS supplies fresh helium to the reactor during initial plant startup and normal operation to compensate for helium losses resulting from leakage or maintenance activities. The system consists of multiple high-pressure helium cylinders located within the Operations Building.

As shown in Figure 5-6, the PCMS includes {{ }}[a(4)] helium cylinder banks {{ }}[a(4)] connected to a common supply manifold. Each cylinder bank consists of several cylinders, providing the nominal helium inventory required by the primary coolant pressure boundary. These banks are used for initial system pressurization and bulk helium supply. In addition, a dedicated helium make-up cylinder, consisting of a single cylinder, is provided and is used to replenish the primary coolant pressure boundary, as required, during normal operation.

Figure 5-6 PCMS Schematic

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}}a(4)

Helium addition to the HTS is controlled {{ }}a(4) Supply pressure is reduced via pressure-regulating and control valves, down to the required operating pressure prior to delivery to the HTS. Regulated helium is distributed through flexible hoses and associated piping until the required HTS' fill is achieved. The filling process is {{ }}a(4) supervised, with automatic isolation provided to prevent backflow when cylinder pressure is below the HTS' helium pressure.

The Operations Building includes an exterior access door to allow safe replacement of depleted helium cylinders with new ones. PCMS piping and associated pressure-retaining components are designed in accordance with ASME B31.3 (Reference 5-5), with piping dimensions and wall thicknesses consistent with ASME B36.10M (Reference 5-9).

5.5.5 System Evaluation

PDC-1

The PCMS is designed, fabricated, installed, and tested in accordance with applicable codes and standards, including the provisions of ASME B31.3 (Reference 5-5) and ASME B36.10M (Reference 5-9) will be performed in a manner which complies with the University of Illinois Urbana-Champaign (U. of I.) Quality Assurance Program. These provisions are consistent with the requirements of PDC-1.

PDC-2

The PCMS is seismically qualified and arranged to prevent adverse interactions during a design basis earthquake through the use of supported and restrained helium supply components and defined interfaces with adjacent SSCs, thereby preserving coolant makeup capability. Accordingly, the PCMS design is consistent with the seismic interaction requirements of PDC-2.

PDC-3

The PCMS is arranged and installed such that fire and explosion hazards are minimized and adverse interactions with nearby safety related SSCs are prevented. Specific fire protection features and detailed fire hazards analysis will be provided in the Operating License Application. These design considerations satisfy the fire protection objectives of PDC-3.

PDC-4

The PCMS is arranged to limit the environmental and dynamic effects of postulated system failures on nearby safety related SSCs. Detailed evaluations of credible failure mechanisms will be addressed in the Operating License Application. This arrangement is consistent with the intent of PDC-4.

PDC-14

The components of the PCMS that interface with the reactor helium pressure boundary are designed to have a low probability of helium leakage through the application of ASME B31.3 (Reference 5-5) code requirements for materials, fabrication, and inspection, and by verifying pressure boundary integrity through standard pressure and leak testing and inspection during installation and maintenance. These design and verification measures support pressure boundary integrity consistent with the objectives of PDC-14.

PDC-15

The components of the PCMS that interface with the reactor helium pressure boundary, including piping, valves, and storage components, are designed in accordance with ASME B31.3 (Reference 5-5), with piping dimensions consistent with ASME B36.10M (Reference 5-9), and with pressure and temperature design limits established with margin above normal operating conditions and postulated events. This design approach establishes appropriate margins and is consistent with the design limit requirements of PDC-15.

PDC-30

The components of the PCMS that interface with the reactor helium pressure boundary are manufactured in accordance with applicable quality standards and code requirements, including ASME B31.3 (Reference 5-5) for pressure-retaining components. Pre-service inspections and testing, such as pressure and leak testing where applicable, are performed to verify pressure boundary integrity prior to operation. These provisions are consistent with the requirements of PDC-30.

PDC-31

The PCMS pressure boundary materials, wall thicknesses, and supports are selected and designed in accordance with ASME B31.3 (Reference 5-5), with wall thicknesses consistent with ASME B36.10M (Reference 5-9), to provide sufficient margin against stresses from operating loads, anticipated transients, thermal effects, fatigue, and corrosion, thereby maintaining pressure boundary integrity. This design provides appropriate structural margins and is consistent with the pressure boundary integrity objectives of PDC-31.

PDC-32

The components of the PCMS that interface with the reactor helium pressure boundary are designed to permit inspection and functional testing of pressure boundary components and interfaces that are important to maintain primary coolant boundary integrity. PCMS piping and components are designed in accordance with ASME B31.3 (Reference 5-5), which provides requirements for inspection and pressure testing of pressurized systems. These design features provide the means to inspect and verify pressure boundary integrity consistent with the objectives of PDC-32.

5.5.6 Testing and Inspections

Consistent with Section 5.4.6, the PCMS incorporates the necessary features to support testing, inspection, and in-service surveillance Those details will be included in the Operating License Application.

5.6 NITROGEN-16 CONTROL SYSTEM

The U. of I. research reactor does not use water as a coolant; therefore, Nitrogen-16 will not be produced in the core.

5.7 AUXILIARY SYSTEMS USING PRIMARY COOLANT

The U. of I. research reactor does not use the primary coolant to add heat or remove heat from auxiliary systems.

5.8 REFERENCES

5-1 NRC. NRC Regulations Title 10, Code of Federal Regulations, Part 20 – Standards for Protection Against Radiation.

5-2 American Society of Mechanical Engineers, ASME Boiler & Pressure Vessel Code, Section VIII, 2021.

5-3 American Society of Mechanical Engineers, ASME TES-1-2023, Safety Standard for Thermal Energy Storage Systems: Molten Salt.

5-4 ASME. Rules for Construction of Nuclear Facility Components – Division 5 – High Temperature Reactors, ASME BPVC.III.5 - BPVC Section III., Division V.

5-5 American Society of Mechanical Engineers, ASME B31.3 Process Piping Code, 2020.

5-6 U. of I. White Paper IMRDD-MMR-22-02, Release 01, “University of Illinois Research and Test Reactor: Proposed Contents of PSAR Using NUREG-1537 Guidance for the Micro Modular Reactor (MMR®),” dated August 25, 2022 (ADAMS Accession Number ML22220A252).

5-7 U.S. Nuclear Regulatory Commission, “Staff Observations Regarding University of Illinois at Urbana-Champaign White Paper – Proposed Contents of Preliminary Safety Analysis Report (PSAR) Using NUREG-1537 Guidance for the Micro Modular Reactor,” dated November 22, 2022 (ADAMS Accession Number ML22321A314).

5-8 American Society of Mechanical Engineers, ASME B16.34 Valves- Flanged, Threaded, and Welding End, 2004.

5-9 American Society of Mechanical Engineers, ASME B36.10M Welded and Seamless Wrought Steel Pipe, 2018.

USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 6 - ENGINEERED SAFETY FEATURES

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
°C	degrees Celsius
ESF	Engineered Safety Features
FQMTR	Fuel Qualification Methodology Report
kg/s	kilograms per second
m^3	cubic meters
MHA	Maximum Hypothetical Accident
NRC	Nuclear Regulatory Commission
ODSL	outer dense surface layer
PDC	Principal Design Criteria
RCCS	Reactor Cavity Cooling System
RV	Reactor Vessel
SARRDLs	specified acceptable radionuclide release design limits
SiC	Silicon carbide coating
SSCs	systems, structures and components
TRISO	Tri-structural Isotropic
UCO	uranium oxycarbide

CHAPTER 6 ENGINEERED SAFETY FEATURES

This chapter discusses and describes the Engineered Safety Features (ESF) for the University of Illinois (U. of I.) research reactor facility. The ESFs are active or passive features designed to mitigate the consequences of postulated events and to keep radiological exposures to the public, the facility staff, and the environment within acceptable limits. The concept for ESFs evolved from the defense-in-depth philosophy of multiple layers of design features to prevent or mitigate the release of radioactive materials to the environment during accident conditions. The need for ESFs is determined by analyzing postulated events and their consequences, even though the reactor facility design has made the incidence of those postulated events very unlikely.

6.1 SUMMARY DESCRIPTION

This section identifies and describes the ESF designed to mitigate the consequences of postulated events, ensuring that any potential dose consequences remain within acceptable values. The ESF safety functions are described and their role in the mitigation of the postulated events as evaluated in Chapter 13, including the Maximum Hypothetical Accident (MHA), are summarized.

The ESFs credited for mitigation of postulated events are functional containment and the Reactor Cavity Cooling System (RCCS).

Functional containment features are those that are credited for the retention of fission products at risk for release during postulated events. These features include the Tri-structural Isotropic (TRISO) fuel kernel, the multiple robust barriers surrounding this kernel being an integral part of the TRISO particles, and the ceramic-based pellet containing TRISO particles intended to retain activated products and fission products resulting from the reactor operation. The effective functional containment barriers include the fuel kernel, two pyrolytic carbon coatings separated by a silicon carbide (SiC) coating, and the fully dense outer SiC surface layer of the fuel pellet. To add to the robustness, the TRISO fuel and SiC pellet are qualified to withstand very high temperatures (> 1600°C), which far exceed temperatures expected to occur as a result of postulated events and the MHA, as described in Section 4.2. Functional containment, described in Section 6.2, is retention of activated products and fission products in all postulated events and the MHA described in Chapter 13. Additional proprietary functional features were added to the typical TRISO design to improve overall irradiation performance, as described in Section 2.2.1 of the "Fuel Qualification Methodology Topical Report" (FQMTR) (Reference 6-1).

The RCCS is the ESF that provides ultimate passive decay heat removal from the Reactor Vessel (RV) and reactor cavity during postulated events where the normal heat transfer pathway is not available. During normal operations, the RCCS acts as an active system to maintain the water contained in the RCCS at its nominal temperature. During postulated events where forced cooling of the reactor by helium is lost, the RCCS readily removes all of the decay heat through the active flow of water contained in the RCCS. In the very unlikely situation where active flow in the RCCS is no longer available, the RCCS automatically enters a passive state, where heat is removed through the heating and eventual boiling of water retained in the RCCS' cassette's large diameter standpipes, providing well over 72 hours of passive decay heat removal. In this scenario, although natural circulation would get established and increase the heat removal capacity, this is not credited. The RCCS, described in Section 6.3, is credited with the removal of decay heat from the reactor cavity in all postulated events and the MHA described in Chapter 13.

6.2 FUNCTIONAL CONTAINMENT

For the U. of I. research reactor, the ESF traditionally described in NUREG-1537 as "confinement and containment" is implemented as a reactor functional containment, consisting of multiple barriers surrounding the fuel that are designed to limit the physical transport (diffusion) and release of radionuclides. This represents a departure from the conventional pressure-retaining containment concept used for many light-water research reactors and is consistent with the functional containment framework described in RG 1.232 (Reference 6-2) and SECY-18-0096 (Reference 6-3) for advanced non-LWR designs.

The U.S. Nuclear Regulatory Commission (NRC) describes a methodology for functional containment in SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water-Reactors," which acknowledges that non-LWR technologies differ from LWRs in operating conditions, coolants, and fuel forms, which allows for a different approach to fulfill the safety function of limiting the physical transport of radioactive material to the environment. The NRC defines functional containment in SECY-18-0096 as "a barrier or set of barriers taken together, that effectively limits the physical transport of radioactive material to the environment." The Commission approved this SECY in SRM-SECY-18-0096.

Therefore, the majority of the radioactive material at risk for release in the reactor is held up by design of the fuel pellets themselves. As described in Section 4.2.1, the fuel design consists of TRISO-coated particles that are embedded in a ceramic matrix within an annular shell to form the fuel pellet.

The physical design features of the TRISO fuel, including the extremely high melting point of the materials used and the several successive layers contribute to the functional containment retention of fission products. The TRISO fuel's high acceptable operating temperature (> 1600°C) provides significant margin to failure in transient conditions. The fuel pellet design provides protection of the TRISO particles from mechanical damage. The design of the TRISO fuel particles and fuel pellet provides a set of physical barriers to retain the fission products. The first is the uranium oxycarbide (UCO) kernel itself (or the fuel kernel) because a large fraction of the radionuclides present in the fuel kernel are retained therein. The fuel kernel is further surrounded by three coatings which include a SiC layer sandwiched between dense inner and outer pyrolytic carbon layers (IPyC and OPyC, respectively). The SiC layer primarily helps to retain metallic fission products and acts as a pressure barrier between the fuel kernel and surroundings. The IPyC and OPyC layers provide effective barriers to mitigate fission gas and add mechanical support to the overall TRISO structure.

The last layer is the SiC outer dense surface layer (ODSL) of the fuel pellet, which adds an additional layer of containment. The ODSL retains fission products that are released from the TRISO particles and contains them inside the SiC matrix that encapsulates the coated kernels. This matrix is supported structurally by the ODSL.

This amounts to a total of five barriers that are each credited as an independent functional containment barrier. This functional separation means that the failure of the outermost barrier (the ODSL layer) does not affect the function of the TRISO particles barriers. The effectiveness of these barriers to fission product release has been demonstrated through extensive testing as described in Section 2.1 of the FQMTR (Reference 6-1).

The specified acceptable radionuclide release design limits (SARRDLs) must not be exceeded to ensure fuel failures do not cause excessive radionuclide releases that could result in unacceptable doses during normal operations or postulated events. The fuel performance evaluation under all those scenarios shows significant margin to failure of the functional containment barriers. A description of the design, evaluation, and testing of the fuel for its effectiveness at retaining radionuclides is provided in Section 4.2.

Fission products that are released from the functional containment of the fuel during normal operation will contribute to the circulating activity of the primary coolant, as discussed in Section 5.2.4.1. The circulating activity of the primary coolant is controlled through the removal of radioactive material by the Helium Purification System (Section 5.4). The MHA, described in Chapter 13, provides the dose consequences that bound all postulated events. The safety limits discussed in Chapter 14 will ensure that the SARRDLs are not exceeded, and potential dose consequences remain below dose targets (dose targets will be established based on the proposed Emergency Plan). The SARRDLs and technical specifications will be described in the Operating License Application.

The design bases, evaluation, qualification methodology, and irradiation and safety testing of the fuel's functional containment features discussed in this section are further described in Chapter 4. Chapter 13 evaluates the integrated functional containment approach by analyzing the external dose within the reactor site boundary associated with postulated events, including the MHA. Chapter 13 demonstrates that this functional containment approach is sufficient to maintain acceptable dose consequences to the research reactor staff and public.

6.3 REACTOR CAVITY COOLING SYSTEM

6.3.1 Summary Description

The primary function of the RCCS is to remove decay heat from the reactor cavity and transfer it to the outside atmosphere. The RCCS can operate as an active or passive system. Only the system's passive cooling mode function is credited in Chapter 13 for providing heat removal during postulated events. The passive cooling function removes heat from the RV and reactor cavity during postulated events through the heat up and boiling of the water contained in the RCCS standpipes. The resultant steam is vented to the atmosphere {{ }}a(4) The water volume contained in the RCCS can provide for well over 72 hours of decay heat removal capacity prior to needing to be refilled. The postulated events considered include those in which the RCCS's active cooling capability becomes unavailable, including the MHA. Components required to guarantee the RCCS's passive safety function are designated as safety related. All other components, or those only used for removal of heat through the RCCS's active cooling function, are designated as not safety related and perform no safety function.

None of the safety related portions of the RCCS require electrical power to perform their safety functions during postulated events. The RCCS also does not require any actuation to transition from active to passive cooling mode.

6.3.2 System Functions and Classification

6.3.2.1 Safety Functions

The RCCS performs the following safety functions:

- Passively remove decay heat from the reactor cavity during postulated events and the MHA described in Chapter 13, to maintain the reactor cavity, the RV, and the core (indirectly) temperatures within their respective design temperature limits during postulated events in which the active cooling mode becomes unavailable.

During and after postulated events, the active cooling mode of the RCCS is not credited for heat removal in Chapter 13. However, if an event occurs and the active cooling mode is still available, the RCCS will remain in this active mode and provide for infinite decay heat removal capacity.

6.3.2.2 Non-Safety Functions

The RCCS performs the following non-safety functions:

- Actively remove heat from the reactor cavity via the forced circulation of water in the RCCS to ensure that systems, structures and components (SSCs) remain within allowable design temperatures during normal operations and events in which active cooling mode remains available.
- Help maintain low RV temperatures during maintenance/planned outages by removing decay heat from the RV.
- Provide an alternative method for cooling the fuel and core (through indirect cooling) when the helium temperature is below the minimum temperature needed to transfer heat to the intermediate loop molten salt.

The direct cooling of the fuel through forced flow of the helium is the primary method for heat removal during normal operating conditions. These conditions include those in which the temperature of the helium in the Heat Transport System is higher than the cold leg temperature of the molten salt in the intermediate loop, as described in Chapter 5. Therefore, indirect heat removal provided by the RCCS becomes the alternative method in normal operating conditions and in conditions where the helium temperature is below the cold leg molten salt temperature.

The RCCS has the capability to provide the operator with the option to control the forced convection cooling parameters as necessary to adjust the heat transfer and maintain desired normal operating conditions.

6.3.2.3 System Classification

The portions of the RCCS that must function to passively remove decay heat during postulated events are classified as safety related and are credited in Chapter 13 safety analyses. All other components only used for removal of heat during normal operation through active cooling through the RCCS are designated as non-safety related and perform no safety function. Details for these components are provided in Section 6.3.4.

Safety related components are designed to Seismic Category I to significantly minimize their risk of failure. Non-safety related components which could fail as a result of a seismic event and impact the functionality of safety related SSCs are designed to Seismic Category I to prevent such an occurrence.

6.3.3 Design Basis

The Principal Design Criteria (PDC) discussed in Section 3.6.1.4 apply to the RCCS as well as the following ones:

- PDC-10, "Reactor design"
- PDC-34, "Passive residual heat removal"
- PDC-36, "Inspection of passive residual heat removal system"
- PDC-37, "Testing of passive residual heat removal system"
- PDC-44, "Structural and equipment cooling"
- PDC-45, "Inspection of structural and equipment cooling systems"
- PDC-46, "Testing of structural and equipment cooling systems"

6.3.4 Design Description

RCCS components are divided into two categories depending on their safety functions or non-safety functions. These categories are:

- **RCCS Passive Components:** These include all components required to remove heat from the reactor cavity including the RCCS cassettes, and outlet piping within the Citadel building as required to perform its safety function.
- **RCCS Active Components:** These include all other RCCS components and piping that perform non-safety functions and provide active cooling during normal operation.

6.3.4.1 RCCS Passive Components

The RCCS is comprised of {{ }}$^{a(4)}$ cassettes that span the height of the reactor cavity, except over the Cross Connection Vessel (Chapter 5) where a reduced-height cassette is installed. Each cassette is made of {{ }}$^{a(4)}$ standpipes connected with 180° return elbows. Each pair contains one downcomer and one riser standpipe. The standpipes are large diameter, thick-walled {{ }}$^{a(4)}$ pipes. Their diameter and wall thickness allow for rigidity, reliability and a volume of water sufficient to satisfy the passive cooling mode availability requirements.

Each cassette has an inlet and outlet manifold for the standpipe pairs to be connected in parallel. An expanded view of a cassette is shown in Figure 6-1.

Those inlet and outlet manifolds are then connected in parallel to the RCCS inlet and outlet ring headers positioned above, as shown in Figure 6-2. The RCCS cassettes are installed next to each other along the inner wall of the reactor cavity to form a concentric ring around the RV. The inner radius of the cassettes allows for sufficient space between the RV and RCCS cassettes to accommodate inspection equipment.

The outlet common ring header directs flow into redundant outlet pipes, which function as vent paths when in the passive cooling mode. The outflow path out of the Citadel building is protected from external hazards, ensuring the safety-related function of the RCCS.

With this configuration, only the water volume contained in the cassettes is credited as available during postulated events. The total loss of water from a single standpipe pair results in a loss of {{ }}$^{a(4)}$ of the total water volume credited for passive cooling.

Additional redundancy is provided {{ }}$^{a(4)}$

Lastly, each cassette includes a {{ }}$^{a(4)}$ on the inner radius of the cassettes that serves as an impingement shield, or a barrier for water transference from the RCCS to the RV. The impingement shield prevents spray that may occur as a result of standpipe leaks from reaching the RV. This shield extends the full height of the reactor cavity and assists in directing any potential water spray to the sump described in Chapter 3.

Figure 6-1 RCCS cassette exploded view

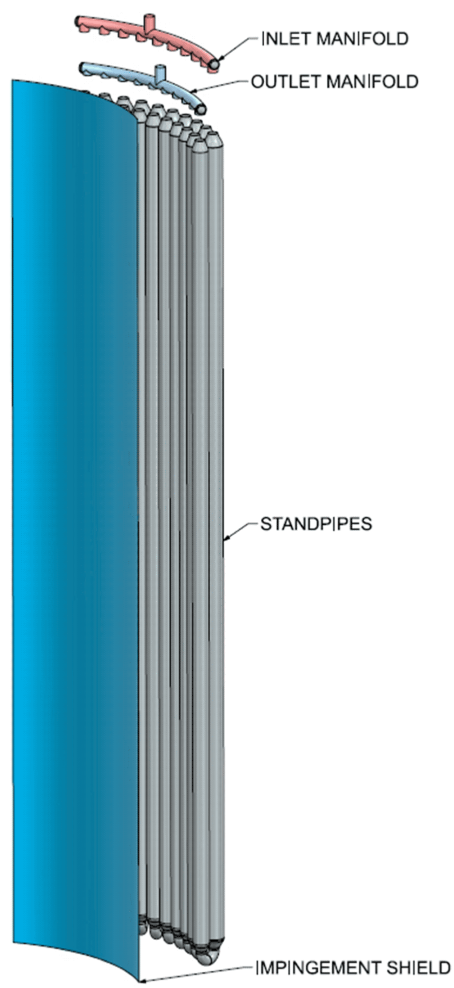

Figure 6-2 Reactor Cavity Cooling System cassettes assembly

{{

}}a(4)

6.3.4.2 RCCS Active Components

Active components are located outside of the Citadel building and connect the RCCS passive components to the rest of the system. These include components that are not required for passive heat removal, including the following:

- Carbon steel inflow pipe
- Redundant carbon steel outflow pipes outside of the Citadel building
- Heat exchanger
- Atmospheric surge tank
- Redundant pumps
- Pump bypass
- Motorized isolation valves
- Check valves

The RCCS flow diagram is presented in Figure 6-3 and shows the relationship between interfacing systems, the RCCS, and the vent line to atmosphere. A redundant means to refill the RCCS through the demineralized water distribution system is provided as part of the defense-in-depth approach. If the demineralized water tank is not sufficient or unavailable, a means to supply water from a reserve supply through the inlet ring header is also available. The RCCS has sufficient water to perform its safety function well beyond a 72-hour period and can be refilled to extend this duration, as necessary. In addition, {{

}}a(4) These design considerations and redundancies allow for the continuation of the RCCS passive heat removal function beyond the initial capacity of the RCCS. The outlet piping {{

}}a(4) are not required for the RCCS to perform its safety function. The loss of this piping does not adversely impact the safety function of any other systems as all safety related systems are located within the Citadel building.

The RCCS cassettes and piping inside the reactor cavity are secured to the concrete structure of the Citadel building through seismic restraints. In addition, {{

}}a(4) protect the inflow and outflow pipes connecting the active components from external hazards. The RCCS atmospheric surge tank is located outside the Citadel building above the elevation of the RCCS cassettes. The Citadel building and reactor cavity are described in Chapter 3.

Figure 6-3 RCCS Flow Diagram

{{

}}a(4)

Table 6-1 provides the basic parameters of the RCCS.

Table 6-1 Summary of RCCS Cassettes

Parameter	Nominal Value
Total Water Volume (m^3)	{{ }}a(4)
Total Standpipe Pairs	{{ }}a(4)
Active Cooling Maximum Flow Rate (kg/s)	{{ }}a(4)
Active Cooling Mode Minimum Temperature (°C)	40
Active Cooling Mode Maximum Temperature (°C)	44

The RCCS will be designed to the following standards as applicable:

- Passive piping components: ASME BPVC III Div 5 Class B (Reference 6-4)
- Active piping components: ASME B31.3 M (Reference 6-5)

The RCCS piping and cassettes will be constructed from {{ }}a(4) as applicable to the design standard.

6.3.5 Basic Operational Overview

The RCCS uses demineralized water as the working fluid for system function in both active and passive cooling modes. In both cooling modes, the heat is predominantly transferred from the RV to the RCCS via convection and radiation. The active cooling mode provides a means for managing heat removal from the reactor cavity during normal operations. During postulated events in which the active cooling mode is no longer available, the passive cooling mode removes heat from the reactor cavity, RV, and the core (due to indirect cooling), through the heat capacity and latent heat of vaporization of water. The RCCS will ensure that the temperature of these structures and components is maintained within their respective design limits as described in Chapters 3 and 4. Therefore, the passive cooling mode is credited for all safety functions. In addition, the passive cooling mode does not require any instrumentation or electrical power to perform the RCCS safety functions during postulated events. If the active cooling mode, active components, and/or instrumentation are lost, none of the RCCS safety functions are adversely impacted. Active functions from the RCCS components may be lost in case of loss of electrical power to the pumps motors, or a failure of active components – none of which are safety-related.

The operational modes of the RCCS can be summarized as follows:

Active Cooling Mode: During normal operations, the primary coolant removes the majority of the heat from the core; however, some heat is transferred through the RV to the RCCS cassettes (via convection and radiation through the reactor cavity) and the demineralized water contained inside. The water is continuously circulated through the RCCS system, and heat is removed from the water via a heat exchanger to the Chilled Water System. Through controlling the flow rate of the water, heat can be removed from the reactor cavity with a measure of control and monitoring to sustain normal reactor operations and to ensure entry conditions required for adequate passive cooling capabilities are maintained during normal operations. Associated technical specifications will be provided in the Operating License Application.

Passive Cooling Mode: The passive cooling mode requires only the RCCS {{

}}a(4) In postulated events where the active components are assumed to be unavailable, resulting in a loss of forced flow, the RCCS retains the water present within the cassettes at the onset of the event and the RCCS defaults to the passive cooling mode. {{

}}a(4) The initial water inventory in the cassettes is sufficient to allow the RCCS to continue rejecting heat via heat-up and steaming to atmosphere for more than 72 hours after active cooling is lost while preventing SSCs from exceeding their design limits from being reached. {{

}}a(4) The passive heat removal pathway is described in Section 4.6.

The demineralized water used as coolant for the RCCS is chemically treated by the Chemical Dosing System described in Section 9.7 to mitigate corrosion on system materials and piping.

Instrumentation is included in the RCCS to monitor and control water conditions such as temperature, pressure and flow rate. Loss of instrumentation will not impact safety as passive heat rejection via heating and boiling of the water inventory retained in the standpipes will still be achieved.

6.3.6 System Evaluation

Selected portions of the RCCS, as described in Section 6.3.1, are responsible for the passive heat removal safety function for postulated events and the MHA, as described in Chapter 13. These components are designed to applicable codes and standards for each component and the Quality Assurance Program requirements described in Section 12.9. These features demonstrate conformance with the requirements of PDC-1.

The safety related RCCS components involved in the passive heat removal safety function are primarily located in the safety related portion of the Citadel building. Section 3.4 discusses design features to address the effects of postulated seismic events on safety related SSCs. These RCCS components are designed with the considerations described in Chapter 3 for seismic and other hazards such that the RCCS can perform its passive heat removal safety function during and following a postulated event. The RCCS passive heat removal design requirements for seismic and other hazards demonstrate conformance with the requirements of PDC-2.

The safety related components of the RCCS are designed and located to minimize the probability and effect of fires and explosions by using low combustible materials and physical separation. These design features, in conjunction with the fire protection plan described in Section 9.3, provide assurance that the RCCS passive heat removal safety function demonstrates conformance with the requirements of PDC-3.

The RCCS is made of materials that will withstand the radiation environment of the reactor cavity and environmental temperatures to ensure the RCCS can perform its safety function associated with normal operation, maintenance, testing, and postulated events. The water for the system will be chemically controlled to reduce chemical attack on system materials and system piping. This piping is designed with a sufficient safety factor to readily accommodate expected lifetime corrosion. Prior to restarting the reactor following a postulated event, the water inventory of the RCCS will be restored to normal using chemically treated water to prevent increased corrosion rate. Any inspection requirements of the RCCS after postulated events will be included in the Operating License Application. The RCCS is designed against equipment failures that could result from water leakage or helium leakage during postulated loss of coolant inventory events. Pipe whip and other similar dynamic failures are avoided by the low-pressure design of the RCCS and the use of restraints. Each component of the RCCS is designed such that failure of one component does not cascade and cause failures of nearby systems, including other RCCS components. These design considerations demonstrate conformance with the requirements in PDC-4.

The RCCS passive cooling mode removes heat from the reactor cavity, the RV, and associated support structures and therefore, indirectly, the fuel. The RCCS ultimately rejects that heat to atmosphere through steam during postulated events. This relies on basic natural phenomena driven by heat radiation, convection, thermal capacity and latent energy of vaporization. The RCCS is capable of passively removing decay heat from the reactor cavity, RV and structural supports without reliance on electric power for more than 72 hours to mitigate postulated events. This ensures that the reactor cavity wall and RV temperatures remain below their respective design limit. The RCCS is designed with sufficient reliability, redundancy and leak detection capability to ensure the passive safety function can be performed in the event of a single failure. Independent standpipe pairs ensure sufficient coolant is retained in the event of multiple standpipes failure. The passive cooling mode and any related components, including the cassettes and the outlet piping that serve as the heat transfer pathway out of the Citadel building, ensure performance of the safety function during postulated events where active cooling mode is no longer available. {{

}}[a(4)] they are not required for the RCCS to perform its safety function. RCCS passive cooling is not required during normal operations but always remains available. RCCS active cooling during normal operations ensures that the conditions exist for adequate heat removal during postulated events that include a loss of the active mode of the RCCS. These design provisions satisfy PDC-10, PDC-34 and PDC-44.

The RCCS design includes the capability for online monitoring of leaks to monitor for system integrity. Additionally, the RCCS design provides sufficient accessibility to perform inspections for system integrity in compliance with appropriate codes and standards. The RCCS is designed such that there is enough space between the RV and the RCCS impingement shield to allow for sufficient accessibility. In addition, the design permits appropriate periodic functional testing using manual actuation of the active components and monitoring of system conditions. This functional testing can be performed to demonstrate there are no leaks or blockages in the system as well as indicating the passive heat transfer of the system is performing as intended. Periodic testing of transitioning between the active and passive modes is not required, as this occurs naturally at the onset of postulated events when active cooling is lost. These features demonstrate conformance with the requirements in PDC-36, PDC-37, PDC-45, and PDC-46.

6.3.7 Testing and Inspection

The details of the inspection and testing programs, if required, will be described in the Operating License Application.

Appropriate technical specifications will be included in the Operating License Application. This will be summarized in Chapter 14.

6.4 REFERENCES

6-1 U. of I. Topical Report IMRDD-MMR-24-01-NP-A, Release 02, "University of Illinois Urbana-Champaign High-Temperature Gas-cooled Research Reactor: Fuel Qualification Methodology – Topical Report," dated April 21, 2025 (ADAMS Accession Number ML25101A107).

6-2 Regulatory Guide 1.232, Revision 0, "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors," April 2018.

6-3 SECY-18-0996, "Functional Containment Performance Criteria for Non-Light-Water-Reactors," Enclosures 1 and 2, September 2018.

6-4 American Society of Mechanical Engineers, ASME Boiler & Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors," 2023.

6-5 American Society of Mechanical Engineers, ASME B31.3 Process Piping Code, 2020.

USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 7 - INSTRUMENTATION AND CONTROL SYSTEMS

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
{{ }}[a(4)]	{{ }}[a(4)]
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CBMS	Citadel Building Monitoring System
CRDUs	Control Rod Drive Unit
DCS	distributed control system
ESF	engineered safety features
HFE	human factors engineering
HSI	human-system interface
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IHX	Intermediate Heat Exchanger
IICSs	Information Instrumentation and Control Systems
ISA	International Society of Automation
MCR	Main Control Room
NRC	Nuclear Regulatory Commission
NSR	non safety-related
OSHA	Occupational Safety and Health Administration
PAM	post-accident monitoring
PDC	principal design criteria
QAP	Quality Assurance Program
RCS	Reactor Control System
RCSS	Reactivity Control and Shutdown System
RMS	Radioactivity Monitoring System
RPS	Reactor Protection System
RTDs	Resistance Temperature Detectors
SARRDLs	specified acceptable radionuclide release design limits
SCA	Secondary Control Area
SR	safety-related
SSCs	structures, systems and components
TRs	Topical Reports
UMSs	Unit Monitoring Systems
V&V	verification and validation

CHAPTER 7 INSTRUMENTATION AND CONTROL SYSTEMS

This chapter describes and discusses the operating characteristics of the research reactor facility Instrumentation and Control (I&C) Systems. Included in this chapter are the design criteria and bases, and the functional and safety analyses of the I&C subsystems. These systems, together with their design criteria and bases, assure that the research reactor facility may be safely operated, monitored, and shut down as warranted. These systems are sufficiently sensitive to ensure control and shutdown of the reactor.

7.1 INSTRUMENTATION AND CONTROL SYSTEMS OVERVIEW

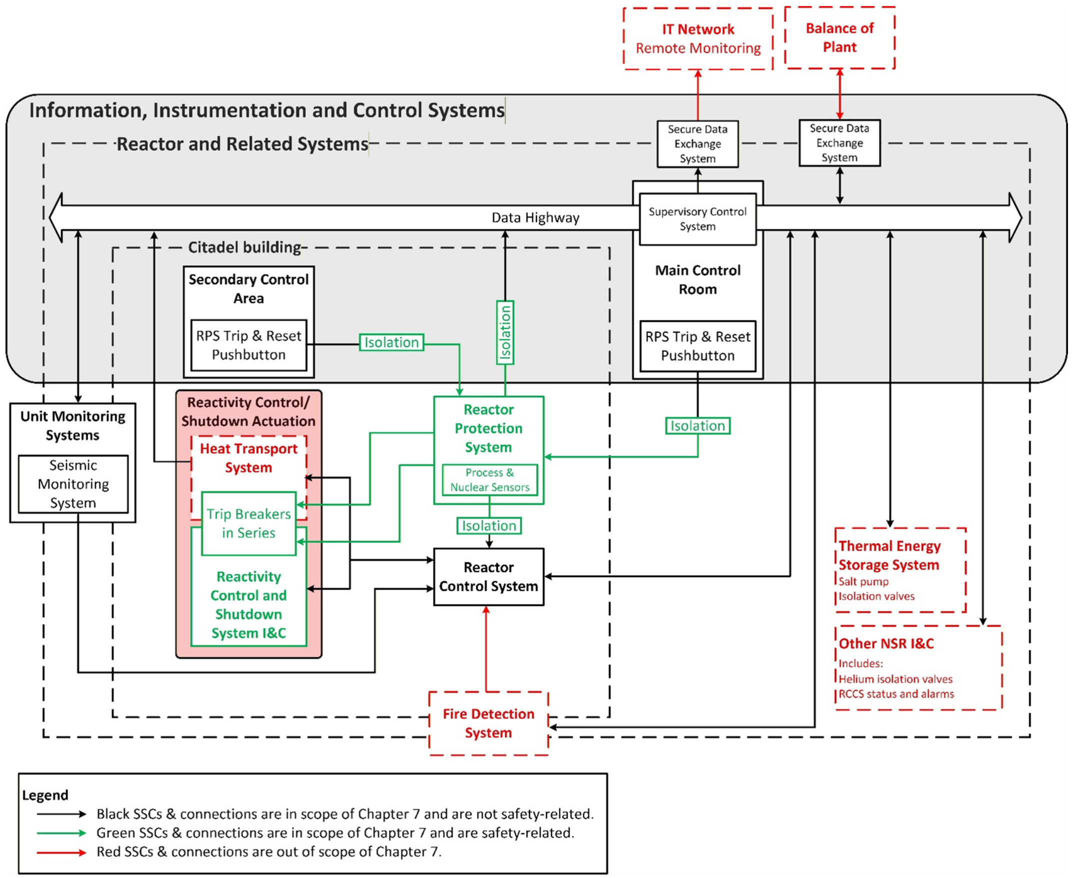
7.1.1 Summary Description

The reactor I&C systems provide the means to monitor, control, and protect the reactor and related systems over all normal operating modes and postulated events, consistent with the non-power research reactor licensing framework under 10 CFR Part 50 (Reference 7-1). The overall I&C architecture supports the fundamental safety functions of controlling reactivity, removing heat from the core, and confining radioactive material by combining highly reliable passive features with a limited set of active safety systems and predominantly non-safety related (NSR) control and monitoring systems.

At the highest level, the facility I&C (shown in Figure 7-1) is structured as:

- The Information, Instrumentation, and Control Systems (IICSs) are implemented primarily on a distributed control system (DCS) platform. The IICSs support functions of the reactor and related systems such as maintaining normal operation, providing information to operators, and supporting post-accident monitoring (PAM).
- The Reactor Control System (RCS) performs automatic and manual control of reactor power and process variables within predefined operating envelopes and reduces the probability of circumstances requiring actuations of the safety-related (SR) systems.
- The Reactor Protection System (RPS) provides automatic reactor trip capability such that, upon conditions that challenge safety limits or protective setpoints derived from considering the safety analysis results in view of the applicable principal design criteria (PDC), the trip actuation path is de-energized and the reactor is placed in a safe shutdown state.
- The Reactivity Control and Shutdown System (RCSS) includes the control rod drives and actuation hardware that enables control rod motion and shut down actuation in response to RCS control commands and RPS trip actuation. The RCSS design details are described in Section 4.2.3; the present chapter addresses the RCSS interfaces to the RCS and RPS, including the hierarchical sharing of Control Rod Drive Unit (CRDU) command and feedback signals.
- The Unit Monitoring Systems (UMSs) include process, area, and radiation monitoring channels that provide continuous indication and alarms in the control facilities and support reactor-inherent (non-I&C) engineered safety features (ESFs) and PAM functions.

Figure 7-1 High-Level Representative Schematic Diagram of I&C Configuration



The I&C architecture implements redundancy and defense-in-depth through multiple channels, functional independence, and physical separation of components. Normal operation is managed by automated control loops and supervisory control functions implemented within the IICS subsystems, with operator manual intervention provided through the Main Control Room (MCR) operating and supervisory workstations. Independent protection logic, embedded in the RPS, monitors selected nuclear and process parameters using dedicated SR sensors and initiates a reactor trip and associated actions. During normal operation, the NSR RCS uses process instrumentation to support control of reactor power and key processes, while the NSR UMSs provide independent indications and alarms of key reactor and related system parameters to support operator response and post-event assessment. UMS monitoring information is sent to the higher-level IICSs for display and recording, and evaluation of the information is performed in the IICSs rather than the UMSs.

As described in Section 7.6, field sensors are predominantly analog, with signal digitization and processing performed in system controls circuits; operator displays and control logic are implemented on digital platforms. Fire protection (including fire detection and alarm) is addressed in Section 9.3 and is only discussed in Chapter 7 where interfaces provide input or status indications to the I&C systems discussed herein. Human-system interface (HSI) features and the human factors engineering (HFE) program basis are summarized in Section 7.1.5, with detailed HFE implementation addressed in the Operating License Application.

7.1.2 I&C System Classification and Safety Functions

The I&C design and its safety classification approach reflect the project licensing strategy, which aligns with the non-power reactor regulatory framework and incorporates risk-informed evaluations of the applicability of advanced reactor and light-water-reactor-oriented regulations, as appropriate. The role of I&C systems in meeting PDC, and in supporting exemption and non-applicability determinations where appropriate, is described in detail in the NRC-approved Topical Reports (TRs), "Applicability of Nuclear Regulatory Commission Regulations" (Reference 7-2) and "Principal Design Criteria" (Reference 7-3). As discussed in the NRC-approved TR, "Events Sequence Identification and SSC Classification Methodology" (Reference 7-4), the I&C structures, systems, and components (SSCs) discussed in Chapter 7 are classified based on their roles in performing the safety functions described in Section 7.1.1. SR I&C functions are ensured by the RPS and associated actuation paths, while NSR I&C functions are ensured by the IICS, RCS, and UMSs.

- The IICSs are classified as NSR because they are not credited with performing any safety functions in response to postulated events considered in the safety analysis. The IICSs provide facility-level control, monitoring, data management, alarm, and communication functions implemented on a distributed control system platform for normal and off-normal operations, but loss or malfunction of these systems does not prevent any SR systems from performing their required safety functions or meeting applicable PDC.

- The RCS is classified as NSR since it is not credited with performing any safety functions in response to postulated events. It performs automatic and manual control of reactor power in normal operation and postulated events, maintaining process variables within specified limits and thereby reducing demands on the RPS and minimizing unnecessary challenges to SR systems. In specific operating modes, the RCS also communicates with the supervisory control system, which adjusts the flows of molten salt coming in to and out of the molten salt tanks. The RCS provides defense-in-depth by initiating reactor reverses when warranted, effectively limiting the frequency and severity of reactor trips that could otherwise wear SR systems.
- The RPS is classified as SR because it is credited with initiating protective reactor trips to support the reactor protection and control of reactivity safety functions. The RPS identifies deviations from acceptable operating limits and initiates reactor trips to perform this safety function during postulated events (e.g., spurious control rod withdrawal). The RPS includes SR sensing, logic, and actuation functions that detect neutron flux and key process parameters and initiate a reactor trip by opening the reactor trip breakers. In turn, this stops forced-flow cooling and allows the control rods to freely drop into the core under gravity, effectively shutting down the reactor. The associated shutdown actuation equipment is described in Section 4.2.3.
- The UMSs and associated radiation and activity monitoring systems detect elevated radionuclide levels, support operator actions in accordance with procedures, and support the effectiveness of radionuclide retention features and radiation protection provisions. These systems are classified as NSR, as they do not perform or support any of the safety functions and are not required as part of the ESFs (which are all passive).

Consistent with this classification and functional analysis, the architecture described in Section 7.1.3 maintains independence of the RPS-credited protection logic and actuation paths from NSR control and information systems, while supporting facility control, monitoring, and operator interface functions.

7.1.3 I&C System Architecture

The I&C systems are implemented as an integrated and functionally hierarchical information, instrumentation, and control architecture that supports automatic protection, control, and monitoring for the facility while maintaining independence of SR systems. The I&C systems include the IICS data networks and interfaces used for communication among the I&C systems and for the transfer of monitoring and control information.

The facility-level IICS architecture is based on a DCS that includes a supervisory control system, local controllers and remote terminal units, a data highway, secure data exchange subsystems, and operating and supervisory workstations. The IICSs provide supervisory control, data acquisition, and operator interfaces for the reactor and related systems, interfacing with I&C systems which independently control the balance-of-plant and other non-reactor-related systems. This interface is achieved through segregated data transfer systems and secure gateways to preserve electrical and functional independence between nuclear and non-nuclear portions of the facility while supporting coordinated load and energy management.

The RCS and UMSs interface with the higher-level IICS layer to support normal control and monitoring, display, recording, and operator response. The RPS protection logic and actuation paths are independent from the facility IICSs and other NSR systems. {{

}}[a(4)]

The RPS does not allow any control or communication feedback from the RCS.

Field instrumentation for the facility I&C systems consists primarily of conventional analog process and nuclear sensors and devices, with signal conditioning and digitization and control/logic execution performed in distributed digital controllers.

Testing, calibration and surveillance concepts applicable to I&C systems are summarized in Section 7.2 and in the system sections, with implementation details to be provided in the Operating License Application.

7.1.4 I&C Architecture Attributes

The I&C architecture is structured to preserve the independence, and ensure the reliability, of credited safety functions while enabling control and monitoring functions needed for normal operation and post-event information.

7.1.4.1 Redundancy, Reliability and Independence

Consistent with PDC-24, protection and control functions are separated such that failure, or removal from service, of any single RCS component or channel cannot prevent the RPS from performing its protective function. The RPS is designed with redundancy and independence such that no single RPS component or channel failure, or removal from service of a single RPS channel, results in loss of protective function. {{

}}[a(4)] As described in the architecture section, RPS protective functions are not inhibited by RCS actions.

7.1.4.2 Fail-safe Behavior

Consistent with PDC-23, the RPS is designed with a de-energize to trip philosophy, such that a loss of electrical power, physical break in the trip command circuit, or system fault will cause the system to fail into a safe state. A loss of electrical power or break in the circuit causes two de-energize-open trip breakers to remove power from the CRDUs and the helium circulator, resulting in gravity insertion of the control rods and circulator shutdown. The self-diagnostics capabilities of the digital RPS platform are programmed to de-energize a channel upon sensing a fault.

7.1.4.3 Diversity

Consistent with PDC-22, technological and functional diversity are incorporated in system design and principles of operation to prevent loss of the protection functions. Diverse signal processing technology will be used within the digital RPS architecture. The RPS measures and trips the reactor from diverse parameters of neutron flux and core temperature. The RPS is an active protection system, diverse from the reactor-inherent passive safety features.

7.1.4.4 Support Monitoring Systems

Monitoring and indication functions that support operations and PAM are implemented in systems classified as NSR, including the RCS and UMSs. These systems provide monitoring and status information to support situational awareness and response and event logging and are not credited with performing safety functions.

7.1.5 Human-System Interface Overview

The I&C design incorporates HSI features to support control and monitoring from the MCR across startup, normal, off-normal, refueling, and shutdown states, and to permit shutdown of major systems and components for equipment protection and personnel safety. The MCR HSI provides the capability to monitor and control all equipment whose failure could have an immediate impact on facility output and will have access to all process information systems.

The overall HSI concept is developed under the facility's HFE program, which defines criteria for workplace and workstation design, HSI design, and environmental conditions to support safe, comfortable, and effective human performance. Detailed HFE implementation is subject to detailed design and will be addressed in the HFE program and Operating License Application.

Personnel workstations in the MCR provide unified operator interfaces for monitoring and controlling the reactor across normal and off-normal operating states. A physically and electrically separated Secondary Control Area (SCA) provides an alternative means to initiate manual reactor trip and perform selected monitoring if the MCR is unavailable or uninhabitable. Detailed HSI content and layout for the MCR and SCA will be detailed in the Operating License Application.

The primary operator interface to the I&C systems is provided in {{ }}[a(4)] The MCR includes operating stations, supervisory workstations, and safety parameter displays that present hierarchically organized information on facility, system, and component status, including safety parameters and alarms. The operating and supervisory workstations provide the primary means to monitor the facility conditions and initiate control actions during normal operations. The operating and supervisory workstations support concurrent facility monitoring and control by providing full control capability at the operating workstation and monitoring/overview capability at the supervisory workstation. During abnormal conditions, the workstation layout supports continued monitoring of overall facility status while control actions are directed from the operating workstation to the affected portion of the facility. Alarm presentation employs HFE principles of prioritization and operating-state-dependent processing of alarms to avoid information overload and to focus operator attention on conditions requiring action. The HSI design separates controls and displays used for normal operation from those relied upon during postulated events (example: the manual trip actuation circuit is implemented on dedicated hardwired inputs to the RPS, and post-event parameters are presented on dedicated displays).

The SCA is situated within the Citadel building. As described in Section 7.7.1, it includes a display capability for monitoring the status of selected SR and NSR systems and a hard-wired manual reactor trip pushbutton. Detailed habitability, accessibility, and event-specific availability evaluations will be provided with the Operating License Application.

7.2 DESIGN OF INSTRUMENTATION AND CONTROL SYSTEMS

7.2.1 Design Criteria

This section summarizes the design criteria used to develop the I&C systems, consistent with the requirements of 10 CFR 50.34(a)(3)(i) (Reference 7-1). I&C SSCs are designed, constructed, and tested to quality standards commensurate with the safety importance of the functions performed. Where generally recognized codes and standards are used, they are identified and evaluated for applicability and adequacy and are supplemented or modified as needed consistent with the safety importance of the function. I&C SSCs are designed consistent with applicable local building and siting codes and to withstand the effects of natural phenomena. System-specific criteria and code/standard evaluations are addressed in Section 7.2 through Section 7.8.

7.2.2 Design Bases and Principal Design Criteria Mapping

This section provides traceability between the design bases for the I&C systems described in Chapter 7 and the applicable PDC identified for the facility. The applicable PDC for the facility are provided in the NRC-approved Topical Report, "Principal Design Criteria" (Reference 7-3).

Table 7-1 identifies the PDC addressed for each I&C system within Chapter 7. As discussed in Section 3.6.1.4, PDC-1 through PDC-5 were identified as applicable to all SR SSCs, including a subset of NSR SSCs considered important to safety, and evaluation of the general design requirements in PDC-1 through PDC-5 is provided in this section for all I&C systems. Evaluation of the remaining PDC is provided in the system-specific sections identified in Table 7-1.

Table 7-1 Mapping of PDC to Systems in Chapter 7

Chapter 7 System	PDC Addressed in System Design Bases
Global I&C Design Criteria (all systems)	PDC-1, PDC-2, PDC-3, PDC-4, PDC-5
Reactor Control System (RCS) (Section 7.3)	PDC-10, PDC-12, PDC-13, PDC-26, PDC-28
Reactor Protection System (RPS) (Section 7.4)	PDC-10, PDC-12, PDC-13, PDC-20, PDC-21, PDC-22, PDC-23, PDC-24, PDC-25, PDC-26, PDC-28, PDC-29
Sensors and Instrumentation (Section 7.6)	PDC-13, PDC-14, PDC-15, PDC-20, PDC-21, PDC-22, PDC-23, PDC-24, PDC-29, PDC-30, PDC-31, PDC-32, PDC-64, PDC-72

Table 7-1 Mapping of PDC to Systems in Chapter 7

Chapter 7 System	PDC Addressed in System Design Bases
Main Control Room / Secondary Control Area (MCR/ SCA) (Section 7.7)	PDC-13, PDC-19
Unit and Radioactivity Monitoring Systems (UMSs/ RMS) (Section 7.8)	PDC-13, PDC-19, PDC-64, PDC-72

7.2.3 Design Description

Chapter 7 describes the facility I&C systems that support monitoring, control, and protection functions, including the RPS, RCS, Sensors and Instrumentation, MCR and SCA interfaces, and the UMSs and Radioactivity Monitoring System (RMS), with detailed system descriptions provided in Section 7.3 through Section 7.8.

These I&C system descriptions provide the associated principal safety considerations by identifying the credited safety-related protection functions, the non-safety monitoring and control functions that support facility operation, and the interfaces and separation between SR and NSR systems.

7.2.3.1 Instrumentation and Control System Functions

Chapter 7 describes I&C systems that collectively implement safety-related protection functions as summarized below. The RPS achieves the primary protection function of shutting down the reactor by disconnecting electrical power (tripping breakers) from fail-safe devices:

- CRDUs de-energized: reactivity control rods will insert under gravity, leading to insertion of maximum negative reactivity.
- The helium circulator de-energized: primary coolant flow stops, reactor temperature increases, and reactor negative temperature coefficients quickly reduce the reactivity.

7.2.4 I&C System Evaluation

The I&C systems described in Chapter 7 collectively implement the design criteria summarized in Section 7.2.2 and are evaluated against the corresponding PDC. System-specific PDC are evaluated in the corresponding subsystem sections. The general PDC (PDC-1 through PDC-5) applicable at the chapter level are addressed below.

Consistent with PDC-1, the I&C systems safety-related or important to safety described in Chapter 7 are developed and controlled under an established Quality Assurance Program (QAP) that complies with the U. of I. NRC-approved QAP (Reference 7-5), with quality standards and records commensurate with the functions performed.

Consistent with PDC-2, PDC-3 and PDC-4, the design and installation of I&C subsystems credited to safety functions (and the supporting structures and environments in which they are installed) provide appropriate protection against anticipated natural phenomena, fire/explosion hazards, and expected environmental and dynamic effects for the conditions under which the subsystems are required to operate.

SR I&C equipment is located inside the Citadel building and is therefore protected against anticipated natural phenomena, fires/explosions, and environmental and dynamic effects in compliance with the relevant PDC.

Consistent with PDC-2, I&C systems are designed in accordance with the requirements of the applicable seismic code (Table 3.6-1), with sufficient margin to withstand anticipated seismic events. Consistent with PDC-4, the I&C systems are also designed using relevant industry codes and standards to accommodate and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated events. The I&C architecture is designed such that failures or malfunctions of NSR I&C do not prevent SR systems from performing their required safety functions.

Consistent with PDC-3, I&C systems are designed and located to minimize the probability and propagation of fires/explosions and use non-combustible and fire-resistant materials when practical. Additional requirements may be identified after performance of a fire hazard analysis for the Operating License Application (see Section 9.3).

Consistent with PDC-5, the facility licensing basis is a single reactor, and the I&C subsystems described in Chapter 7 support that single-reactor facility configuration.

7.2.5 I&C Design and Development, Quality Assurance, and Cybersecurity

The I&C systems described in this chapter are developed using a lifecycle approach that supports design implementation, integration, and verification prior to operation, and supports maintenance and controlled modification during operation. The I&C design includes features intended to support testing, inspection, and maintenance during and after commissioning.

Configuration management, including verification and validation (V&V) of I&C modifications, is addressed in the facility QAP and supporting software lifecycle documentation, with implementation details provided in the Operating License Application, as applicable.

Cybersecurity considerations are incorporated into the I&C architecture, including use of secured data exchange functions (e.g., firewalls and related hardware/software) to limit communication access between networks and/or systems in accordance with the Cyber Security Plan. The Cyber Security Plan and associated access control and cybersecurity documentation will be provided in the Operating License Application. The I&C cybersecurity design is informed by recognized standards, including IEC 62645 (Reference 7-6), IEC 62859 (Reference 7-7), and ISA-TR84.00.09 (Reference 7-8), as applicable. Access to IICS terminals used to perform system software configuration changes is controlled through defense-in-depth measures, including physical access controls (e.g., keylocks) and secure authentication methods.

The I&C architecture incorporates provisions for testing, calibration, surveillance, and maintenance using a graded calibration and surveillance philosophy for protective trips, interlocks, and annunciators. SR RPS channels are designed to be periodically tested in conditions as close as practical to normal operating conditions, with provisions for independent channel testing and test equipment connection points, and provisions to permit periodic testing and calibration of protection channels and associated trip logic during operation. NSR control and monitoring systems, including the RCS and UMSs, are provided with test attachments, internal diagnostics, and support for in-situ calibration and routine surveillance of field instruments and alarm functions. Detailed test methods, frequencies, and allowable tolerances will be defined in the facility procedures, technical specifications, and the Operating License Application.

Detailed QAP implementation, software lifecycle documentation (including configuration management and V&V), cybersecurity and surveillance/maintenance program details will be provided in the Operating License Application, as applicable.

7.3 REACTOR CONTROL SYSTEM

7.3.1 Summary Description

The RCS automatically and manually controls the reactor to maintain reactor power and key process variables within the predefined normal operating envelope. The RCS receives reactor power demand setpoints from the higher-level supervisory control and from operator inputs and maintains reactor power at the requested level by coordinated adjustment of reactivity and primary coolant flow. The RCS also supports automated reactor startup sequences and programmed power changes so that the reactor can be brought from shutdown to normal power and returned to lower power conditions in a controlled manner.

To perform these functions, the RCS programmable logic controllers and associated input/output modules interface with instrumentation loops and generate control signals to (i) the RCSS CRDUs to position the control rods and (ii) the helium circulator speed to adjust primary coolant flow. The RCS supports automatic operation in multiple modes, including a 'normal mode' {{ }}[a(4)] and an 'alternate mode' {{ }}[a(4)] thereby maintaining its outlet temperature at the nominal operating setpoint.

RCS control logic is implemented on programmable digital devices which interface with the supervisory control system via the data highway and with operators via the MCR workstations HSI.

The RCS is not credited with performing any safety functions in response to postulated events. Its primary role is to operate and control the reactor and associated process variables within predefined boundaries during normal operation and postulated events, thereby reducing unnecessary challenges to the RPS and ESFs.

7.3.1.1 Safety Functions

The RCS does not perform any safety functions.

7.3.1.2 Non-safety Functions

The RCS performs the following non-safety functions:

- Maintain safe operation by:
  - Maintaining energy production
  - Maintaining shutdown
  - Maintaining refueling
  - Maintaining startup/shutdown transition
- Maintain facility protection by:
  - Protecting capability to maintain energy production
  - Protecting capability to maintain shutdown
  - Protecting capability to maintain reactor refueling
  - Protecting capability to maintain startup / shutdown transition

These non-safety functions ensure operational stability, minimizing challenges to the RPS, and reducing the frequency of reactor trips.

7.3.1.3 Classification

As described in Section 7.1.2, the RCS is classified as NSR. Following the methodology outlined in Section 3.6.2, compliance with the applicable local building code seismic design requirements defines the seismic design basis for NSR SSCs.

7.3.2 Design Bases and Criteria

In addition to the global PDC listed in Table 7-1, the following PDC from the NRC-approved Topical Report (Reference 7-3) are used in the development of the RCS design bases:

- PDC-10: "Reactor design"
- PDC-12: "Suppression of reactor power oscillations"
- PDC-13: "Instrumentation and control"
- PDC-26: "Reactivity control systems"
- PDC-28: "Reactivity limits"

Pressure-boundary design considerations for sensors and associated connections supporting RCS functions are addressed in Section 7.6.

The RCS design identifies and documents applicable codes, standards, and regulations as part of the design effort, including applicable state and local requirements. The RCS is designed in conformance with Occupational Safety and Health Administration requirements in 29 CFR 1910 (Reference 7-9). RCS cybersecurity requirements are informed by NRC guidance (e.g., RG 5.71 (Reference 7-10) and RG 1.152 (Reference 7-11)) and industry standards for programmable digital devices (e.g., IEEE 7-4.3.2 (Reference 7-12)).

The RCS hardware and software are developed, verified, configured, and maintained under the I&C design and development, QA, and cybersecurity processes described in Section 7.2.5, consistent with its non-safety classification and interfaces with safety-related systems.

7.3.3 Design Description

The RCS is a dedicated digital control system that adjusts reactor power and primary coolant system conditions by coordinated control of reactivity and primary coolant flow. It supports controlled changes in reactor power between defined operating modes and states by adjusting the position of the seven control rods inside the reactor core and modulating the helium circulator speed. The RCS cabinet includes programmable digital devices and associated input/output modules that receive signals from nuclear, temperature, pressure, and flow instrumentation and generate control signals to the RCSS CRDUs and helium circulator control panels. {{

}}a(4) The RCS also interfaces with other NSR systems, including the molten salt and helium isolation valves, via the IICS data highway as necessary to ensure the proper operation of the facility. Details of these interfaces will be provided as part of the Operating License Application.

Figure 7-2 illustrates the schematic layout of the RCS.

Figure 7-2 Representative Schematic Diagram of RCS Configuration

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}}a(4)

Consistent with PDC-26, the RCS provides individual control signals to each of the CRDUs and uses position feedback {{ }}a(4) to position rods in accordance with control demands. Consistent with PDC-28, the maximum rate of control rod withdrawal is limited to {{ }}a(4) by the RCS. Rod-drive condition is monitored using torque and temperature signals from the CRDUs, with abnormal conditions triggering alarm signals to the RCS which are transmitted onwards to the IICSs.

The RCS interfaces with the helium circulator control panels for control and monitoring functions. It provides a speed-setpoint signal via an analog output, start/stop commands, run permissives, quick-stop and reset commands, and receives circulator status, current speed, overspeed, and condition indications. Consistent with PDC-10, by increasing the circulator speed, the RCS increases helium primary coolant mass flow and heat removal from the core; the reactor's negative temperature coefficient causes reactivity and power to increase to a new equilibrium consistent with the increased heat removal, while reduced circulator speed similarly results in a corresponding reduction in power.

Consistent with PDC-13, Instrument signal inputs to the RCS include:

- Reactor inlet/outlet temperature signals {{ }}a(4) from the isolated RPS temperature detection instrumentation,
- Helium primary coolant pressure signals {{ }}a(4) from the isolated RPS pressure detection instrumentation,
- Reactor power and period signals {{ }}a(4) from the isolated RPS nuclear (neutron flux) instrumentation,
- RPS trip status signals, received via an isolated unidirectional interface,
- Salt temperatures, pressure and flow measurements {{ }}a(4)
- Helium primary coolant mass flow measurements {{ }}a(4)
- CRDU parameters,
- Pressure boundary temperature measurements {{ }}a(4) and
- Selected facility monitoring inputs (including fire detection, seismic alarms, and Reactor Cavity Cooling System (RCCS) status/alarm indications).

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}}a(4)

The RCS cabinet interfaces electrically and logically with the shared RPS instrumentation, CRDU signal-conditioning cabinets, the helium circulator control cabinet, the MCR control console, the common data highway, and selected monitoring systems. The RCS also has a dedicated hard-wired connection to the Fire Detection System (Section 9.3) and the seismic monitoring system (Section 7.8). The data highway connection allows the supervisory control system and operating workstations in the MCR to send power-demand setpoints and receive RCS status, alarms, and historical data, while the operating workstation provides the interactive operator interface for adjusting setpoints, selecting control modes, and monitoring RCS-controlled parameters.

The RCS is designed to accommodate the reactor operating states and modes defined in Chapter 14. It supports all operational modes and includes automatic and manual control modes and 'normal' and 'alternate' operating regimes. {{

}}a(4) In manual modes, control rod and circulator setpoints can be directly adjusted using the operating workstations, subject to the interlock and limit protections implemented in the control logic.

Consistent with PDC-26, the RCS also incorporates logic functions that initiate predefined control responses when abnormal conditions are detected. One such response is a reactor reverse action, which results in insertion of the control rods and reduction or termination of the forced cooling. This results in a reduction of the reactor power, and in the possible isolation of the primary coolant loop from the Helium Purification System. The RCS will initiate these control responses, for instance, upon detection of primary system depressurization, loss-of-cooling events, or seismic or fire alarm signals. These control responses act to stabilize plant conditions and limit the progression of the initiating disturbance, thereby reducing the likelihood of reaching the RPS trip setpoints. These control responses are coordinated with, but not credited in place of, the safety-related reactor trip initiated by RPS when protective setpoints are reached.

The RCS interfaces with the CRDUs at the CRDU cabinets by providing command signals to the stepper motor driver and associated signal-conditioning equipment located in the cabinets. CRDU position and equipment condition indications are returned to the RCS through the same cabinet interfaces for control and indication functions. Electrical power for the CRDU stepper motor drives is supplied via an interruptible feed controlled by the RPS trip breakers, so that a reactor trip removes CRDU power to permit gravity insertion.

There are no system-unique deviations from the generic I&C design and development and cybersecurity approach described in Section 7.2.5.

Overall, the RCS architecture and interfaces provide a dedicated, NSR means for precise control of reactor power and primary system conditions during normal operation and postulated events, while preserving the independence of SR protection and shutdown functions credited to the RPS and the RCSS.

7.3.4 System Evaluation

General design criteria (PDC-1 through PDC-5) applicable to Chapter 7 systems are addressed in Section 7.2.4 and are not repeated here. Additionally, the RCS satisfies the applicable PDC identified in Section 7.3.2 as follows.

The RCS maintains reactor power and temperature within prescribed operating ranges by coordinated control rod motion and helium circulator speed control. It is designed such that operation within control limits prevents exceeding specified acceptable radionuclide release design limits (SARRDLs) during normal operation and expected operational transients. These design considerations demonstrate conformance with PDC-10. In conformance with PDC-12, the RCS also includes reactor control strategies intended to avoid and suppress reactor power oscillations. It is designed such that coordinated control rod motion and helium circulator speed control provide stable control response over expected operating conditions. Transient scenarios and results that demonstrate acceptable performance for reactivity insertion and loss-of-forced-cooling event classes are provided in Chapter 13.

The RCS provides the I&C functions needed to monitor and adjust variables affecting the fission process and reactor helium pressure boundary during normal conditions as well as during postulated events. Protective action is provided by the RPS, which is completely independent and circumvents the RCS output when protective setpoints are reached. This design consideration demonstrates conformance with PDC-13. Pressure-boundary design, inspection, and testing considerations for sensing-related penetrations and associated connections are addressed in Section 7.6.

The RCS provides the control logic and commands for positioning the control rods and adjusting helium primary coolant flow to control reactor power during normal operation and the postulated events. Protection against reactivity insertion accidents is provided by the RPS and RCSS, which maintain priority over all RCS actions. These design considerations demonstrate conformance with PDC-26.

The RCS control logic is designed with constraints on control rod withdrawal sequencing and the rate of withdrawal. This limits positive reactivity insertion during normal operation. The RPS provides independent protective action to terminate conditions approaching established design limits. Bounding reactivity-insertion analyses are presented in Chapter 13. These design considerations demonstrate conformance with PDC-28.

7.3.5 Testing and Inspection

The RCS includes test equipment attachments on instrumentation penetrations and connection points to support testing, inspection, and maintenance during and after commissioning. The RCS is designed to allow comprehensive testing in all modes and states for which it is designed to operate. It incorporates internal diagnostic monitoring to detect malfunctions and support condition-based maintenance and troubleshooting.

Inspection and testing of sensing-related pressure-boundary penetrations and associated connections are addressed in Section 7.6.

Continuous online self-testing of hardware components will be performed via human-system interfaces, with the objective of monitoring system availability and functionality and, where practical, reducing reliance on separate manual periodic functional tests.

A detailed RCS surveillance, inspection, and in-service test program, including specific test procedures, configurations, and frequencies, will be established and documented in the Operating License Application.

7.4 REACTOR PROTECTION SYSTEM

7.4.1 Summary Description

The RPS is an independent, safety-related, control system that initiates protective actions when trip conditions are reached or when internal faults occur. The RPS is operable during all plant modes and continuously monitors plant status, with automatic self-diagnostics and abnormal condition indication.

The RPS monitors neutron flux and a defined set of key primary system process variables, and compares the measured values against protective setpoints informed by the safety analyses and overall reactor safety considerations. Consistent with PDC-21, for each credited trip parameter, the RPS provides {{ }}[a(4)] independent channels and applies {{ }}[a(4)] coincidence logic to command initiation of protective action. Each RPS channel includes its own sensor or sensor set, signal conditioning, and a trip comparator that generates a channel trip signal when that parameter reaches its protective setpoint. The {{ }}[a(4)] coincidence logic initiates a protective action when any {{ }}[a(4)] channels for that parameter are in the tripped state.

Consistent with PDC-20, when the RPS determines that a trip condition exists, it generates a reactor trip signal that opens {{ }}[a(4)] circuit breakers installed in series, thereby removing electrical power from the CRDUs and the helium circulator motor so that the control rods insert into the core under gravity and forced flow is stopped. Both those means of inserting strong negative reactivity are individually capable of quickly bringing the reactor to a safe shutdown state with adequate safety limit margin.

The RPS trip logic and signal processing are implemented on a digital platform. The associated input parameter sensors are primarily analog field devices (e.g., pressure, temperature, level, flow, and nuclear instruments) that provide analog output signals. The types of sensors used are described in Section 7.6.

Consistent with PDC-13, key parameters monitored are:

- Neutron flux,
- Reactivity rate-of-change / period (calculated from neutron flux, consistent with PDC-28),
- Helium primary coolant pressure (redundant transmitters {{ }}[a(4)] and
- Reactor inlet/outlet temperature (redundant thermocouples {{ }}[a(4)]

Consistent with PDC-23, the RPS is designed to fail to a safe, predictable state (fail-safe) where loss of electrical power, loss of signal continuity, or internal faults cause the trip output signals to assume their de-energized (trip) state. Redundant trip signals actuate redundant trip relays (de-energize to open) arranged in series along the CRDU motor circuit; opening either of the redundant trip breakers causes all control rods to insert into the core under gravity and the helium circulator to stop, shutting down the reactor. The RPS design does not rely on any operator action to perform its safety functions.

The RPS is operable in all reactor modes, performs continuous self-monitoring, and supports bypass of individual channels for testing and maintenance while retaining the required protective function by virtue of redundancy and automatic adjustment of the coincidence logic when a channel is out of service. Automatic reactor trip can be initiated when any of the trip-initiating conditions are met, and equivalent protective actions can also be initiated manually from dedicated trip pushbuttons located in the MCR and SCA, but no operator action is required to fulfill the safety function of the RPS. After a reactor trip, the reactor will remain shutdown until all trip conditions have been cleared and applicable procedures, such as investigation and remediation of the cause, have been completed.

Interfaces displaying RPS data to the operator are provided in both the MCR and the SCA and consist of displays and controls that support monitoring, manual trip initiation, and reset functions. {{

}}[a(4)] No commands or data requests from non-safety systems are permitted into the RPS, preserving functional independence of the safety functions.

7.4.1.1 Safety Functions

The RPS is credited with performing the fundamental safety functions of protection of the reactor and control of reactivity. The RPS therefore has the following safety functions:

- Detecting the approach to unsafe conditions via parameter limit comparison,
- Tripping breakers to permit gravity-insertion of negative reactivity, and
- Tripping breakers to cease circulation of primary coolant.

These functions are credited in the accident analyses (Chapter 13) to prevent fuel and pressure-boundary limits from being exceeded during the postulated event sequences.

7.4.1.2 Non-safety Functions

The RPS does not perform any non-safety functions.

7.4.1.3 Classification

Based on the above and the SSC safety classification methodology (Reference 7-4), the RPS is classified as SR. Applying the seismic classification methodology of ASCE/SEI 43-19 (Reference 7-13), as described in Section 3.6.2, the RPS is Seismic Design Category (SDC-)3.

7.4.2 Design Bases and Criteria

In addition to the global PDC listed in Table 7-1, the following PDC from the NRC-approved Topical Report (Reference 7-3) were used in the development of the RPS design bases:

- PDC-10: “Reactor design”
- PDC-12: “Suppression of reactor power oscillations”
- PDC-13: “Instrumentation and control”
- PDC-20: “Protection system functions”
- PDC-21: “Protection system reliability and testability”
- PDC-22: “Protection system independence”
- PDC-23: “Protection system failure modes”
- PDC-24: “Separation of protection and control systems”
- PDC-25: “Protection system requirements for reactivity control malfunctions”
- PDC-26: “Reactivity control systems”
- PDC-28: “Reactivity limits”
- PDC-29: “Protection against anticipated operational occurrences”

Pressure-boundary design considerations for sensors and associated connections supporting RPS functions are addressed in Section 7.6.

Consistent with PDC-29, the RPS is designed to meet the applicable requirements of IEEE 603-2018 (Reference 7-15) and IEEE Std 7-4.3.2-2016 (Reference 7-16) for nuclear safety-related systems and SR programmable digital devices. RPS equipment is environmentally and seismically qualified in accordance with IEEE Std 323-2016 (Reference 7-17) and IEEE Std 344-2020 (Reference 7-18) and implements the single-failure, independence, and accident-monitoring provisions of IEEE Stds. 379-2014 (Reference 7-19), 384-2018 (Reference 7-20), and 497-2016 (Reference 7-21), as guided by NRC Regulatory Guides, RG 1.105 (Reference 7-22), RG 1.152 (Reference 7-11), and RG 1.153 (Reference 7-23).

7.4.3 Design Description

The RPS is a SR digital protection system that uses SR instrumentation to monitor neutron flux, reactor period (derived from neutron flux), primary coolant inlet/outlet temperatures, and primary coolant pressure, generating protective reactor trip signals upon detection of pre-defined trip conditions. When the {{ }}a(4) logic for any credited trip parameter determines that a trip condition exists, or when internal faults or loss of electrical power place the system in its fail-safe state, the RPS generates a reactor trip signal that opens the reactor trip breakers and de-energizes the connected systems. Consistent with PDC-26, this action removes electrical power from all CRDUs, allowing control rods to insert into the core by gravity, and de-energizes the helium circulator. Forced-flow stoppage leads to a temperature increase and associated negative reactivity feedback. Both those means of inserting negative reactivity are individually capable of quickly bringing the reactor to a safe shutdown state, with adequate provisions for faults and uncertainties. The RPS logic is represented schematically in Figure 7-3.

Consistent with PDC-21, the {{ }}a(4) architecture design provides the capability to permit periodic testing during reactor operation, including independent testing of individual channels to detect failures and losses of redundancy. Each reactor trip breaker receives an independent trip signal from the RPS trip logic through separate actuation paths, preserving channel independence and ensuring that a single failure in the trip actuation circuitry cannot prevent the required reactor trip function. The trip actuation pathway, including trip breakers, is designed with sufficient redundancy to support the removal from service or bypass of individual components for maintenance, surveillance or testing activities while maintaining the required protection system redundancy. These design features provide functional reliability and in-service testability commensurate with the safety functions of the RPS and ensure that no single failure of any component or channel results in loss of the protection function.

Table 7-2 summarizes the RPS trip functions/setpoints and associated variables. The presented nominal setpoints are used for analysis, with consideration for calibration errors, instrument accuracy, transient overshoot, instrument and setpoint drift, etc., in accordance with IEEE Standard 603 (Reference 7-15) and ISA Standard S67.04 (Reference 7-24). Actual setpoints will be bound by the presented values and finalized in the Operating License Application.

Table 7-2 RPS Trip Functions and Nominal Analysis Setpoints*

Measured Variable	**Protective Trip Function**	**Nominal Trip Setpoint**	**Time Delay**	**RPS Measurement Channels**
Neutron flux	High	{{ }}a(4)	{{ }}a(4)	{{ }}a(4)
Reactor period	Low	{{ }}a(4)	{{ }}a(4)	{{ }}a(4)
Primary coolant pressure	Low	{{ }}a(4)	{{ }}a(4)	{{ }}a(4)
Reactor inlet temperature	High	{{ }}a(4)	{{ }}a(4)	{{ }}a(4)
Reactor outlet temperature	High	{{ }}a(4)	{{ }}a(4)	{{ }}a(4)
Neutron flux in shutdown mode	High	{{ }}a(4)	{{ }}a(4)	{{ }}a(4)
Operator manual actuation	Manual trip	N/A	N/A	2 pushbuttons

* Nominal trip setpoints and time delays are preliminary and subject to refinement as the design is finalized. Final values, associated uncertainty allowances, and the applicable setpoint methodology will be provided in the Operating License Application.

Figure 7-3 Representative Schematic Diagram of RPS Configuration

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}}a(4)

Operator interfaces, both direct and indirect, for the RPS are provided in the MCR and in the SCA. These include displays, alarms, and dedicated manual trip pushbuttons, all of which are NSR, that allow the operator to initiate a reactor trip on demand and to monitor RPS status and trip indications. The operator interfaces are unable to affect the performance of the underlying RPS logic and actuation functions in performing the credited safety functions. Information exchange with the RCS and facility monitoring systems is limited to unidirectional, electrically isolated outputs that provide status, trip indication, and selected measured variables such as neutron flux, reactor period, reactor inlet/outlet temperatures, and helium primary coolant pressure to the RCS.

The detailed list of RPS interlocks and inhibits will be provided in the Operating License Application and associated operating procedures.

RPS logic and signal-processing equipment are located in seismically qualified cabinets in the Citadel building and are powered by an NSR power supply (see Chapter 8).

Consistent with PDC-10, setpoints for SR instrumentation, including those that initiate RPS trips and interlocks, are established in accordance with ANSI/ISA 67.04.01-2018 (Reference 7-24) and NRC RG 1.105 (Reference 7-22), with explicit consideration of instrument and process uncertainties. The RPS applies this methodology to ensure that the analytical limits credited in the safety analyses are protected with appropriate margins, with detailed setpoint calculations and surveillance requirements provided in the Operating License Application and technical specifications.

Access control, cybersecurity, and software design and development for the RPS follow the facility-wide I&C processes described in Section 7.2.5. Additionally, RPS hardware and software are developed, verified, validated, configured, and maintained under the facility QAP as discussed in Section 7.2.5. The RPS equipment and configuration interfaces are located within physically protected secure development areas, with access limited to authorized personnel. NSR systems, including the RCS and facility supervisory networks, cannot issue commands or configuration changes to the RPS. Communication interfaces are implemented in accordance with the project I&C isolation and cybersecurity requirements, with detailed implementation and programmatic controls to be described in the Operating License Application and associated access control and cybersecurity documentation.

7.4.4 System Evaluation

The following discussion summarizes how the RPS satisfies the applicable PDC identified in Section 7.4.2. The RPS follows the I&C design and development, QA, and cybersecurity processes described in Section 7.2.5, with implementation controls commensurate with its SR classification. Detailed performance analyses, setpoints, response times, and surveillance requirements will be provided in the Operating License Application. General design criteria (PDC-1 through PDC-5) applicable to Chapter 7 systems are addressed in Section 7.2.4 and are not repeated here.

For the RPS, design features that support PDC-2, PDC-3 and PDC-4 include physical separation of redundant channels to enhance the probability that safety-related functions are performed in all postulated scenarios. This includes the location of RPS control logic in separate cabinets.

The RPS monitors neutron flux, reactor period, primary coolant pressure and reactor inlet/outlet temperatures, using independent and redundant SR sensors, and initiates a reactor trip when protective setpoints are reached. A reactor trip is implemented by opening the reactor trip breakers, thereby de-energizing the CRDUs so the control rods insert into the reactor core under gravity and de-energizing the helium circulator. These protective functions maintain margin to the acceptance criteria and limiting conditions assumed in the Chapter 13 accident analyses for normal operation and postulated events, consistent with the intent of PDC-10.

The RPS recognizes incipient power oscillations or unstable power conditions by monitoring neutron flux, reactor period, and key thermal-hydraulic variables and trips the reactor before conditions can challenge SARRDLs. This capability supplements the inherent negative reactivity feedback of the research reactor core to prevent damaging power oscillations, in compliance with PDC-12.

The RPS provides independent and redundant sensors and digital logic that continuously monitors reactor power, reactor inlet/outlet temperatures, and helium pressure over the ranges applicable to normal operation and postulated events. Monitoring of these parameters supports protection of the fission process, reactor core integrity, helium pressure boundary, and functional containment. Pressure-boundary design, inspection, and testing considerations for sensing-related penetrations and associated connections are addressed in Section 7.6. In compliance with PDC-13, the RPS automatically initiates control rod insertion and helium circulator stoppage to bring the reactor to a safe shutdown state when the associated sensors and logic identify a departure from acceptable limits or suffer an internal fault.

The RPS is designed to automatically initiate operation of systems to ensure that SARRDLs are not exceeded as a result of postulated events. The RPS detects abnormal reactor conditions using qualified, redundant sensors and {{ }}a(4) voting protection channels and generates redundant reactor trip and associated protective action signals to maintain nuclear and process parameters within the design limits. Automatic RPS actuation commands insertion of the control rods via the RCSS and simultaneously de-energize the helium circulator, providing a rapid reduction in reactor power during RPS trips. By providing these automatic trips and protective actions, in coordination with reactor systems and the RCSS, the RPS supports compliance with PDC-20 for facility protection functions.

The RPS is designed with high functional reliability and in-service testability. To satisfy PDC-21, the RPS is designed with equipment redundancy and independence, as well as {{ }}a(4) voting logic, such that no single failure or removal from service of any component or channel results in loss of the protection function. The RPS is designed to permit for periodic testing during reactor operation. The RPS reliability design includes physically separate sensors feeding redundant channel logic and comparator functions, providing fault tolerance and allowing individual channels to be tested while the protection function remains available.

The RPS is designed with independence and diversity in component design and principles of operation, to the extent practical, to prevent loss of the protection function consistent with PDC-22. As described in Section 7.2.4, the RPS is designed and installed such that anticipated natural phenomena (including seismic events) do not prevent the system from performing its required protective functions, including through location within the Citadel building and design to relevant codes and standards. Protection channels use independent, isolated sensing, logic, and actuation paths, so that failures in one channel do not propagate to the others and do not prevent the RPS from performing required trips. Interfacing control systems are required to communicate with the RPS in an isolated, one-way manner that prevents NSR systems from sending inputs into the RPS, which supports functional and cyber independence.

The RPS is designed with a de-energize-to-trip philosophy so that loss of energy or adverse environmental conditions cause the system to fail into a safe or demonstrably acceptable state. The RPS's digital self-testing is also designed to remove a faulted channel from service, resulting in a trip if the minimal number of channels for operation is not met. Removal of trip actuation power de-energizes the shutdown actuation path, resulting in gravity insertion of the control rods and trip of the helium circulator. Accordingly, loss of RPS power or internal RPS faults results in the same shutdown actuation outcome as a demanded trip, consistent with PDC-23.

The RPS is separated from the RCS employing electrical and communication isolation protocols so that failure or removal from service of any single component or channel common to both does not compromise the reliability, redundancy, or independence of the protection functions. {{

}}[a(4)] The facility I&C architecture further enforces separation by requiring that communications from the RPS to NSR systems be one-way only. In this way, NSR signals are prevented from adversely influencing RPS channels.

Reactivity control malfunctions are addressed by the facility reactivity control and shutdown design features credited in the accident analyses (Chapter 13). When a trip condition is met, the RPS de-energizes the trip actuation path, resulting in shutdown actuation through insertion of all seven control rods and shutdown of the helium circulator. The reactivity worth of six control rods, in case of one failing to insert, is amply sufficient to reach a safe shutdown state. Similarly, the circulator shutdown will result in inherent reactivity feedback that will readily bring the reactor into a safe shutdown state. Therefore, the RPS supports compliance with PDC-25. Chapter 13 analyses demonstrate that the credited shutdown response, considering applicable single-failure assumptions and response time, is sufficient to ensure SARRDLs and reactor helium pressure boundary design limits are not exceeded for the postulated reactivity events.

The RPS does not provide reactivity control directly but supports compliance with PDC-26 by providing SR trip signals that remove power from the CRDUs and initiate gravity-driven insertion of the control rods when protective actions are required. Thus, the RPS supports the overall reactivity control system design by ensuring that control rod insertion and shutdown functions are actuated promptly and reliably when reactivity excursions or other events warrant protective action, consistent with the intent of PDC-26.

In accordance with PDC-28, the RPS responds to postulated events, including those initiated by reactivity events, by tripping the reactor well before safety limits are reached to prevent exceeding SARRDLs and reactor helium pressure boundary design limits. Detailed reactivity limits, associated RPS setpoints, and reactor behavior during postulated events are discussed in Chapter 13 and will be further documented in the Operating License Application and technical specifications.

The RPS is designed with the requisite architectural reliability features, fail-safe characteristics, and environmental protections to ensure a high probability of initiating its protective actions under postulated event conditions consistent with the intent of PDC-29.

7.4.5 Testing and Inspection

The RPS will be tested and inspected under a quality-assurance framework that requires the system to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of its safety functions, and to comply with an established QAP. The RPS is designed in accordance with IEEE 603 (Reference 7-15), which provides criteria for SR system surveillance, testing, calibration, and inspection.

Individual RPS channels can be bypassed or taken out of service for surveillance, testing, or maintenance while maintaining the required protective function. The coincidence logic is configured to preserve protective capability with one channel available, by operating in a one-out-of-three configuration when a single channel is out of service.

Continuous RPS diagnostics are implemented predominantly through automated self-testing. Manual verification is performed periodically, during which processors, power supplies, and instrumentation are exercised and adjusted as necessary, with additional special test activities performed when required. When repairs or replacements are performed, qualified replacement components are pretested before installation and are functionally checked after installation to confirm correct operation before the system is returned to service.

Detailed RPS surveillance, inspection, and in-service test program, including specific test procedures, configurations, and intervals, will be established and documented in the Operating License Application and associated technical specifications.

7.5 ENGINEERED SAFETY FEATURES

Due to the U. of I. research reactor ESFs all being passive, as described in Chapter 6, no I&C ESF actuation systems (i.e., automatic initiation/actuation systems for active ESFs) are required or relied upon to mitigate consequences of postulated events.

7.6 SENSORS AND INSTRUMENTATION

7.6.1 Description

Sensors and instrumentation described in this section provide measurement inputs used by the facility I&C systems covered in this chapter for reactor protection, control, and monitoring functions. A summary of the sensors used by the subsystems discussed in this chapter is provided in Table 7-3. Instrumentation associated with facility support systems outside of the scope of this chapter is described only to the extent needed to identify interfaces with these Chapter 7 I&C systems.

As discussed in Section 7.4, the RPS monitors SR parameters to detect abnormal reactor conditions. The RPS field instrumentation includes {{ }}a(4) independent pressure transmitters and {{ }}a(4) temperature transmitters {{ }}a(4) (Section 4.3) to monitor primary coolant pressure and reactor inlet/outlet temperature. The associated sensing circuits are electrically continuous only with the RPS channel equipment. In addition to supporting the RPS protective functions, selected RPS channel measurements are provided to the RCS via buffered and electrically isolated outputs. {{ }}a(4)

Nuclear instrumentation for the facility is provided by {{ }}a(4) ex-core neutron flux detectors installed {{ }}a(4) in a {{ }}a(4) configuration to monitor each reactor {{ }}a(4). The ex-core neutron flux detectors are designed to provide logarithmic and linear outputs over the full range of normal operation. The signals from these detectors are processed by neutron monitoring channels. The RPS integrates those signals into trip voting and sends the nuclear instrumentation data via isolation to the control room HSI and RCS.

As discussed in Section 7.3, the RCS monitors parameters to adjust reactor power level, primary coolant conditions, secondary coolant conditions, and other equipment as applicable. RCS measurement inputs include {{ }}a(4) RCS-specific instrumentation for balance-of-plant and other non-protection measurements, as applicable. RCS instrumentation therefore includes sensors for detection of primary coolant mass flow, reactor vessel wall temperature, salt inlet and outlet temperature, and salt mass flow. The RCS also receives discrete inputs from the seismic monitoring system, RCCS, and fire protection system.

The RCSS includes instrumentation to monitor CRDU condition and control rod position. These include sensors for rod position feedback, pulse-count position measurement, torque measurement, and Resistance Temperature Detectors (RTDs) to measure the temperature within the CRDU windings and housing.

As discussed in Section 7.8, the UMSs comprise NSR monitoring subsystems such as molten salt activity monitoring, helium primary coolant leakage monitoring, radioactivity monitoring, and seismic monitoring. The UMSs are a grouping of lower-level monitoring subsystems, each with its own dedicated sensing arrangement tailored to the monitored variables. Detailed descriptions, including monitor types and locations, will be provided with the Operating License Application.

The instrumentation described in this section measures key reactor and plant process variables to support reactor protection, control, and monitoring functions. The expected operating ranges for these variables and the corresponding instrument measurement ranges will be established as the detailed instrumentation specifications are finalized. A table summarizing the measured variables and the anticipated ranges over which the selected instrumentation will operate will be provided in the Operating License Application.

Table 7-3 Sensors, Instrumentation, and Monitoring Channels in Scope of Chapter 7*

Parameter	Instrument / monitor Type	Quantity / channelization	Location	System
Reactor inlet temperature	Thermocouple	{{ }}a(4)	{{ }}a(4)	RPS {{ }}a(4)
Reactor outlet temperature	Thermocouple	{{ }}a(4)	{{ }}a(4)	RPS {{ }}a(4)
Primary coolant pressure	Pressure transducer	{{ }}a(4)	{{ }}a(4)	RPS {{ }}a(4)
Neutron flux	Neutron flux detector	{{ }}a(4)	{{ }}a(4)	RPS {{ }}a(4)
Reactor period	Derived from neutron flux	{{ }}a(4)	{{ }}a(4)	RPS {{ }}a(4)
Primary coolant mass flow rate	Flow transducer	{{ }}a(4)	{{ }}a(4)	RCS
Salt inlet temperature	Temperature transmitter	{{ }}a(4)	{{ }}a(4)	RCS
Salt outlet temperature	Temperature transmitter	{{ }}a(4)	{{ }}a(4)	RCS
Cold salt mass flow rate	Flow sensor	{{ }}a(4)	{{ }}a(4)	RCS
Reactor vessel wall temperature	Temperature transmitter	{{ }}a(4)	{{ }}a(4)	RCS
CRDU temperature	Resistance temperature detector	{{ }}a(4)	{{ }}a(4)	RCS
CRDU torque	Torque flange sensor	{{ }}a(4)	{{ }}a(4)	RCSS / shared with RCS
Control rod position	Resolver	{{ }}a(4)	{{ }}a(4)	RCSS / shared with RCS
UMS parameters	Varies	{{ }}a(4)	{{ }}a(4)	UMSs

*Sensor types, quantities and locations are not final. Updated details will be provided with the Operating License Application.

7.6.1.1 Classification

The RPS sensors and instrumentation support the safety functions of the RPS as described in Section 7.4.1.1 and therefore are SR. None of the other sensors and instrumentation perform any safety functions and are thus classified as NSR. As described in Section 3.6.2, the SR sensors and instrumentation are seismic Category I, and consequently designed in accordance with ASCE/SEI 43-19 (Reference 7-13). NSR sensors and instrumentation are designed in accordance with the seismic requirements of the local building code.

7.6.2 Design Bases and Criteria

In addition to the global PDC listed in Table 7-1, the following PDC from the NRC-approved TR (Reference 7-3) were used in the development of the Sensors and Instrumentation design bases:

- PDC-13: “Instrumentation and control”
- PDC-14: “Reactor helium pressure boundary”
- PDC-15: “Reactor helium pressure boundary design”
- PDC-20: “Protection system functions”
- PDC-21: “Protection system reliability and testability”
- PDC-22: “Protection system independence”
- PDC-23: “Protection system failure modes”
- PDC-24: “Separation of protection and control systems”
- PDC-29: “Protection against anticipated operational occurrences”
- PDC-30: “Quality of reactor helium pressure boundary integrity including reducing fluid ingress”
- PDC-31: “Fracture prevention of reactor helium pressure boundary”
- PDC-32: “Inspection of reactor helium pressure boundary”
- PDC-64: “Control of releases of radioactive materials to the environment”
- PDC-72: “Provisions for periodic Citadel inspection”

7.6.3 System Evaluation

The following discussion summarizes how the sensors and instrumentation satisfy the applicable PDC identified in Section 7.6.2. General compliance with PDC-1 through PDC-5 is addressed for all SSCs in Section 3.6. The sensors and instrumentation follow the I&C design and development, QA, and cybersecurity processes described in Section 7.2.5, with implementation controls commensurate with their safety classifications. General design criteria applicable to Chapter 7 systems are addressed in Section 7.2.4 and are not repeated here.

Consistent with PDC-13, the RPS sensors include nuclear instrumentation (neutron flux detectors) and process instrumentation (pressure and temperature) to detect abnormal process and nuclear conditions and initiate protective actions. Additional RCS instrumentation supports process control, including primary coolant mass flow rate and salt (intermediate loop) inlet/outlet temperature. The RCS monitors additional parameters which can affect the fission process and helium pressure boundary, such as CRDU condition and the reactor vessel wall temperature.

To the extent that sensor and instrumentation connections supporting the RCS, RPS and UMSs are mounted through or on reactor vessels forming the pressure boundary, PDC-14 applies to those pressure-boundary components. Where possible {{ }}$^{a(4)}$ thermowells are used to facilitate inspection, maintenance and replacement of sensors without breaching the pressure boundary. Consistent with PDC-30, these components are designed, fabricated, erected, and inspected to the same standards as the helium primary coolant pressure boundary (i.e., ASME Boiler and Pressure Vessel Code (BPVC); (Reference 7-25) to minimize leakage and prevent ingress of foreign fluids.

Similarly, compliance with PDC-15 and PDC-31 is demonstrated by designing the sensors, instrumentation and associated connections with appropriate stress, fatigue and fracture-prevention margins such that pressure boundary integrity is maintained during normal operation and postulated events, per the applicable ASME BPVC (Reference 7-25) requirements.

Consistent with PDC-32 (and also in accordance with the applicable BPVC requirements), sensing-related pressure-boundary components permit periodic inspection, pressure testing, and functional checks to verify structural and leak-tight integrity in coordination with the primary coolant pressure boundary inspection program, the details of which will be provided with the Operating License Application.

Consistent with PDC-20, the RPS-monitored sensors and associated signal paths are arranged in {{ }}$^{a(4)}$ redundant safety channels, with each channel including field-mounted sensors, signal conditioning equipment, and trip setpoint comparators that generate channel trip signals when measured values reach the trip setpoints. Channel trip signals are combined using redundant {{ }}$^{a(4)}$ coincidence logic such that a protective action initiation signal is generated {{ }}$^{a(4)}$ Those sensors and the associated logic ensure timely detection of abnormal conditions and provide the necessary information to the RPS for protective actions to be initiated.

Consistent with PDC-21, the {{ }}$^{a(4)}$ configuration of the RPS is implemented by separate, redundant signal processors and allows individual channels to be disabled for testing and maintenance without inhibiting the protective function. Periodic testing steps include verifying calibration status of test instrumentation, checking sensor calibration with "as-found" and "as-left" data, confirming input channel operability, confirming channel trip operability and setpoints, exercising logic test circuitry and diagnostics, and checking interlocks, inhibits, and bypasses. In addition, the RCS uses three independent measurements for selected process variables, allowing monitoring of sensor condition and rejection of outlier values.

Consistent with PDC-22, the RPS trip input sensors include both nuclear instrumentation (neutron flux) and process instrumentation (pressure and temperature), which provide diversity in sensing principles across credited trip inputs. The RPS architecture {{ }}$^{a(4)}$ separate, redundant signal processors, supporting independence between channels at the signal processing level and supporting overall reliability of the protection function.

Consistent with PDC-23, the RPS-related sensors are designed so that loss of a sensor or sensing channel does not defeat protective action and results in a conservative condition. Protective action initiation is based on {{ }}[a(4)] coincidence logic such that no single channel failure or loss of a single input channel prevents protective action when required. {{ }}[a(4)] consistent with maintaining protective capability during surveillance and testing activities.

Consistent with PDC-24, RPS trip input sensors and their associated sensing circuits are electrically continuous only with the RPS channel equipment. {{ }}[a(4)]

For postulated events, the control and protection sensor architecture provides multiple independent measurements to reduce vulnerability to single sensor or channel failures and to support high-probability automatic protective actuation, consistent with PDC-29. Sensors and instrumentation credited to protective actions will be specified to accommodate the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents/events, with detailed sensor specifications provided with the Operating License Application.

Consistent with PDC-30, the UMSs include a leakage monitoring system (Section 7.8) whose primary function is to identify helium leakage from the primary heat transfer system that is larger than expected leakage during normal operation. Dedicated helium detection sensors are located in rooms/building areas where the primary coolant and heat transfer system are located, including the Citadel building and Reactor/IHX cavities, and provide concentration information to the IICSs.

Consistent with PDC-64, the UMSs include RMS sensors that monitor dose levels and effluent radionuclide concentrations at locations in the unit and at the site boundary and provide alarms for assessment and response to abnormal radiation conditions. The RMS sensors include area monitors, airborne monitors, and process radiation monitors.

Consistent with PDC-72, the UMSs include a Citadel Building Monitoring System (CBMS), described in Section 7.8, which consists of an array of temperature sensors and strain gauges to support the surveillance of the Citadel building throughout the facility lifespan.

7.6.4 Testing and Inspection

Sensors and instrumentation credited to the RPS are designed to support routine surveillance and periodic testing using a combination of automated self-diagnostics and manual verification. A single sensor's channel may be bypassed for surveillance, testing, and maintenance without inhibiting the protective function. Routine inspection of the RPS sensors includes manual inspection of equipment conditions and mountings, wiring, and mode settings. Surveillance activities include functional checks of field instrumentation, alarms, diagnostics, and communication paths.

Sensors and instrumentation that provide inputs to the RCS are designed to support commissioning and surveillance, inspection, and testing. The RCS incorporates internal diagnostic monitoring to detect sensor malfunctions and is designed to allow comprehensive testing across the operating modes and states for which it is designed to operate.

UMS sensors and instrumentation are designed to support periodic surveillance, testing, calibration, and maintenance in accordance with system procedures and applicable standards and original equipment manufacturer guidance.

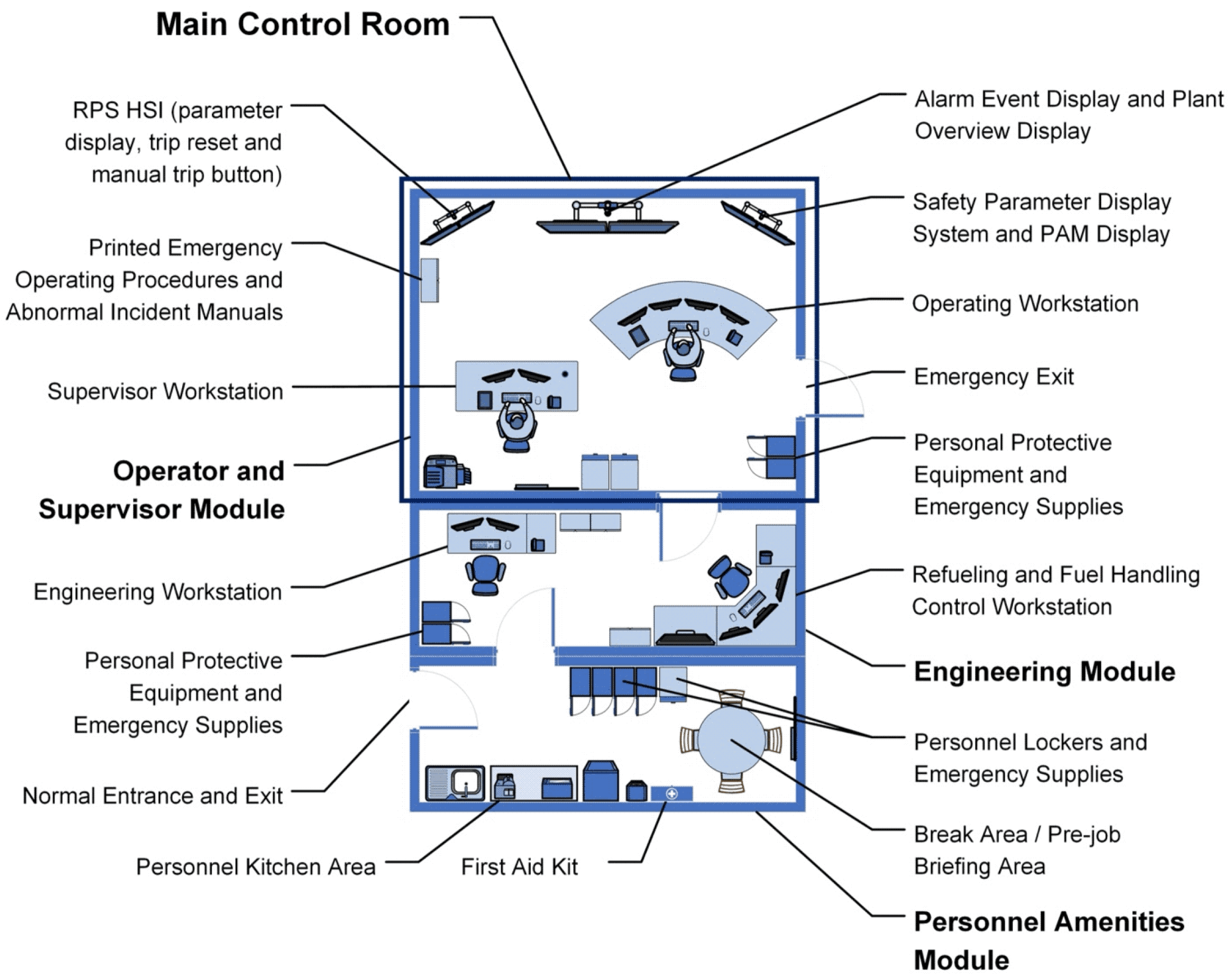
7.7 OPERATOR INTERFACES

7.7.1 Description

Operator interfaces to the I&C systems are implemented using HSI equipment (e.g., video displays with input devices and keyboards) in the MCR and SCA. {{ }}[a(4)] is the principal location for monitoring and controlling facility operations. The SCA, located in the Citadel building, provides an alternate control location with the capability to support safe shutdown and post-event monitoring if the MCR is not available. As described in Section 3.5, the MCR and SCA do not contain SR SSCs and perform no safety functions.

Consistent with PDC-13 and PDC-19, the primary operator interface in the MCR is the operating workstation, which is designed to support normal reactor operation by a single operator. A supervisory workstation in the MCR provides monitoring and higher-level information access (e.g., display and print functions and access to facility information) and does not provide control functions. The MCR HSI includes displays for system status and alarms, including RPS status display, and accepts UMS signals for display. Operator interfaces, including the operating workstation, incorporate access control mechanisms to prevent unauthorized operation of reactor systems. A conceptual layout for the MCR is shown in Figure 7-4.

Figure 7-4 Conceptual Control Center Layout. Acronyms: RPS = Reactor Protection System; HSI = Human-System Interface; PAM = Post-Accident Monitoring



Consistent with PDC-13 and PDC-19, in addition to the workstation-based HSI described above, operator interfaces provide RPS status indication and manual trip capability. Consistent with the facility design philosophy that credited safety functions are not dependent on operator action, the RPS performs automatic protective functions without reliance on operator input. As part of the RPS manual trip capability, hard-wired reactor trip pushbuttons are provided in both the MCR and SCA. Because no operator action is required for the safety functions credited to the RPS, the pushbuttons are classified as NSR. An NSR safety parameter display system and PAM system are provided in the MCR and SCA to support monitoring of safety parameters and reactor state following reactor shutdown or a postulated event.

Detailed HFE/HSI design, cybersecurity/access control implementation, and software lifecycle process details will be provided with the Operating License Application. The human-system interface concept, including the presentation and organization of I&C data in the MCR and SCA, will also be finalized and included with the Operating License Application.

7.7.1.1 Safety Functions

The MCR and SCA perform no safety functions.

7.7.1.2 Non-safety Functions

The MCR and SCA I&C perform the following non-safety functions:

- Display energy production information,
- Annunciate energy production conditions, including during startup, shutdown and off-normal conditions,
- Annunciate module conditions during refueling,
- Accept facility operations support direction,
- Effect facility operations support control,
- Report facility operations support information,
- Accept facility-level supervisory direction,
- Effect facility-level supervisory control, and
- Report facility-level supervisory information.

7.7.1.3 Classification

The MCR and SCA do not perform or support safety functions and therefore their operator interface equipment and associated controls are classified as NSR. Because the MCR {{ }}[a(4)] Because the SCA structural enclosure is part of the Citadel building, it inherits the Citadel building's classification as Seismic Category I. Consistent with the methodology described in Section 3.6.2, NSR equipment located within the MCR and SCA is classified as non-Seismic Category I in accordance with its safety classification.

7.7.2 Design Bases and Criteria

In addition to the global PDC listed in Table 7-1, the following PDC from the NRC-approved Topical Report (Reference 7-3) were used in the development of the design bases of the MCR and SCA:

- PDC-13: "Instrumentation and control"
- PDC-19: "Control room or secure remote monitoring facility"

7.7.3 System Evaluation

This section describes how the MCR and SCA satisfy the PDC identified in Section 7.7.2. The design of the MCR and SCA follows the QA and cybersecurity processes described in Section 7.2.5. General design criteria (PDC-1 through PDC-5) applicable to Chapter 7 systems are addressed in Section 7.2.4 and are not repeated here.

The MCR provides operators with the indications, alarms, and controls needed to monitor and control facility energy production, startup, shutdown, and associated support systems over the ranges expected for normal operation and postulated events. The SCA provides an alternate control location with the capability to initiate manual reactor trip and to monitor SR parameters and other key variables for PAM, supporting the ability to bring the reactor to and maintain it in a safe condition if the MCR is unavailable. This configuration supports conformance with PDC-13.

Hard-wired reactor trip pushbuttons are provided in both the MCR and SCA; these pushbuttons provide manual trip capability but are not credited towards a safety function. The MCR and SCA are designed to provide at least one control location that remains habitable and capable of performing supervisory and operational functions following postulated events, thereby demonstrating compliance with PDC-19.

7.7.4 Testing and Inspection

The MCR and SCA HSI design provides testability features applicable to the operator interfaces, including the capability to test I&C equipment. The design allows comprehensive testing across the modes and states for which the systems are designed to operate and includes testing equipment capable of recording data for commissioning and handover reports.

Surveillance, inspection, and testing of IICS equipment supporting MCR and SCA functions are implemented as part of preventive and corrective maintenance activities, including inspection, self-tests, calibration with verification, and functional checks. Routine maintenance includes manual inspection of equipment condition and mountings, wiring, and configuration settings. Surveillance includes functional checks such as diagnostic/alarm checks and communication network operation. Periodic maintenance is predominantly automated self-testing with manual verification, as applicable.

7.8 UNIT MONITORING SYSTEMS

7.8.1 Description

The UMSs provide NSR monitoring and indication to support facility operation, operator awareness, and post-event assessment. The UMSs perform sensing and data acquisition and transmit unit monitoring data to the IICSs via the data highway for operator display, recording, and higher-level evaluation. Where implemented, local subsystem processing (e.g., seismic local recording/central processing prior to data-highway transmission) supports data quality and availability but does not change the role of the IICSs for facility-level presentation and evaluation. Field sensors are primarily analog with digitization/processing in system cabinets/controllers.

Consistent with PDC-64, the UMSs include an RMS that monitors and supports control of on-site and public radiation exposure through monitoring of dose levels and effluent radionuclide concentrations at various locations in the reactor and at the site boundary and provides alarms to support assessment and response to abnormal radiation conditions. RMS capabilities include installed and portable radiation monitoring instrumentation suitable to support area, airborne, and process/radiological effluent monitoring functions; specific monitor types, locations, alarm setpoints, and sampling configurations will be detailed in the Operating License Application.

The RMS provides for determination of airborne activity concentration in accessible rooms in the radiologically controlled area and provides surface contamination meters in the radiologically controlled area. Radioactive release monitoring provides continuous measurement of HVAC exhaust air using particulate and gaseous monitoring. Environmental monitoring includes an external radiation measuring network installed close to the licensed site, with linked monitors provided at a minimum at the site boundary and local community areas. RMS monitoring signals are enabled for real-time capture and archiving of area monitor dose rates, dose rates for release monitoring, and alarm conditions through the data highway.

Each RMS monitor provides indication of measured value and audio/visual alarm state, both at the field location and in the MCR on the operating workstation. At the field location, a continuous reading and alarms warn personnel in the area when preset limits are exceeded. In the MCR, the measured value is displayed with audio and visual alarms when preset limits are exceeded. Response actions and coordination (including interaction with radiation protection personnel) will be addressed in the operating procedures provided in the Operating License Application.

In addition to the RMS, the UMSs include the following lower-level systems:

- The PAM system, which supports operator and operating-organization assessment of unit conditions during and following design basis accidents by providing a dedicated display capability for reactor status and other significant process and status parameters. The PAM has dedicated displays in the MCR and is implemented as an NSR system with monitored variables and equipment developed using an approach informed by the general intent of RG 1.97 (Reference 7-26), as applicable to this facility;
- The salt activity monitoring system, which is located in the secondary coolant loop and monitors activity in the molten salt system to support operator assessment of salt activation levels;
- The leakage monitoring system, which uses dedicated helium detection sensors in rooms/building areas where the primary heat transfer system is located (including the Citadel building and IHX cavities) and supports alarm generation via the IICSs using measurement and trend data;
- The helium activity monitoring/failed fuel detection system, which monitors activity levels within the primary coolant and supports alarm annunciation on the MCR operating workstation;
- The seismic monitoring system, which generates an alarm to indicate that a seismic event exceeding the design basis earthquake has occurred, with triaxial accelerometers including a free-field sensor and sensors located within the Citadel building;
- The CBMS, which uses an array of temperature sensors and strain gauges wired to a remote terminal unit that communicates via the data highway, with processed data and trend evaluation supporting alarm generation; and
- The ex-core instrumentation system, which supports verification/validation activities for the ex-core instrumentation of the RPS and RCS, and interfaces to the IICSs via remote terminal units and the data highway.

7.8.1.1 Safety Functions

The UMSs do not perform any safety function.

7.8.1.2 Non-safety Functions

Consistent with PDC-13, PDC-19, PDC-64 and PDC-72, the UMSs perform the following monitoring functions to support facility operation and protection:

- Monitor the radiological environment within the facility.
- Monitor the seismic environment within and the ground motion around the facility.
- Monitor radiological parameters associated with the salt process loop.
- Monitor radiological parameters associated with the helium process loop.
- Monitor helium concentration in the reactor and IHX cavities in the Citadel building.
- Monitor strain and temperature parameters of predefined locations in the Citadel building civil structure.
- Provide a post-event monitoring display capability for monitoring reactor status and significant unit process/status parameters during and following postulated accidents.
- Provide radiation monitoring (stationary and portable) to support control of onsite and public radiation exposure through monitoring of dose levels and effluent radionuclide concentrations at unit locations and the site boundary, including alarms for abnormal radiation conditions.

7.8.1.3 Classification

The UMSs are a group of lower-level monitoring systems that do not have any control or safety functions. All subsystems of the UMSs are, therefore, classified as NSR. Consistent with the methodology described in Section 3.6.2, the UMSs are therefore designed in accordance with the seismic requirements of the local building code.

7.8.2 Design Bases and Criteria

In addition to the general PDC presented in Section 7.2.2, the design bases and criteria for the UMSs are derived from the system-specific PDC listed below:

- PDC-13: “Instrumentation and control”
- PDC-19: “Control room or secure remote monitoring facility”
- PDC-64: “Monitoring radioactivity releases”
- PDC-72: “Provisions for periodic Citadel inspection”

Pressure-boundary design considerations for sensors and associated connections supporting UMS functions are addressed in Section 7.6.

7.8.3 System Evaluation

The following discussion summarizes how the UMSs satisfy the PDC identified in Section 7.8.2. The UMSs' design follows the QA and cybersecurity processes described in Section 7.2.5. General design criteria (PDC-1 through PDC-5) applicable to Chapter 7 systems are addressed in Section 7.2.4 and are not repeated here.

- In support of PDC-13, the UMSs include lower-level monitoring systems for post-event and safety parameter monitoring, salt activity monitoring, leakage monitoring, helium activity monitoring/failed fuel detection, radioactivity monitoring, seismic monitoring, and Citadel building monitoring. The UMSs communicate identified environmental and process parameters to the IICSs, which perform the coordination function of the facility, including transmitting information for display. Functions include continuous detection, indication and reporting of radionuclide concentrations and radiation levels, and continuous detection, indication, reporting and recording of seismic activity through all modes of operation. Pressure-boundary design, inspection, and testing considerations for sensing-related penetrations and associated connections are addressed in Section 7.6.
- Consistent with PDC-19, PAM provides a display system from which reactor status and other significant unit process and status parameters can be monitored during and following postulated events, with intent to provide sufficient information to assess unit conditions and support bringing the unit to a safe state during- and post-event. PAM includes dedicated displays in the MCR and SCA which, at a minimum, display parameters available from the RPS and additional parameters.
- Consistent with PDC-64, the UMSs include an RMS that monitors the Citadel building atmosphere, effluent flow paths and facility environs for radioactivity that may be released from normal operations and postulated events. The RMS includes area monitors, airborne monitors, and process radiation monitors located throughout the unit and at the site boundary. Environmental monitoring is implemented via a radiation measuring network installed close to the licensed site, with linked monitors provided, at a minimum, at the site boundary and local community areas; the quantity and placement of these monitors will be provided in the Operating License Application.
- Consistent with PDC-72, the UMSs include a CBMS which consists of an array of temperature sensors and strain gauges to support the surveillance of the Citadel building throughout the facility lifespan.

7.8.4 Testing and Inspection

UMS lower-level systems, including the RMS, undergo periodic surveillance, testing, calibration, and routine maintenance performed in accordance with system procedures, applicable international standards, and original equipment manufacturer requirements. Maintenance and test activities may involve temporary configurations, which are restored to permanent configurations upon completion of the activity.

UMS monitoring and display functions are supported through the IICS maintenance program for I&C systems, which includes inspection, self-tests, calibration with calibration verification, and functional checks. Routine maintenance includes manual inspection of equipment condition and mountings, wiring, and mode settings, and surveillance includes functional checks such as diagnostic/alarm checks and communication network operation. Periodic maintenance is predominantly implemented through automated self-testing with manual verification, with adjustments, comparisons, and resets performed as applicable.

7.9 REFERENCES

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7-24 International Society of Automation (ISA). ANSI/ISA 67.04.01-2018, Setpoints for Nuclear Safety-Related Instrumentation.

7-25 American Society of Mechanical Engineers (ASME). ASME Boiler and Pressure Vessel Code (BPVC).

7-26 U.S. Nuclear Regulatory Commission (NRC). Regulatory Guide 1.97, Rev 5, Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, 2019.

USNRC Project No. 99902094

UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 8 - ELECTRICAL POWER SYSTEMS

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
AC	Alternating Current
CRDUs	Control Rod Drive Units
DGs	Diesel Generators
HVAC	Heating, Ventilation, and Air Conditioning
I&C	Instrumentation & Control
MMR	Micro Modular Reactor
NFPA	National Fire Protection Association
PDC	Principal Design Criteria
RCCS	Reactor Cavity Cooling System
RPS	Reactor Protection System
SSCs	Structures, Systems, and Components
U. of I.	University of Illinois Urbana-Champaign

CHAPTER 8 ELECTRICAL POWER SYSTEMS

This chapter discusses and describes the electrical power systems designed to support reactor operation at the facility. The information in this chapter is provided under two categories: normal electrical power systems and backup electrical power systems.

8.1 INTRODUCTION

This chapter outlines the systems in place to supply and manage electrical power for the University of Illinois Urbana-Champaign (U. of I.) research reactor facility, distinguishing between normal electrical power system (Section 8.2) and backup electrical power system (Section 8.3).

The facility receives its primary electrical service from local utilities which deliver alternating current (AC) power. This electrical power supports all routine reactor functions, including instrumentation and control (I&C), monitoring equipment, and auxiliary infrastructure. Major electrical loads include the helium circulator, molten salt heat tracing, and heating, ventilation, and air conditioning (HVAC) equipment.

To ensure continuity of critical operations during a loss of normal electrical power, the research facility is equipped with a backup power system capable of supplying electrical loads to support essential operational convenience, and to maintain the integrity of sensitive equipment. These systems are described in detail in Section 8.3. Due to the passive safety features of the reactor design, safety related structures, systems, and components (SSCs) do not depend on electrical power to perform their safety functions. No electrical power is required to mitigate postulated events and thus AC power from off-site or backup power sources is not required to ensure safe reactor shutdown and public health and safety. Both the normal and backup electrical power systems are classified as non-safety related.

The following sections provide a comprehensive overview of the design basis, capabilities, and configurations of both the normal and backup electrical power systems, including their role in supporting reactor safety and operational reliability. Figure 8-1 provides a high-level diagram of the reactor's major electrical system components.

The below Principal Design Criteria (PDC), as outlined in U. of I. Topical Report IMRDD-MMR-23-06-A, Release 02, "University of Illinois Urbana-Champaign High-Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria," (Reference 8-1), align with the electrical power systems described in this chapter:

- Consistent with PDC-1 – "Quality standards and records", The electrical power system will be designed, fabricated, erected, and tested to meet relevant industry quality standards commensurate with its function, and appropriate records of those activities shall be maintained.
- Consistent with PDC-17 – "Electrical power systems", Electrical power is not required for anticipated operational occurrences or postulated events. The design will demonstrate that power for functions important to safety is provided.
- Consistent with PDC-18 – "Inspection and testing of electrical power systems", Electrical power system components important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be

designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among systems.

Figure8-1 U. of I. Research Reactor Electrical Configuration Diagram

{{

}}a(4)

8.2 NORMAL ELECTRICAL POWER SYSTEM

The normal electrical power system provides the primary source of electrical power for the operation of the reactor and its supporting infrastructure. This includes facility auxiliaries, control, protection, and surveillance systems. Normal electrical power will be supplied by an off-site power source from the local utility grid. This system supports normal facility operations and is not required to maintain safe reactor shutdown or ensure public health and safety during a loss of off-site electrical power. A loss of voltage or a degraded voltage condition of the normal electrical power supply from the local utility grid will not adversely impact the safety functions of the facility.

8.2.1 Design Basis

The normal electrical power system is designed to maintain the functions required for all phases of normal facility operation, including energy production, shutdown, refueling, and startup activities. To ensure facility availability and enable online maintenance, the electrical power system will be designed with a 2 x 100% redundant configuration.

The normal electrical power system does not perform any safety functions, nor is it required to mitigate postulated events. Consequently, it does not contribute to safe reactor shutdown capability during postulated events, and no technical specifications are required.

The reactor electrical power distribution systems will be designed in accordance with National Fire Protection Association (NFPA) 70 National Electrical Code (Reference 8-2), and any other applicable local codes and standards.

8.2.2 Design Description

8.2.2.1 AC Electrical Power

AC power distribution is normally supplied by an off-site power source from a local utility grid through a medium voltage feeder. From the point of connection, this voltage is stepped down via an appropriate transformer to three-phase 480 VAC 60 Hz. The low voltage AC distribution system operates at three-phase 480 VAC 60 Hz for major and process loads, and 208Y/120 VAC 60 Hz for smaller loads.

8.2.2.2 DC Electrical Power

The low voltage direct current (DC) power system provides reliable 48 VDC to the I&C systems including the monitoring systems and the Reactor Protection System (RPS). One DC system serves non-essential loads (Citadel building monitoring systems), while {{ }}[a(4)] independent DC systems supply essential loads (e.g. to the RPS). The DC system includes batteries, chargers, and distribution switchgear. Its design minimizes common-cause failures and ensures high reliability for critical facility functions. The non-safety backup power provisions are further described in Section 8.3.

8.2.2.3 Other Power System Considerations

Grounding and lightning protection systems provide a low resistance path to dissipate ground fault current or lightning discharge at the facility. These systems do not perform any safety functions.

The electrical power system design will include measures to maintain electrical and signal integrity. This may involve the implementation of components for suppression, filtering, noise control, and protection against external influences where appropriate. Provisions for managing transient conditions and disturbances will be considered to support system reliability. Interference control and shielding practices will be applied as needed to ensure proper functioning of critical I&C systems. Attention will also be given to maintaining signal quality and preventing unintended activations in sensitive circuits. The design of electrical wiring will prevent inadvertent electromagnetic interference between the electrical power service and safety related I&C circuits. These considerations will be detailed in design documentation to demonstrate adherence to applicable requirements and standards for the Operating License Application.

8.2.3 Safety and Accident Considerations

In the event of a loss of normal electrical power, whether short-term or long-term, the reactor is designed to shut down safely without reliance on active electrical systems. As described in Chapter 7, the RPS is engineered to detect abnormal conditions and respond by de-energizing key components, thereby performing safety functions without requiring electrical power. For example, upon a loss of normal electrical power the following will occur:

- Control rod drive units (CRDUs) deenergize, allowing the control rods to drop into the core under gravity, inserting strong negative reactivity and rapidly shutting down the reactor.
- Helium circulator deenergizes, halting the coolant flow, and resulting in an increase of the reactor core temperature. This increase in temperature results in a strong insertion of negative reactivity due to the negative temperature reactivity coefficients, rapidly shutting down the reactor.

These passive responses alone are sufficient to each bring the reactor to a safe subcritical state, independently. They each ensure a swift and reliable transition to a safe shutdown condition, even in the absence of electrical power. Essential monitoring and control functions such as the Citadel building radiation monitors are maintained through backup batteries, or uninterruptible power supplies (UPS), and backup diesel generators (DGs) upon a loss of normal electrical power. These systems do not have any safety function.

The Citadel building HVAC system maintains a slight negative pressure and exhaust to a monitored release path during normal operations and postulated events where AC power is maintained or restored by the backup electrical power system. The reactor fuel relies on the functional containment concept described in Section 6.2 and does not require electrical power to perform its safety functions.

8.2.3.1 Short- and Long-Term Power Loss Scenarios

In the event of a loss of off-site power, the CRDUs are deenergized, causing the control rods to automatically insert and shut down the reactor. Immediately following the electrical power loss, UPS and DC batteries maintain selected facility loads {{

}}a(4) Within seconds to a few minutes, backup diesel generators start and supply electrical power to select systems. Along with the essential monitoring and control systems, the diesel generators will provide electrical power to the heat tracing system {{

}}a(4) If all backup power sources are eventually depleted, passive systems such as the reactor cavity cooling system (RCCS) continue to provide passive heat removal, ensuring the reactor remains in a safe and stable condition. The loss of electrical power to the active I&C components of the RCCS has no safety impact as the passive removal of decay heat is still guaranteed. The normal electrical power system does not perform any safety functions.

During a long-term loss of power, the RCCS automatically transitions to passive operation without operator action. Section 6.3 provides a detailed description of the RCCS operation.

8.2.4 Testing and Inspection

The normal electrical power system will be designed to permit periodic inspection and testing of wiring, insulation, connections, and switchboards to verify continuity and component condition. The system will include provisions to periodically test the operability and functional performance of on-site power sources, relays, switches, and buses, as well as the overall system operability under conditions that simulate design operation. Those details will be included in the Operating License Application.

8.3 BACKUP ELECTRICAL POWER SYSTEM

Emergency electrical power is not required to initiate a reactor shutdown or mitigate postulated events. Therefore, AC power from off-site or backup electrical power sources is not required to ensure safe reactor shutdown and public health and safety. No technical specifications are required for the backup electrical power system. Non-safety related, continuous backup power will be supplied to emergency lighting, the fire alarm system, security systems, and monitoring systems. Backup power can be supplied from batteries, UPS, diesel generators, or a combination of equipment. Non-safety related backup AC power will be provided to select loads for operational convenience and integrity of research assets. The backup electrical power systems are further described throughout this section.

8.3.1 Design Basis

As described in Section 8.2, the electrical power system (normal and backup power systems) does not provide any safety functions. All safety related SSCs that utilize electricity are designed to fail safely, independent of their electrical supply in the event of a loss of normal electrical power. Therefore, the backup electrical power system is not required for reactor safety or for protecting public health and safety. The system maintains essential facility operations and monitoring capabilities during a loss of normal AC power but are not required for safe shutdown of the reactor in the event of a loss of normal electrical power.

The backup electrical power system is designed in accordance with National Fire Protection Association (NFPA) 70 National Electrical Code (Reference 8-2), and any other applicable local codes and standards.

8.3.2 Design Description

In the event of a loss of normal AC power, the backup electrical power system will supply electrical power to select facility loads. The low-voltage backup electrical power system consists of UPS, DC battery systems, and AC diesel generators that are sized to provide sufficient power to the selected loads.

The facility employs a passive cooling approach to manage decay heat that does not rely on backup electrical power or active systems following a loss of normal electrical power. The non-safety related backup power system will supply selected monitoring systems, I&C systems, and select facility loads. This will be further defined for the Operating License Application.

8.3.2.1 Uninterrupted Power Supplies and Battery Supplies

Selected non-safety non-motor loads will receive continuous, regulated backup electrical power via a UPS system for operational continuity, as shown in Figure 8-1. During normal operations, the UPS receives 480 VAC input power from the AC main normal electrical power system and powers low voltage AC loads through either a bypass transformer or through a rectifier/inverter in series that allows a backup battery to supply AC loads in the event of a loss of AC main normal electrical power. This system supplies 120 VAC, single-phase, 60 Hz power to non-motor I&C loads. The UPS performs no safety functions. The UPS is sized to provide sufficient backup electrical power to the selected loads to maintain functionality during the startup sequence of the backup diesel generators.

The non-safety related DC battery backup power systems supply 48 VDC power to select loads, as shown in Figure 8-1. These systems are not required to mitigate postulated events or ensure a safe reactor shutdown. {{

}}a(4) The batteries operate at a nominal voltage of 48 VDC and supply power to both essential and non-essential DC systems when the normal electrical power source is unavailable. Each of the {{ }}a(4) essential DC buses and the non-essential DC bus have their own dedicated backup battery, sized to provide the required backup electrical power during a loss of normal electrical power. Additionally, the batteries provide the DC reference voltage required for battery charger startup.

A complete description of non-safety UPS and battery systems, and the required non-safety backup electrical power to the selected systems will be provided in the Operating License Application, including power requirements and duration for effluent, process, and area radiation monitors.

8.3.2.2 Backup Generators

Upon loss of normal electrical power, backup diesel generators will automatically start to supply electrical power to essential facility loads. The design will follow a 2x100% redundant configuration philosophy, providing two 480 VAC diesel generators with separate output breaker connections capable of delivering 480 VAC to select loads to maintain operational continuity. The selected loads and their duration of support will be defined in the Operating License Application. The generators will be sized at a minimum to provide power to selected non-safety related loads. Preliminary considerations include post-shutdown neutron monitors, radiation detection systems, reactor vessel temperature and pressure systems, essential lighting, security and communication systems, as well as molten salt heat tracing systems.

8.3.3 Testing and Inspection

The backup electrical power system does not perform any safety functions. The system will incorporate features that allow periodic inspection and testing of critical components, including diesel generators, UPS units, batteries, and associated switchgear. Testing will ensure operability and functional performance of these components and verify the ability of the system to transfer power and maintain selected essential loads under simulated loss-of-power conditions.

8.4 SYSTEMS EVALUATION

The electrical power system design satisfies the following PDC:

- PDC-1 (see Section 8.2.1 and Section 8.3.1): The electrical power system is designed in accordance with National Fire Protection Association (NFPA) 70 National Electrical Code (Reference 8-2) and any other applicable local codes and standards.
- PDC-17 (see Section 8.2.3, Section 8.3, and Section 8.3.1): The electrical power system is designed to provide power to important to safety SSCs {{ }}[a(4)] Safety related SSCs do not rely on electrical power to perform their safety functions during anticipated operational occurrences or postulated events. The normal and backup electrical power systems are classified as non-safety related, and isolation between safety related and non-safety related circuits is not required. Failures in normal or backup electrical power systems will not affect reactor trip capability or other essential safety functions.
- PDC-18 (see Section 8.2.4 and Section 8.3.3): Electrical power systems important to safety are designed to allow periodic testing and inspection to verify operability and performance.

8.4.1 Considerations for Operating License Application

The following design details of the normal and backup electrical power systems will be addressed in the Operating License Application:

- Ranges of normal and backup electrical power capability required for reactor operation and utilization,
- Special processing of the normal and backup electrical service,
- Design and performance specifications of principal components (standard and non-standard),
- Special routing or isolation of wiring or circuits for operations and experimental facilities,
- Deviations or exceptions from national or local electrical standards, and
- Spectrum of reactor operations and associated electrical power requirements.

8.5 REFERENCES

8-1 U. of I. Topical Report IMRDD-MMR-23-06-A, Release 02, "University of Illinois Urbana-Champaign High-Temperature Gas-cooled Research Reactor: Micro Modular Reactor (MMR™) Principal Design Criteria," dated July 30, 2024 (ADAMS Accession Number ML24208A066).

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USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 9 - AUXILIARY SYSTEMS

Revision 0

submitted by

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March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
AFU	air filtration units
AHU	air handling unit
ALARA	as low as is reasonably achievable
CRDU	Control Rod Drive Unit
dBA	A weighted decibal
DOT	Department of Transportation
EPABX	Electronic Private Automatic Branch Exchange
FCU	fan coil units
{{ }}a(4)	{{ }}a(4)
FHA	Fire Hazard Analysis
{{ }}a(4)	{{ }}a(4)
FHSS	Fresh Fuel Handling and Storage System
FM	Frequency Modulation
FPS	Fire Protection System
HPS	Helium Purification System
HVAC	Heating Ventilation and Air Conditioning System
IHX	Intermediate Heat Exchanger
LEU	low enriched uranium
MCR	Main Control Room
MHz	megahertz
MMR	Micro Modular Reactor
NFPA	National Fire Protection Association
PA	Public Address
PDC	Principal Design Criteria
PSTN	public switched telephone network
PZ1	Pressure Zone 1
PZ2	Pressure Zone 2
RPP	Radiation Protection Program
SARRDLs	specific acceptable radionuclide release design limits
SCA	Secondary Control Area
{{ }}a(4)	{{ }}a(4)
SNM	Special Nuclear Material
SSCs	Structures, Systems or Components
TRISO	TRi-structural ISOtropic
UHF	Ultra High Frequency
UMTS	Universal Mobile Telecommunications System
UPS	Uninterruptible Power Supply

CHAPTER 9 AUXILIARY SYSTEMS

This chapter provides an overview description of the auxiliary systems that are part of the research reactor facility. Auxiliary systems are those systems not previously described elsewhere in the Construction Permit Application.

Details are provided for those auxiliary systems that are important to the safe operation and shutdown of the reactor or to the protection of the health and safety of the public, the facility staff, and the environment to support an understanding of those aspects of the design.

This chapter covers the research reactor facility Heating Ventilation, and Air Conditioning System (HVAC), the Fuel Handling and Storage System (FHSS), the fire protection system, the communication systems, the possession and use of byproduct, source, and Special Nuclear Material (SNM), the water services systems, and lifting & rigging equipment.

9.1 HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS

9.1.1 Reactor Facility Heating, Ventilation, and Air Conditioning System Summary Description

The primary functions of the reactor facility HVAC system are to condition the air in areas where facility workers may have to carry out their activities, to ensure adequate environmental conditions to ensure proper equipment function, and to control airborne contaminants.

The reactor facility HVAC system consists of two subsystems:

- HVAC System - Pressure Zone 1 (PZ1) which serves the areas of relatively low potential for contamination.
- HVAC System - Pressure Zone 2 (PZ2) which serves the areas of relatively high potential for contamination.

9.1.2 Functions, Classification and Design Criteria

9.1.2.1 Safety Functions

The reactor facility HVAC system is not credited for performing any safety functions or mitigation of postulated events. It is therefore classified as non-safety related and is not credited in safety analyses for performing any function required to meet the specific acceptable radionuclide release design limits (SARRDLs) or offsite dose limits; its role is limited to providing defense-in-depth for control of airborne radioactivity and environmental conditions in pursuit of the as low as is reasonably achievable (ALARA) principle.

9.1.2.2 Non-Safety Functions

PZ1 Non-Safety Functions:

- Condition the facility environment to maintain acceptable temperatures and humidities for operators and equipment.
- Maintain occupational exposures pursuant to ALARA.
- Prevent the spread of contamination.

PZ2 Non-Safety Functions:

- Condition the facility environment to maintain acceptable temperatures and humidities for operators and equipment.
- Maintain occupational exposures ALARA.
- Prevent the spread of contamination.
- Provide pressure release paths to maintain the facility environment.
- Controlling leakage (pressure) and protecting the capability to control leakage.

9.1.2.3 System Classification

The reactor facility HVAC system is non-safety related. Components which could fail due to an earthquake and impact the functionality of safety related structures, systems, or components (SSCs) are designed to Seismic Category I to prevent such a condition.

9.1.3 Design Bases

9.1.3.1 Principle Design Criteria

Principal Design Criteria, identified in the U. of I. Topical Report, “Micro Modular Reactor (MMR) Principal Design Criteria” (Reference 9-1), was used to develop design bases for the SSCs in the nuclear plant. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC were identified as applicable to the reactor facility HVAC system:

- PDC-19, “Control room or secure remote monitoring facility”
- PDC-60, “Control of releases of radioactive materials to the environment”
- PDC-64, “Monitoring radioactivity releases”

9.1.3.2 Code of Federal Regulations Criteria

The reactor facility HVAC system is designed to meet:

- 10 CFR 20, “Standards for Protection Against Radiation” (Reference 9-2)

9.1.3.3 Codes and Standards

The following codes and standards have been identified as relevant to the design of the reactor facility HVAC system. This is not a complete list of applicable codes and standards and inclusion in this list is not a commitment to comply with all aspects of the codes and standards.

- ANSI/HPS N13.1, Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts of Nuclear Facilities, 2021 (Reference 9-13)
- ANSI/SMACNA-005, Round Industrial Duct Construction Standards, 2013 (Reference 9-14)
- ANSI/SMACNA-006, HVAC Duct Construction Standards - Metal and Flexible, 2020 (Reference 9-15)
- ASHRAE 55, Thermal Environmental Conditions for Human Occupancy, 2020 (Reference 9-16)
- ASHRAE 62.1, Standards for Ventilation and Indoor Air Quality, 2025 (Reference 9-17)
- ASHRAE 90.1, Energy Standard for Sites and Buildings Except Low-Rise Residential Buildings, 2022 (Reference 9-18)
- ASME AG-1, Code on Nuclear Air and Gas Treatment, 2023 (Reference 9-19)
- NFPA 105, Standard for Smoke Door Assemblies and Other Opening Protectives, 2025 (Reference 9-20)
- NFPA 204, Standard for Smoke and Heat Venting, 2024 (Reference 9-21)
- NFPA 80, Standard for Fire Doors and Other Opening Protectives, 2025 (Reference 9-22)
- NFPA 801, Standard for Fire Protection for Facilities Handling Radioactive Materials, 2020. (Reference 9-5)
- NFPA 90A, Standard for the Installation of Air-Conditioning and Ventilating Systems, 2024 (Reference 9-23)
- NFPA 90B, Standard for the Installation of Warm Air Heating and Air Conditioning Systems, 2024 (Reference 9-24)
- NFPA 91, Standard for Exhaust Systems for Air Conveying of Vapors, Gases, Mists, and Particulate Solids, 2026 (Reference 9-25)
- NFPA 92, Standard for Smoke Control Systems, 2024 (Reference 9-26)
- RG 1.140, Design, Inspection, And Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants, 2016 (Reference 9-27)
- SMACNA, HVAC System Duct Design, 2023 (Reference 9-28)
- SMACNA, HVAC Systems Testing, Adjusting and Balancing, 2024 (Reference 9-29)
- SMACNA, Rectangular Industrial Duct Construction Standards, 2024 (Reference 9-30)
- UL 1995, Heating and Cooling Equipment, 2022 (Reference 9-31)
- UL 555, Standard for Fire Dampers, 2026 (Reference 9-32)
- UL 555S, Standard for Smoke Dampers, 2025 (Reference 9-33)
- UL 586, High-Efficiency, Particulate, Air Filter Units, 2022 (Reference 9-34)
- UL 900, Standard for Air Filter Units, 2015 (Reference 9-35)

9.1.4 Design Description

The reactor facility HVAC system is functionally designed for controlling air temperatures, humidities, and controlling airborne contamination.

The areas for PZ1 are:

- Secondary Control Area (SCA)
- Citadel access area
- Storage and laydown area
- Reactor equipment area
- Radiation waste and decommissioning area
- HVAC room
- Change over area
- Main Control Room (MCR)
- Security room
- Electrical room

The areas allocated to PZ2 are:

- Reactor equipment room
- Intermediate Heat Exchanger (IHX) cavity
- Reactor cavity
- Control Rod Drive Unit (CRDU) cavity

Those areas that are not currently detailed in the Construction Permit Application will be detailed in the Operating License Application.

9.1.4.1 Description of PZ1 Components

The major components of PZ1 include an air handling unit (AHU), fan coil units (FCU), radiators, air filtration units (AFU) and exhaust fans.

- **AHU:** The AHU conditions outside air prior to distribution throughout PZ1. The AHU, as necessary, adjusts humidity and temperature. Filtration is also included.
- **FCUs:** Each FCU recirculates, filters, heats and/or cools, and dehumidifies air within its responsible area.
- **Exhaust AFU:** The AFUs in PZ1 provide filtration of air prior to discharge to the environment. These AFUs include dampers such that the exhaust path can be isolated if necessary.
- **Exhaust Fan:** The exhaust fans in PZ1 provide unfiltered discharge of air to the environment.
- **Radiators:** Radiators provide supplemental heating to the area served. Hot water flow through the radiator is regulated to maintain the appropriate temperature in the served area.

9.1.4.2 Description of PZ2 Components

The major components of PZ2 include an AFU, dampers, FCUs, a dehumidifier, rupture discs, and a vacuum pump.

- **FCU:** Each FCU recirculates, filters, heats and/or cools, and dehumidifies air within its responsible area.
- **AFU:** The AFU is responsible for filtering PZ2 exhaust, when necessary, to reduce offsite releases. Appropriate filtration stages are provided to ensure releases are ALARA.
- **Filter and Coil Box:** The filter and coil box provides filtered and conditioned air intake for PZ2. It provides a similar function to the AHU in PZ1 but relies on the PZ2 AFU to provide the suction force necessary to draw air into the unit.
- **Dehumidifier:** The dehumidifier is located in the reactor equipment room (to facilitate maintenance) which draws air from the reactor cavity, condenses out the moisture, and returns the air to the reactor cavity to maintain humidity in the reactor cavity to the desired range.
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  }}a(4)
- **Rupture Discs and Ductwork:** Rupture discs are located in ductwork which bypasses the AFU. These discs are selected to rupture before the pressure from a line break would cause damage to the AFU. The ductwork which comprises this bypass is designed to retain the increased pressure from a medium or smaller pipe break.
- **Vacuum Pump:** A vacuum pump is included in PZ2 to maintain the negative pressure in the CRDU cavity. This pump cycles on and off to keep the pressure within the acceptable range. The pump discharges to the PZ2 exhaust ductwork.

9.1.5 Basic Operational Overview

The reactor facility HVAC system is not credited for any safety or radiation control function during normal and postulated events in order to satisfy exposure target limits for the workers, public and environment. Therefore, potential failures within the system will not impact the ability to safely shut down the reactor or control releases within allowable limits. PZ1 services the areas with a relatively low likeliness of airborne contamination. These areas include all of the services building and part of the Citadel building ({{ }}a(4) and the Citadel access area). PZ2 services the areas with a higher likeliness of airborne contamination. These areas are limited to rooms in the Citadel building in close proximity to the reactor core. Figure 9-1 and Figure 9-2 show details about the Citadel building area serviced by PZ1 and PZ2. In these figures, the areas shown in red have relatively high potential for contamination while the areas in green have decreasing potential for contamination as distance from the reactor core increases. PZ1 and PZ2, where necessary, provide temperature and humidity control to protect the function of all SSCs in the areas served and acceptable conditions for operators. Contamination is also controlled by PZ1 and PZ2 to ensure occupational doses and releases are ALARA. The operation of each system is fully automated and allows limited changes in setpoint temperatures in certain areas such as the Main Control Room (MCR) and SCA. Temperature and humidity measurements are used as inputs to this control system. Filters are equipped with differential pressure measurements and alarms to notify the operators of a need for a filter change.

Figure 9-1 Citadel Building Radiological Zoning Section View

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Figure 9-2 Citadel Building Radiological Zoning Floor Plan
(Section C-C from Figure 9-1)

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9.1.5.1 PZ1 Basic Operational Overview

PZ1 is designed to distribute conditioned air, bring in sufficient outside air, control temperatures, and control humidities in the areas served. The PZ1 AHU supplies conditioned outside air to the entirety of PZ1. The AHU supplies air at a setpoint temperature. Additional FCUs and/or radiators are included where necessary to further adjust room temperatures. Exhaust AFUs and exhaust fans are used to exhaust air from the building and maintain the desired airflow distribution. Exhaust AFUs are provided in areas where small quantities of contamination may be present such as the decontamination areas. These areas are maintained at a slight negative pressure to prevent spreading of contamination to other areas. The Unit Monitoring Systems (see Section 7.8) monitor airborne radioactivity in key areas of PZ1 and provide the ability to identify potential off-normal conditions. These areas of PZ1 can be isolated, under operator action, from the air supply and exhaust to prevent the spread of contamination until the situation can be remedied.

9.1.5.2 PZ2 Basic Operational Overview

PZ2 is designed to maintain acceptable temperatures, and control humidities in the areas served and, depending on the operating mode, confine or filter and exhaust contaminants. During normal operation, the FCUs provide normal cooling to the reactor equipment room and IHX vessel cavity. {{

}}$^{a(4)}$

Radiation monitoring equipment is included {{

}}$^{a(4)}$ If contamination is acceptably low, a bypass and exhaust fan are included to allow for direct exhaust. {{

}}a(4) Appropriate alarms are included to identify abnormal contamination levels (See Section 7.8 Unit Monitoring Systems). All discharges from PZ2 are monitored through the PZ2 common exhaust. PZ2 does not provide temperature control to the CRDU cavity or reactor cavity but does provide dehumidification to the reactor cavity and pressure control to both the reactor cavity and the CRDU cavity. {{

}}a(4) Controlled releases from the reactor cavity are provided by a bubble tight damper which opens when the reactor cavity pressure reaches the top of the desired range. This discharges to the PZ2 exhaust ductwork. {{

}}a(4) Dehumidification is provided to the reactor cavity to prevent condensation on the RCCS and corrosion of the RCCS and reactor vessels. {{

}}a(4)

PZ2 also connects to the Helium Purification System (HPS) (Section 5.4) to accommodate discharges. The HPS discharge line is directly connected to the exhaust ductwork of PZ2.

9.1.6 System Evaluation

PDC-1

The reactor facility HVAC system is non-safety related and is not credited for mitigating the consequences of postulated accidents. It is fabricated and tested in accordance with suitable codes and standards (Section 9.1.3.3). The established quality assurance program (Reference 9-3) is followed through design, construction, commissioning and operations to ensure satisfactory performance of the systems' function. This is consistent with PDC-1.

PDC-2

The reactor facility HVAC system is non-safety related, therefore only components which could potentially fail during an earthquake and impact the function of safety related SSCs are designed to meet Seismic Category I requirements. This classification is applied primarily to SSCs in the Citadel building. A complete list of Seismic Category I SSCs will be provided in the Operating License Application. This is consistent with PDC-2.

PDC-3

The reactor facility HVAC system is designed and its components are located to minimize the probability and effects of potential fires. This is accomplished through the minimization of flammable materials, use of fire-resistant materials, and appropriate integration with fire detection and protection systems. This is consistent with PDC-3.

PDC-4

The reactor facility HVAC system SSCs are designed to be compatible with the environmental conditions during normal operating conditions, including maintenance, and testing. The reactor facility HVAC system is non-safety related and therefore not designed for or credited in postulated events but is designed to prevent adverse interactions with safety related equipment. This is consistent with PDC-4.

PDC-19

The reactor facility HVAC system provides outside air, temperature control, and humidity control to the MCR during normal operations to ensure a suitable environment. During postulated events, the reactor facility HVAC system is not credited in achieving the PDC-19 dose limits in the control room and is not required to support occupancy for the duration of the event. This is consistent with PDC-19.

PDC-60

The reactor facility HVAC system provides a filtered release path for gaseous effluents but is not credited for mitigating the consequences of postulated events. This is consistent with PDC-60.

PDC-64

The reactor facility HVAC system provides a common path for monitoring releases to the environs during normal operations and postulated events. This is consistent with PDC-64.

10 CFR 20

The reactor facility HVAC system is designed to keep radiation exposure ALARA and below the limits of 10 CFR 20 by:

- providing 100% outside air to prevent the recirculation of any contaminants,
- directing air from areas of low potential contamination to areas of high potential contamination,
- by maintaining appropriate pressures in contaminated areas to control the spread of contamination, and
- by providing filtration of exhausts airflows.

This also minimizes contamination of the facility and the environment to facilitate eventual decommissioning. This is consistent with 10 CFR 20.

9.1.7 Testing and Inspection

The system will be monitored and periodically functionally tested. HVAC filters will also be periodically replaced. A complete testing and inspection plan will be included in the Operating License Application. Technical requirements will be included in Chapter 14 in the Operating License Application.

9.2 HANDLING AND STORAGE OF REACTOR FUEL

9.2.1 Fuel Handling and Storage System Summary Description

The FHSS is responsible for managing the fresh fuel between receipt and refueling, performing refueling of the reactor and managing the spent fuel between removal from the reactor and shipment offsite. To prevent radionuclide release, the system is designed to maintain sufficiently low fuel temperatures and to prevent damage to fuel during handling or storage. Appropriate shielding and sealed handling mechanisms are provided during all modes of spent fuel handling to maintain occupational doses ALARA. The refueling sequence is designed to ensure sufficient margin to criticality in all core states throughout refueling. Similarly, criticality is prevented all throughout fuel handling and storage by design relying on the fuel block geometry and their subcritical configuration. The FHSS system only interfaces with the reactor after it has been shut down and in a stable and safe state. Fuel is protected against theft or diversion all throughout fuel handling and storage. This includes physical barriers and monitoring systems. Details will be provided in the Physical Security Plan discussed in Chapter 12 in the Operating License Application.

9.2.2 System Functions, Classifications and Design Basis

9.2.2.1 Safety Functions

There is no safety functions associated with the FHSS.

9.2.2.2 Non-Safety Functions

- Manage fuel and reflector block elements from site delivery to placement into the reactor and from removal from the reactor and off-site shipment
- Transport fuel and core elements between the reactor core and temporary storage areas
- Install and remove Control Rod Drive Unit (CRDU) assemblies.
- Prevent leakage of contaminants into or out of the reactor vessel during fuel handling operations
- Provide for decay heat removal from fuel elements
- Provide material control accounting and physical protection of the fuel
- Prevent criticality in all states of FHSS operation including fuel storage
- Maintain occupational exposures and offsite releases ALARA

9.2.2.3 System Classification

The FHSS is non-safety related as there are no postulated fuel handling events which have been determined to exceed release limits. This is, in part, due to the robustness of the fuel and its ability to retain fission products. See Chapter 4.2.1 for discussion on the fuel and Chapter 13 for discussion on potential releases during fuel handling. Where necessary to preclude damage to safety related systems, the FHSS is designed to Seismic Category I.

9.2.3 Design Basis

Principal Design Criteria, identified in the U. of I. Topical Report, "Micro Modular Reactor (MMR) Principal Design Criteria" (Reference 9-1), was used to develop design bases for the SSCs in the nuclear plant. In addition to the general PDC discussed in Section 3.6.1.4, the following specific PDC were identified as applicable to FHSS.

- PDC-61, "Fuel storage and handling and Radioactivity control"
- PDC-62, "Prevention of criticality in fuel storage and handling"
- PDC-63, "Monitoring fuel and waste storage"
- PDC-64, "Monitoring radioactivity releases"

Additionally, the FHSS is designed to meet:

- 10 CFR 20, "Standards for Protection Against Radiation" (Reference 9-2)
- 10 CFR 70, "Domestic Licensing of Special Nuclear Material" (Reference 9-4)

9.2.4 Design Description

The FHSS is responsible for the management and handling of new and spent fuel as well as the refueling process. {{

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9.2.4.1 {{ }}a(4)

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9.2.4.1.1 {{ }}a(4)

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}}a(4)

9.2.4.1.2 {{ }}a(4)

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9.2.4.2 {{ }}a(4)

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9.2.4.2.1 {{ }}a(4)

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9.2.4.2.2 {{ }}a(4)

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9.2.4.2.3 {{ }}a(4)

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9.2.4.2.4 {{ }}a(4)

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9.2.4.2.5 {{ }}a(4)

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9.2.4.2.6 {{ }}a(4)

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9.2.4.3.4 {{ }}a(4)

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9.2.5 Basic Operational Overview

The fuel handling and storage system has three main operational phases: receipt of new fuel and placement into the reactor, alignment of equipment and refueling of the reactor, removal of spent fuel and shipment offsite. These concepts are detailed in Figure 9-3, Figure 9-4, and Figure 9-5, respectively. During all phases of operation, the movement of fuel is administratively tracked and logged to ensure traceability of the fuel all the way from fresh fuel delivery through offsite shipment of the spent fuel.

When new fuel arrives at the facility, it is inspected and placed under material control and accounting in the fresh fuel handling area. {{

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Prior to receiving the fuel, the reactor must be prepared for refueling. After shutting down for refueling, the reactor will be allowed to cool down {{ }}a(4) prior to start of the refueling operations. Once decay heat and activity levels have been reduced sufficiently, the reactor is prepared for refueling. During refueling, one of the CRDUs is removed to allow access to the reactor. {{

}}a(4)

Figure 9-3 Fuel Receipt and Placement into the Reactor

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Figure 9-4 Alignment of Equipment and Refueling of the Reactor

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Figure 9-5 Removal of Spent Fuel and Shipment Offsite

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9.2.6 System Evaluation

PDC-1

The FHSS is non-safety related and is not credited for mitigating the consequences of postulated accidents. The FHSS is fabricated and tested in accordance with generally recognized codes and standards, to be provided in the Operating License Application. A quality assurance program (Reference 9-3) is also established and implemented to ensure satisfactory performance of the systems' function. This is consistent with PDC-1.

PDC-2

FHSS components which could potentially fail during an earthquake and impact the function of a safety related SSC are designed to Seismic Category I. This includes all components which interface with the reactor or are in the vicinity of spent fuel. During a seismic event, spent fuel handling equipment is designed to, but not credited, prevent a fuel handling accident. {{

}}$^{a(4)}$ These design features address PDC-2.

PDC-3

The FHSS is designed and its components are located to minimize the probability and effects of potential fires. This is accomplished through the minimization of flammable materials and use of fire-resistant materials. This is consistent with PDC-3.

PDC-4

FHSS components which interface with the reactor or irradiated core components (fuel/shield/reflectors) are designed for the anticipated maximum temperature of the reactor pressure boundary interfaces, the core helium, and the core components as appropriate. FHSS components which interface with the reactor pressure boundary are anticipated to operate at near atmospheric pressure, however, they are designed to withstand internal pressure and elevated temperatures. The FHSS components which handle spent fuel are designed to maintain appropriate temperatures during refueling, transport, and interim storage. There are no anticipated internal missiles, pipe whipping, or discharging fluids anticipated to impact the FHSS. These design features address PDC-4.

PDC-61

{{ }}$^{a(4)}$ are designed with adequate radiation shielding in place throughout the refueling process. Additionally, the reactor is kept at below atmospheric pressure to confine airborne contamination from the reactor core. All equipment which handles potentially radioactive material will be hermetically sealed during its transport either to the reactor core, from the reactor core to temporary storage, and from temporary storage to the spent fuel repository. These design features address PDC-61.

PDC-62

The fresh fuel handling area will have provisions and procedures in place that ensure that subcritical geometry is always kept. {{

}}[a(4)] The spent fuel handling and interim storage system is designed to store fuel in a subcritical configuration relying on arrangement and neutron absorbers, as necessary. These design features address PDC-62.

PDC-63 and 10 CFR 70.24

Temperature sensors are assigned to all areas that interact with spent fuel in a means to detect abnormal conditions. Additionally, all active or passive systems required to remove decay heat will be continuously monitored to ensure that system performance is as expected. The FHSS is designed such that criticality is prevented by arrangement of the fuel elements and criticality monitoring systems will be implemented. The details of this system will be provided as part of the Operating License Application.

PDC-64

During refueling activities, discharges from the maintenance enclosure, including fuel storage areas, are monitored by the radioactivity monitoring system. Additional monitors are included in fuel storage areas as discussed above. These design features address PDC-64.

10 CFR 20

The FHSS is designed to provide sufficient shielding of radiation sources and to confine contaminants within the boundaries of fuel handling equipment and casks. Spent fuel is shipped offsite as soon as practical. The system is designed to minimize the need for operator action in the vicinity of radiation sources. These features support the minimization of occupational exposures, minimization of contamination, and support eventual decommissioning. These design features address 10 CFR 20.

9.2.7 Testing and Inspection

A complete testing and inspection plan will be included in the Operating License Application and will detail decontamination procedures for the refueling equipment. Appropriate technical requirements will be included in the Operating License Application.

9.3 FIRE PROTECTION SYSTEMS AND PROGRAMS

The fire protection program for the U. of I. research reactor will be implemented as a set of planned, coordinated, and controlled activities such as maintenance, inspections, training, and testing that ensure the facility's fire protection goals are achieved and that transparent processes exist to confirm the required activities are performed. A complete description of the fire protection program, how it integrates with the emergency plan, the effects SSCs may have on safety or licensed materials protection, and the fire hazard analysis will be provided in the Operating License Application.

9.3.1 Summary Description

The Fire Protection Systems (FPS) are designed to detect, control and extinguish fires so that a continuing fire will not prevent safe reactor shutdown or result in an uncontrolled release of radioactive material that exceeds the SARRDLs. The FPS include fire detection and alarm systems as well as automatic and manual fire suppression systems. FPS elements and design features will be identified and installed in defined fire areas and fire zones based on the results of the fire hazard analysis. Features of the FPS will include:

- Passive fire protection features such as fire rated floors, walls, ceilings and doors.
- Fire detection systems, including smoke and heat detectors.
- Fire alarms and manual pull stations.
- Active fire suppression systems such as fire hydrants, standpipes / hose stations, and automatic sprinkler systems.
- Where required by the fire hazard analysis, clean agent gaseous systems for sensitive equipment rooms, control spaces, and Instrumentation and Control (I&C) areas.
- Portable fire extinguishers.

In summary, the main objectives of the FPS are to:

1. Minimize the risk of radiological releases to the public due to fire,
2. Protect facility occupants from injury or death due to fire,
3. Protect safety-related SSCs so that fires or the FPS themself do not prevent safety functions to be performed,
4. Minimize economic loss resulting from fire damage to SSCs, and
5. Minimize the impact on the environment of potential radioactive and hazardous material releases due to fire.

9.3.2 Functions, Classifications and Design Bases

9.3.2.1 Safety Functions

The FPS perform no safety function.

9.3.2.2 Non-Safety Functions

The FPS is primarily responsible for fire prevention, detection, mitigation, and control and has the following key functions:

1. Aid in fire prevention,
2. Speedily detect and announce the occurrence and location of combustion by-products or fire within the facility,

3. Promptly control and extinguish fires that occur, and

4. Provide protection for SSCs such that the performance of their safety functions is not diminished.

The FPS are classified as a non-safety related system because it does not perform any active or passive fundamental safety functions and is not credited during postulated events.

9.3.2.3 Design Bases

PDC, identified in the U. of I. Topical Report, "Micro Modular Reactor (MMR) Principal Design Criteria" (Reference 9-1) were used to develop design bases for the FPS. Only the general PDC discussed in Section 3.6.1.4 apply to the FPS.

9.3.3 Design Description

9.3.3.1 Fire Detection and Alarms

A combined manual and automatic fire detection and alarm system will be provided in all nuclear plant fire zones, i.e. areas of the facility where SSCs important safety need to be safeguarded from fire. The intent of the detection and alarm systems are to give early warning of dangerous conditions and enable safe evacuation. The system is designed and installed in accordance with applicable National Fire Protection Association (NFPA) standards and includes detectors, manual pull stations, fire alarm control panels, annunciation, batteries, and cabling required for automatic operation.

Fire detectors are installed in safety related areas, adjacent areas, and non-safety related areas with high fire loads or ignition sources, with the objective of detecting fires quickly enough to limit damage and support timely suppression.

9.3.3.2 Fixed Fire Suppression Systems

The reactor will primarily rely on non-water-based fire suppression technologies for the Citadel building and areas outside of the Citadel building containing sensitive electronic equipment such as the MCR, and water-based fire suppression systems for other structures present at the facility. For the Citadel building, fixed clean-agent fire extinguishing systems are provided for the {{ }}[a(4)], electronic equipment rooms, and electrical house structure. Fixed clean-agent extinguishing system inside the Citadel building will be designed to meet Seismic Category I criteria. These systems will be designed in accordance with NFPA 2001 (Reference 9-7) and consist of high-pressure cylinders connected to manifolds, piping, and discharge nozzles, with automatic and manual actuation capabilities and associated audio-visual alarms.

The clean-agent systems use cross-zoned detection logic. Upon actuation of the first detector, local fire alarms are initiated; upon confirmation by a second detector, a release signal is sent to the fire alarm control panel, a time delay is initiated to allow evacuation, ventilation units are shut down, and fire dampers are closed. At the end of the delay, the solenoid valves open and discharge the clean agent into the protected room, with feedback to the main fire alarm control panel confirming discharge.

The facility also makes use of water-based automatic sprinkler and water-spray systems in selected areas outside of the Citadel building, supplied from a dedicated firewater storage tank with redundant fire pumps, designed in accordance with NFPA 13 (Reference 9-8), NFPA 15 (Reference 9-9), NFPA 20 (Reference 9-10), and NFPA 22 (Reference 9-11). These systems include status monitoring and alarm actuation upon operation.

9.3.3.3 Portable Fire Extinguishers

Portable fire extinguishers are located at strategic positions along escape routes and in occupied spaces throughout the reactor facility. Extinguishers are wall-mounted and selected in type, rating, size, and quantity based on the classification of fire hazards in each protected area, in accordance with NFPA 10 (Reference 9-12).

9.3.3.4 Fire Water Supply and Water Based Systems

A dedicated firewater system will be present, consisting of a firewater storage tank sized to supply the worst-case automatic water-based suppression demand plus hose stream allowance for worst-case duration as determined in the Fire Hazard Analysis (FHA) with two redundant 100 percent capacity fire pumps (one diesel-driven and one electric-motor-driven), pressure-maintenance equipment, and distribution piping designed in accordance with NFPA 20 (Reference 9-10) and NFPA 22 (Reference 9-11).

9.3.4 System Evaluation

PDC-1

The FPS conform to applicable building and fire codes. Additionally, the FPS will be designed compliant to NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors" and NFPA 801 (Reference 9-5). Life safety provisions are included in the facility design in accordance with the Life Safety Code, NFPA 101 (Reference 9-6).

PDC-2

The FPS are non-safety related but may be physically located near safety related SSCs. The FPS in the Citadel building are designed using clean-agent fire suppression systems rather than water and are designed to meet Seismic Category I criteria so that postulated failures of the FPS or actuation of the FPS will not preclude a safety related SSC from performing its safety function. This is consistent with PDC-2.

PDC-3

Safety related SSCs and equipment required for safe shutdown of the reactor are located within the Citadel building. The floors, walls, and ceilings of the Citadel building are constructed almost entirely of {{ }}[a(4)] concrete, and by design the Citadel building minimizes the amount of combustible material present in the building. Fire detection and suppression capability provided by the FPS minimize the potential for adverse effects of fires on safety related SSCs and those required for safe reactor shutdown. For selected areas of the facility outside of the Citadel building, fire water piping is used that is not located near safety-related SSCs so that a rupture or inadvertent operation of the FPS will not significantly impair safety related or safe reactor shutdown functions. These design features, in conjunction with the fire protection program, satisfy PDC-3. The details of the fire protection program plan will be provided with the Operating License Application.

PDC-4

The FPS will be designed to be compatible with the environmental conditions during normal operating conditions, including maintenance, and testing. The FPS are non-safety related and therefore not designed for or credited in during postulated events but is designed to prevent negative interactions with safety related equipment. The FPS in the Citadel building are designed using clean-agent fire suppression systems rather than water and are designed to meet Seismic Category I criteria so that postulated failures of the FPS or actuation of the FPS will not preclude a safety related SSC from performing its safety function. This is consistent with PDC-4.

9.3.5 Testing and Inspection

Functional tests of FPS are performed prior to startup and periodic functional tests and inspections of the system will be performed after installation. Technical requirements for testing and surveillance will be provided with the Operating License Application and included in the fire protection program.

9.4 COMMUNICATION SYSTEMS

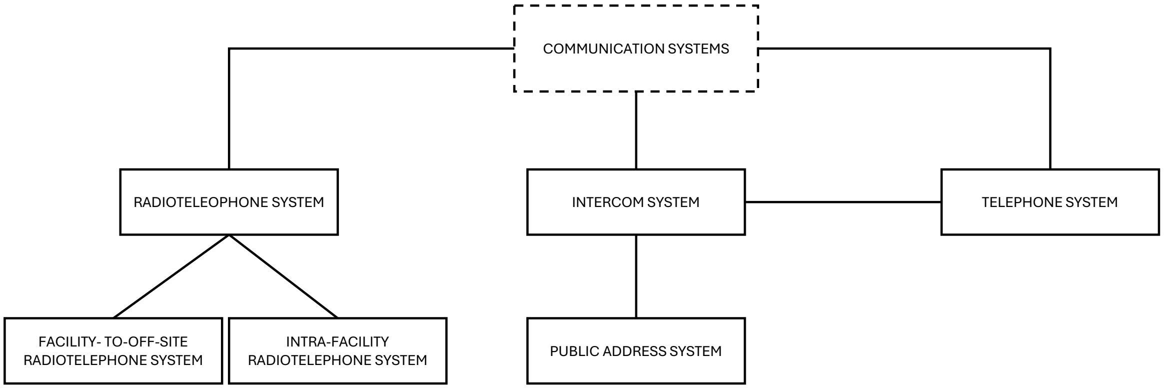
9.4.1 Summary Description

The communication systems consist of four separate subsystems:

- Telephone system: The telephone system provides facility-wide intercom capability for private conversation between personnel via Electronic Private Automatic Branch Exchange (EPABX).
- Radiotelephone system: The radiotelephone system provides a means of site internal communication (intra- facility) between facility personnel, or facility personnel and external parties (facility-to-off-site), when other methods are unavailable.
- Public Address system: The Public Address (PA) system provides one-way communication through a paging system to distribute voice instructions and emergency warning signals to all areas of the facility including associated buildings and facilities, and outdoor perimeter areas.
- Intercom system: The intercom system provides two-way communication and is functionally integrated with and operates through the PA system. The intercom system is also integrated with the telephone and PA systems.

The purpose of these systems is to ensure dependable communication between critical areas of the facility and remote locations, both during normal operations and postulated events. The communication systems are designed such that failure of one type of communication does not impair the reliability of the other communication systems. This capability is provided by the four diverse communication systems, and the communication systems function as an integral part of the Information, Instrumentation, and Control System discussed in Section 7.1. Figure 9-6 below illustrates the hierarchy of the communication systems and its four subsystems.

Figure 9-6 Hierarchy of the Communication Systems and its Four Subsystems



9.4.2 System Functions, Classification and Design Criteria

9.4.2.1 Safety Functions

The communication systems perform no safety functions.

9.4.2.2 Non-Safety Functions

The functions of the communication systems are to facilitate communications internal and external to the facility in the event of normal and emergency operations.

9.4.2.3 System Classification

The communication systems do not perform or support any fundamental safety function and are therefore classified as non-safety related consistent with the methodology described in Section 3.6. While not safety related, the communication systems have been assessed against applicable PDC to ensure that any accident- or emergency-related functions it may perform has been adequately evaluated.

Consistent with its non-safety related classification, the communication systems are Non-Seismic Category I.

9.4.3 Design Basis

The PDC used to develop design criteria are provided in the approved Topical Reports referenced in Section 3.6.1. No additional PDC beyond those discussed in Section 3.6.1.4, were identified as applicable to the communication systems.

9.4.4 Design Description

9.4.4.1 Telephone System Design Description

The telephone system is one of the sub-systems of the communication systems. The telephone system provides reliable communication between essential areas of the facility, and remote locations both during normal operations and postulated events. The telephone system provides facility-wide intercom capability for private conversation between personnel via EPABX. The structure of an EPABX typically includes the following components:

- Telephone exchange – a computer server or other piece of specialized hardware that is responsible for managing the EPABX,
- Phone lines,
- Extension numbers,
- Key system units,
- Phone handsets,
- Cable and wiring, and
- Distribution panel and junction boxes, if applicable.

The telephone system is powered by an Uninterruptible Power Supply (UPS) and does not share cables with the PA and intercom systems. It is provided with adequate isolation and redundancy between equipment. If part of the system is non-functional or damaged, the remaining sections remain operational. Section 8.2 contains additional information about the backup electrical power system and UPS. The telephone system provides communications capability in the following areas:

- MCR,
- SCA, and
- Services Building.

All telephones within the reactor facility are connected to the EPABX or the EPABX network switch. The EPABX is connected to the public switched telephone network (PSTN) via trunk lines. Hence, all telephones can use one external voice line on a time-shared basis. The EPABX includes features such as call routing, call forwarding, voicemail, auto-attendant, call recording, call monitoring, conference calling. The connection between the PSTN and the Mobile Switching Center used by cellular networks such as Global System for Mobile Communications/Code Division Multiple Access /Universal Mobile Telecommunications System (UMTS) is inherent to how the telephone system operates at U. of I. Using these mobile cellular networks, a cell phone user can connect to any telephone set in the reactor facility using an extension number.

Fixed telephone stations are provided and located strategically throughout the facility to support general communication needs. The telephone system is designed to be compatible with the PA system and intercom system and is kept available during incidents for emergency and two-way communications.

9.4.4.2 Radiotelephone System Description

The radiotelephone system provides a means of communication between facility personnel when other methods are unavailable. These systems consist of two-way radio systems (including mobile and hand-held radios) that are kept available during normal operation, maintenance outages, and emergency conditions. Systems are operated at licensed frequencies (i.e., bands between 25 to 960 megahertz [MHz]). Dual-frequency working, or duplex, splits the communication into two separate frequencies, but only one is used to transmit at a time with the other frequency dedicated to receiving which allows one person to talk and the other to listen alternately. The full duplex is used in cell phones and requires the radio system to simultaneously transmit and receive on two separate frequencies. The half-duplex is common to hand-held radios.

The radiotelephone systems support the following functions:

- provide two-way voice communication and be able to have two or more party communication facilities for personnel engaged in commissioning, operation, and maintenance of equipment,
- maintain functionality in 'clean' and radioactive contaminated areas, and
- be capable of receiving immediate and automatic reception of all emergency signals and messages by all persons connected to the system.

The two-way radio consists of features such as multiple channels, shock resistant, voice-activated transmission, Ultra High Frequency (UHF)/Frequency Modulation (FM) – ultra-clear long-distance reception, privacy codes, water resistant, long range, batteries and charger.

9.4.4.2.1 Intra-Facility Radiotelephone System

The intra-facility radiotelephone system, a subsystem of the radiotelephone system, is operational in the UHF band, and fully functional in all accessible facility areas. The intra-facility radiotelephone system is configured with the following features:

- compatibility with headsets that can be attached to a standard hard hat,
- usage of noise canceling microphones and noise attenuating earpiece units to ensure that the noise levels at the users' ears do not exceed 85 A weighted decibels (dBA),
- ability for individuals wearing equipment under protective clothing to establish communication without needing to remove any part of their clothing when they are in areas where such attire is required,
- radiotelephone equipped with rechargeable batteries that have a minimum operating time of 36 hours for listening and 6 hours for talking, and
- radiotelephone operable in both Simplex and repeater modes.

9.4.4.2.2 Facility-to-off-site Radiotelephone System

The facility-to-off-site radiotelephone system, a subsystem of the radiotelephone system, provides a means of communication between facility personnel and off-site parties when other methods are unavailable. The facility-to-off-site radiotelephone system provides two-way voice communication and acts as an emergency backup to the telephone and the intra-facility radiotelephone system.

9.4.4.3 Public Address System Description

The PA system is a one-way communication solution provided through a paging system to distribute voice instructions and emergency warning signals to all areas of the facility including the Citadel building, associated buildings and facilities, and the outdoor perimeter areas.

The integrated PA system and intercom system provides communication through a server-based PA system with talk back facility for two-way communication in full duplex.

The PA system is server-based and supports the following functions:

- Allows routine announcements (voice paging) during normal operations,
- Provides communication between all areas of the facility, and the outdoor perimeter areas,
- Provides automatic broadcasting of "stored safety voice messages/alarm tones" over the PA system during emergency situations. These messages can be safety/security/fire alarm/evacuation instructions/public utility or general messages,
- Provides communication between the PA system and the EPABX/telephone system,
- Provides communication between the PA system and the radiotelephone system, and
- Interfaces with external visual beacons (signaling devices) in high noise areas.

The PA system will be designed to have a sound level of at least 10 dB above the background noise in each area and will maintain a sound level below 60 dB in the office area.

In high noise areas (85 dB or greater), the alarms will consist of visible and audible signals. In low noise areas, the alarms will consist of audible signals. The visual alarm indicators will meet the applicable requirements for proper functionality and usability.

In areas that are subject to a wide variation of noise levels, amplifiers' volumes will be set for the maximum noise levels. Where a reduced output level is required, individual volume control will be provided for each speaker. The MCR microphone signals will be equipped with noise cancelling and delay line to prevent acoustic feedback.

The alarm signals will have the highest priority and override all voice announcements. They will be initiated by a manual switch for General and Evacuate alarms. Fire alarms will be generated manually or automatically with operator reset ability.

The alarm signals are interruptible for voice announcements. Only the top two priority levels of microphones, PA and intercom systems, have this function.

Three distinct alarm sounds will be clearly detectable by frequency and format:

- General alarm,
- Fire alarm, and
- Evacuate alarm.

The alarm sounds and signals will be identical to the existing PA system. Provisions will be made for the recording of PA announcements with a time stamp.

The PA system has the following different zones that may be selected individually or in combination:

- Reactor Facility, and
- Site boundary.

9.4.4.4 Intercom System Description

The intercom system allows two-way communication and is designed to be interactive - the parties on the receiving end of the message can respond directly to the sender. Broadcasting of intercom voice instructions are limited to selected areas within the facility.

The intercom system is functionally integrated with and operates through the PA system and is additionally integrated with the telephone and PA systems.

The intercom system consists of accessories such as the intercom station with built-in microphone, integrated dynamic speaker and microphone cables. The intercom system is powered by the normal electrical power system, which passes through the backup electrical power system UPS. If the normal electrical power system becomes unavailable, the UPS will supply power to the intercom system. The UPS power source for the intercom system is independent from other communication systems' power sources. The intercom system will not share common cables with other communication systems.

9.4.5 System Evaluation

PDC-1

The communication systems are non-safety related and are not credited for mitigating the consequences of postulated events. The communication systems are designed, fabricated, erected, and tested to quality standards commensurate with the systems' functions to be performed. This is consistent with PDC-1.

PDC-2

The communication systems are non-safety related, therefore the communication systems are not designed to meet Seismic Design Category I requirements. Only those components which could potentially fail during an earthquake and potentially impact the function of safety related SSCs are designed to meet Seismic Design Category I requirements. This is consistent with PDC-2.

PDC-3

The communication systems utilize non-combustible and fire-resistant materials, along with physical separation where practicable, to minimize potential adverse effects on safety related SSCs. These design features, in conjunction with the fire protection system (Section 9.3), provide assurance that the communication systems demonstrate conformance with PDC-3.

PDC-4

The communication systems are designed to be compatible with the environmental conditions during normal operating conditions, including maintenance, and testing. The communication systems are non-safety related and therefore not designed for or credited during postulated events but are designed to prevent negative interactions with safety related equipment. This is consistent with PDC-4.

9.4.6 Testing and Inspection

The diverse implementation of the communications systems permits routine testing and inspection without disruption to normal communications. Testing of the communication systems is used to detect and correct any problems or degradation.

Technical requirements for testing and inspection of the communication systems will be included in the Operating License Application.

9.5 POSSESSION AND USE OF BYPRODUCT, SOURCE, AND SPECIAL NUCLEAR MATERIAL

Special nuclear material (SNM), and byproduct material will be present at the research reactor facility. There is no source material used in the research reactor facility. The applicable requirements in 10 CFR Part 30, 10 CFR Part 40, and 10 CFR Part 70 will be satisfied using content contained within this Construction Permit Application. However, material license(s) are not being requested at this time and necessary license application(s) or amendments will be submitted at a future date. This section describes the systems that interact with the SNM or byproduct material, and the design basis for those systems to prevent uncontrolled release of radioactive materials and to maintain personnel exposure limits within 10 CFR Part 20 dose limits and ALARA objectives. Additional information on ALARA practices is discussed in Chapter 11.

Spaces in which these materials are handled and equipment used to handle them, are subject to the licensee's administrative controls to minimize contamination, to prevent radiological sabotage, theft or diversion, and to prevent uncontrolled release of the materials. A description of the administrative procedures related to use of byproduct and SNM will be provided in the Operating License Application. The amounts of SNM and byproduct material expected to be handled throughout the life of the facility will also be discussed in the Operating License Application.

Waste from SNM, source material, or byproduct material is handled through the radioactive waste management program described in Section 11.2.

9.5.1 Special Nuclear Material

The facility will receive SNM shipped in Department of Transportation (DOT) certified shipping containers containing prismatic fuel block assemblies. SNM will be managed in compliance with 10 CFR Part 70. The SNM consists of low-enriched uranium (LEU) used as the kernel of tri-structural-isotropic particles (TRISO) that are micro-encapsulated in a fully ceramic (SiC) matrix. The fresh fuel will be stored onsite in a vault-type storage facility. Per 10 CFR 73.2, vault-type rooms provide security via locked doors protected by an intrusion alarm which creates an alarm upon entry, exit, or movement of a person within the room.

SNM is handled in the fuel intake area, the FHSS, and the reactor vessel. Use, storage, and transfer of radioactive materials is governed by the Radiation Protection Program (RPP) described in Section 11.1.2. The RPP includes radiation detection instrumentation to allow for measuring ambient dose rates, monitoring effluent releases or contamination at the reactor facility. Section 9.2 details how the FHSS protects against criticality and uncontrolled release of SNM. The reactor vessel is designed to ensure radionuclide release is within the SARRDLs, per Section 4.3. Other features that contribute to the prevention of uncontrolled releases are the functional containment features discussed in Chapter 6, and the HVAC systems discussed in Section 9.1.

9.5.2 Source Material

There is no source material used in the research reactor facility.

9.5.3 Byproduct Material

Byproduct materials are generated during reactor operation as a result of the fission process and subsequent neutron activation reactions. Byproduct material is present in the reactor systems, RCCS, HPS, Thermal Energy Storage System, Citadel building, and radioactive waste management systems. Byproduct radionuclides within these systems may include tritium, C-14, Na-24, Ar-41, Mn-56, Kr-85, Sr-90, I-131, Cs-137 and Xe-135. The startup neutron source Cf-252 and its decay products are considered byproduct material as well. Further details on radiation sources, byproducts, and system locations can be found in Section 11.1.1.

Byproduct materials are managed in compliance with 10 CFR Part 30. The HPS includes a sampling system for gaseous media discussed in Section 5.4.4 which monitors the quality of the primary coolant to ensure radiological impurity levels remain within acceptable limits. The HPS is additionally responsible for maintaining purity of the primary coolant. The RPP monitoring equipment referenced in Section 11.1.2 detects radioactivity produced by operation of the reactor.

9.6 COVER GAS CONTROL IN CLOSED PRIMARY COOLANT SYSTEMS

The primary coolant is pressurized helium. Use of a cover gas is not applicable to the research reactor. The Helium Purification System (Section 5.4) and Primary Coolant Make-up System (Section 5.5) are used for controlling gas composition and quantities.

9.7 OTHER AUXILIARY SYSTEMS

9.7.1 Water Services Systems

The water services systems consist of several subsystems that distribute or condition water throughout the reactor facility.

9.7.1.1 Functions, Classification and Design Criteria

9.7.1.1.1 Safety Functions

The water services systems perform no safety functions.

9.7.1.1.2 Non-Safety Functions

The water services systems supply demineralized water to users around the plant.

The water services systems supply chilled water through the chilled water subsystem to applicable systems, in order to remove heat as necessary to maintain operability of those systems.

The water services systems supply a source of hot water through the hot water subsystem to applicable systems, in order to supply heat as necessary for system processes.

The water services systems distribute potable water across the operations building facilities through the potable water distribution subsystem.

The water services systems remove sanitary effluent from the operations building facilities through the sewage subsystem.

9.7.1.1.3 System Classification

The water services systems do not perform any safety functions and are not credited for the mitigation of postulated events. The water services systems are designed such that a malfunction does not impact safety functions of nearby or interfacing safety related SSCs.

Non-safety related components which could fail as a result to a seismic event and impact the functionality of safety related SSCs are designed to Seismic Category I to prevent such an occurrence.

9.7.1.2 Design Basis

The general PDC discussed in Section 3.6.1.4 apply to the RCCS as well as the following ones:

- PDC-44, "Structural and equipment cooling"
- PDC-45, "Inspection of structural equipment cooling systems"
- PDC-46, "Testing of structural and equipment cooling systems"

9.7.1.3 Design Description

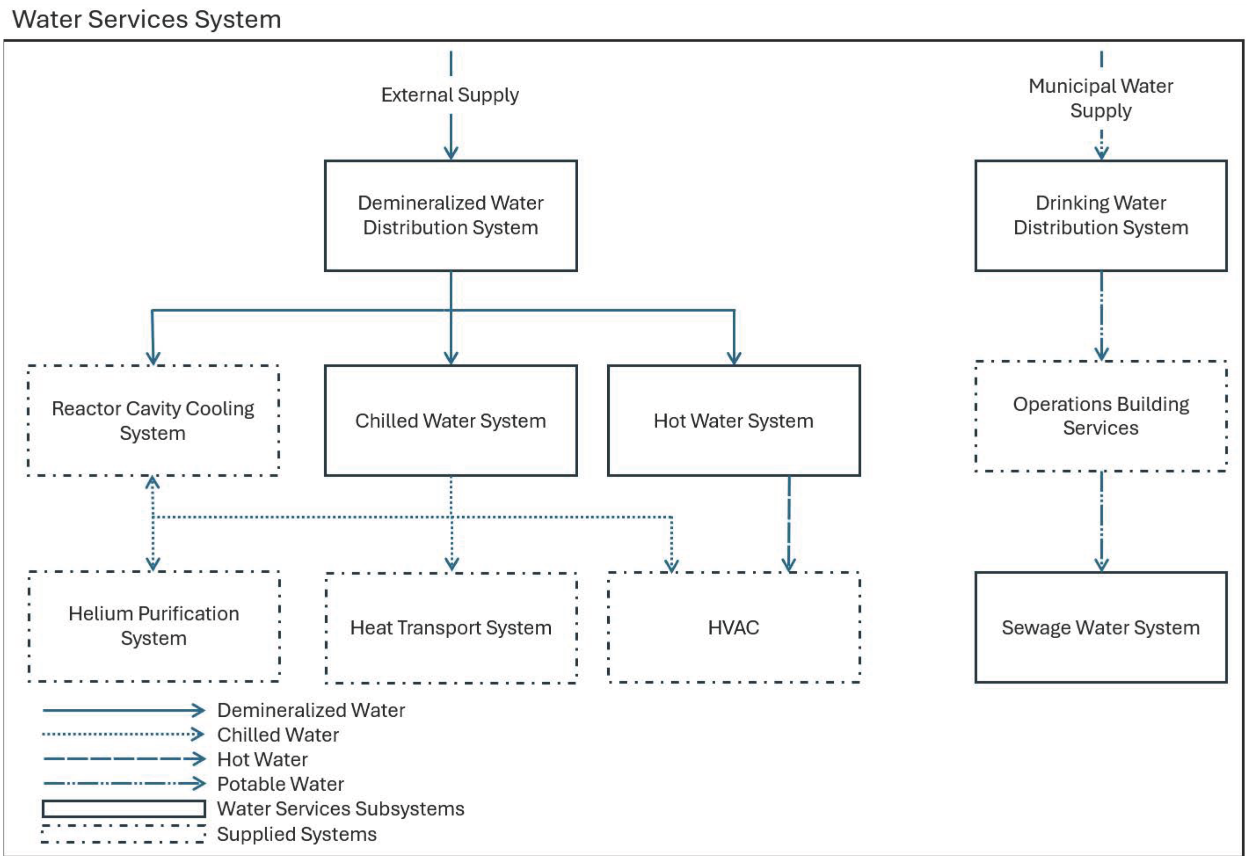
The WSS consists of the following subsystems:

- Demineralized Water Distribution System
- Chilled Water System
- Hot Water System
- Potable Water Distribution System

- Sewage Water System

The subsystems interface with each other and other SSCs across the reactor facility based on where each is required to distribute demineralized, hot, chilled, or potable water, as shown in Figure 9-7.

Figure 9-7 Interface diagram of Water Services Systems related systems distribution



9.7.1.3.1 Demineralized Water Distribution System

The demineralized water distribution system consists of pumps, piping, and valves necessary to distribute demineralized water to the SSCs across the facility that utilize demineralized water including the chilled water system, the hot water system, and the RCCS. It includes a reserve holding tank for retaining water prior to distribution.

9.7.1.3.2 Chilled Water System

The chilled water system consists of the necessary valves, piping, a surge tank, chiller units, and pumps to provide chilled water as necessary to remove heat from various SSCs. These SSCs include the helium circulator (Section 5.1), the Helium Purification System (Section 5.4), and the HVAC systems

(Section 9.1) and the RCCS (Section 6.3) during normal operation. During normal operation, the chilled water system acts as part of the heat removal pathway for these systems to the ultimate heat sink. During postulated events described in Chapter 13, the chilled water system is not credited for any heat removal.

The system includes 100% redundant pumps and chillers to ensure the chilled water system is available in the event of equipment failures or during maintenance periods. The chillers and pumps cool and circulate water through a multi-branched closed loop system, with each branch supplying chilled water to a single interfacing system.

9.7.1.3.3 Hot Water System

The hot water system provides heat in the form of heated water to the HVAC systems. It includes a closed water loop containing a heat pump, 100% redundant pumps, a surge tank, piping and valves.

9.7.1.3.4 Potable Water Distribution System

The potable water distribution system supplies water for human consumption to buildings for operator use. The system includes the necessary pipes and valves to distribute potable water from the municipal offsite connection to users in the operations building.

9.7.1.3.5 Sewage Water System

The sewage water system collects normal sanitary effluent from the facility and discharges it to the offsite sewer. The system consists of sanitary drainage components which collect and direct wastewater originating in the operations building.

9.7.1.4 Operational Overview

The water services systems distribute water to applicable SSCs in two separate piping circuits; one related to demineralized water uses associated with plant operations and one for potable water and sanitary effluent uses related to the operations building.

The demineralized water is supplied to and held in the system's holding tank as a source of makeup water. This water is distributed to the chilled water system, hot water system, and RCCS to refill the systems after maintenance periods or provide makeup for losses during normal operation. In addition, the holding tank can act as a redundant non-safety related supply of additional water for the RCCS.

The chilled water system and hot water system both circulate the demineralized water to relevant SSCs to remove or provide heat through dedicated heat exchangers.

The systems utilize the sensors and controls to monitor and maintain desired system operational states, including temperature, flow rate, and pressure.

9.7.1.5 System Evaluation

PDC-1: The water services systems are non-safety related and are not credited for mitigating the consequences of postulated accidents. The water services systems are fabricated and tested in accordance with generally recognized codes and standards. A quality assurance program is also established and implemented to ensure satisfactory performance of the systems' function. This is consistent with PDC-1.

PDC-2: Portions of the water services systems may be located in proximity to SSCs with safety related functions. Those safety related functions are protected from failure of the water services systems during a design basis earthquake by either seismically mounting the applicable components, physical separation, or barriers to preclude adverse interactions. These design considerations demonstrate conformance with PDC-2.

PDC-3: The water services systems are designed and located to minimize the probability and effect of fires and explosions through the use of low combustible materials, where possible, and physical separation. These design features, in conjunction with the fire protection system discussed in Section 9.3, provide assurance that the water services systems demonstrate conformance with the requirements of PDC-3.

PDC-4: Nearby safety related SSCs will be protected from the effects of discharging fluid and missiles by design. The water services systems are protected from potential missiles generated from the helium circulator by the design features of the helium circulator (missile shield) and vessel systems. There are no pressurized piping systems in or around the water services systems thus precluding the design from pipe-whip hazards. These design considerations demonstrate conformance with PDC-4.

PDC-44, PDC-45, PDC-46: The water services system interfaces with the safety related RCCS and the non-safety related helium circulator, helium purification system, and reactor facility HVAC system to transfer heat from the SSCs to the environment under normal operating conditions through the chilled water system. These systems are also designed to permit appropriate inspection and testing to ensure the integrity and the capacity of the system to cool the SSCs and to adequately transfer heat to the ultimate heat sink. These features demonstrate conformance with PDC-44, PDC-45 and PDC-46.

9.7.1.6 Testing and Inspection

The details of the inspection and testing programs will be described in the Operating License Application.

9.7.2 Lifting and Rigging Equipment

The lifting and rigging equipment encompass any equipment or associated permanent rigging points provided to lift and move equipment within the Citadel building. This section does not cover equipment associated with fuel handling operations, which is described in Section 9.2.

9.7.2.1 Functions, Classification and Design Criteria

9.7.2.1.1 System Safety Functions

Lifting and rigging equipment perform no safety functions.

9.7.2.1.2 System Non-Safety Functions

Lifting and rigging equipment are used to lift and move equipment needed for maintenance.

9.7.2.1.3 System Classifications

All lifting or rigging equipment is non-safety related.

9.7.2.2 Design Basis

The general PDC discussed in Section 3.6.1.4 apply to the Lifting and Rigging Equipment.

9.7.2.3 Design Description

There is no permanent lifting equipment inside the Citadel building. The upper surface of the Citadel building, located at ground level, is designed to accommodate the use of mobile cranes required during maintenance periods This surface is designed to support the loads of mobile cranes, other heavy loads, and equipment that may be required for maintenance activities.

Refueling equipment (Section 9.2) includes the lifting equipment required for transport of necessary materials and is not characterized as part of this section.

Instrumentation and control systems for mobile crane and rigging systems will be supplied as part of those pieces of equipment as required.

9.7.2.4 System Evaluation

PDC-2: Because no permanent crane systems are installed, crane systems are stored away from safety related SCCs when not in use. Adverse effects of natural phenomena are mitigated through the implementation of appropriate design features, layout considerations and storage arrangements. These considerations satisfy the requirements of PDC-2.

PDC-4: The mobile cranes and rigging equipment are selected based on required load rating for a lift. Permanent rigging points attached to building structure are designed in accordance with appropriate codes and standards. Considerations for equipment associated with refueling are considered as described in Section 9.2. This demonstrates conformance with PDC-4.

9.7.2.5 Testing and Inspection

Crane systems, rigging equipment and rigging points will be periodically inspected to applicable industry standards prior to use.

9.7.3 Chemical Dosing System

The chemical dosing system provides a means to maintain the chemistry of demineralized water to mitigate corrosion of applicable systems.

9.7.3.1 Functions, Classification and Design Criteria

9.7.3.1.1 Safety Functions

The chemical dosing system does not perform any safety functions.

9.7.3.1.2 Non-Safety Functions

The chemical dosing system provides chemicals to maintain water chemistry in the demineralized water used in the reactor facility to avoid excessive corrosion.

9.7.3.1.3 System Classifications

The chemical dosing system is non-safety related and is not credited for the mitigation of postulated events.

Non-safety related components which could fail as a result to a seismic event and impact the functionality of safety related SSCs are designed to Seismic Category I to prevent such an occurrence.

9.7.3.2 Design Basis

The general PDC discussed in Section 3.6.1.4 apply to the chemical dosing system.

9.7.3.3 Design Description

The chemical dosing system provides the means to maintain appropriate demineralized water chemistry in order to mitigate corrosion of applicable systems. The system is comprised of valves, piping, tanks and controls necessary to supply chemicals to SSCs when necessary.

9.7.3.4 Operational Overview

The chemical dosing system treats the demineralized water as necessary to maintain effective water chemistry for each applicable system. This is through the addition of chemicals needed to mitigate corrosion.

Postulated malfunctions of the chemical dosing system do not impact the performance of the safety functions of the reactor.

9.7.3.5 System Evaluation

PDC-1: The chemical dosing system is non-safety related and is not credited for mitigating the consequences of postulated accidents. The chemical dosing system is fabricated and tested in accordance with generally recognized codes and standards. A quality assurance program is also established and implemented to ensure satisfactory performance of the system’s function. This is consistent with PDC-1.

PDC-2: Any safety related SSCs located in proximity with the chemical dosing system are protected from failure of the chemical dosing system during a design basis earthquake by either seismically mounting the applicable chemical dosing system components, physical separation, or barriers to preclude adverse interactions. These design considerations for the chemical dosing system demonstrate conformance with PDC-2.

PDC-3: The chemical storage part the chemical dosing system is not in the proximity of safety related SSCs. Chemicals used in the system are non-flammable, or appropriate precautions are taken to protect chemical storage in accordance with the fire protection plan described in Section 9.3. These considerations provide assurance that the chemical dosing system demonstrates conformance with the requirements in PDC-3.

PDC-4: Nearby safety related SSCs will be protected from the effects of discharging fluid and missiles by design. There are no pressurized piping systems in the chemical dosing system thus precluding the design from pipe-whip hazards. These design considerations demonstrate conformance with PDC-4.

9.7.3.6 Testing and Inspection

The details of the inspection and testing programs, if required, will be described in the Operating License Application.

9.7.4 Other Auxiliary Systems

Several other auxiliary SSCs, all non-safety related, provide functions necessary to facilitate maintenance and operation the research reactor facility. None of those SSCs are credited with any safety function or with mitigating consequences of postulated events:

- *Facility Lighting, including emergency lighting*
- *Warehouses for storing equipment*
- *Storage of contaminated equipment*
- *Non-hazardous waste management services*
- *Firewater storage systems*
- *Heat tracing system*
- *Stormwater drains*
- *Grounding systems*
- *Cathodic protection systems*
- *Lightning protection systems*
- *HVAC systems for buildings other than the reactor facility*
- *Access roads*

These auxiliary SSCs are designed in accordance with local building code and relevant permits. The services are designed so that they do not interfere with safety related SSC's ability to perform their safety function. Portions of the auxiliary SSCs may be located in proximity to safety related SSCs. Those safety related SSCs are protected from failure of the auxiliary SSCs during a design basis earthquake by either seismically mounting the applicable auxiliary site services components, physical separation, or barriers to preclude adverse interactions. This satisfies the requirements of PDC-2 for the auxiliary site services.

Auxiliary SSCs that involve handling of radioactive material may include remote manipulation capabilities, as appropriate, to facilitate limiting personnel occupational exposures in accordance with ALARA practices.

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USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 10 - EXPERIMENTAL FACILITIES AND UTILIZATION

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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CHAPTER 10 EXPERIMENTAL FACILITIES AND UTILIZATION

This chapter describes and discusses the mission as well as the core and cross-cutting objectives of the University of Illinois Urbana-Champaign (U. of I.) research reactor, based on the KRONOS™ Micro Modular Reactor technology. This includes its use of experimental facilities, their intended application, and the experiment program.

10.1 SUMMARY DESCRIPTION

The mission of the U. of I. research reactor deployment is to de-risk advanced reactor deployment and enable a new paradigm of nuclear power through education, research, and at-scale demonstration of next generation technologies and applications. Realization of this mission is through the deployment of advanced nuclear technology in a setting representative of next generation nuclear energy applications. Historically, university leadership in research and development has paved the way for optimized deployment of new technologies. The U. of I. research reactor continues this rich tradition to improve operational and deployment characteristics of emerging nuclear technologies.

The first core objective of the facility is to help cultivate the future workforce and address public perception of nuclear power through *Education, Training and Engagement*. This includes educating the next generation of engineers and scientists, training operators in reactors operations, preparing technicians for installation and maintenance, and engaging/informing the general public.

The second core objective is to enable a new paradigm of nuclear energy—characterized by enhanced safety, efficiency, and sustainability—through *Research and Development*. This includes the optimization of reactor components, critical enabling technologies (like high temperature materials, advanced instrumentation, and cybersecurity) and synergistic applications (including integration with additional energy systems).

The cross-cutting objective is to demonstrate the future of nuclear power by *At-scale Demonstration*. Many vendors are proposing co-location of Small Modular Reactors and Microreactors with industrial applications which need both electricity and high temperature heat as a means of enhancing the economic proposition of their technology. This includes generation of electricity, district heat, integrated thermal energy storage, and potential hydrogen production. By showcasing large-scale applications—such as reliable electricity generation, district heating solutions, advanced thermal energy storage, and hydrogen production—the project aims to highlight nuclear power's versatility and future potential.

Some of the experiments are expected to include the performance of start-up physics testing, conduct and demonstration of maneuvering operations at various power levels to assess plant performance and capabilities, and measure standard reactor parameters such as individual and group control rod worth, temperature coefficients, excess reactivity, shutdown margin, and changes in power demand.

Experimental wells will be situated externally to the reactor pressure vessel where in-situ testing and monitoring can be performed via fission chambers, temperature detectors, pressure sensors, etc.

All experiments must undergo a formal and rigorous review and approval process to ensure they comply with the reactor license, including U. of I. research reactor Technical Specifications and Radiation Protection Program, and 10 CFR 50.59 regulations.

10.2 EXPERIMENTAL FACILITIES

Following the experiment definition as stated in ANSI/ANS 15.1-2007 (R2023), “The Development of Technical Specifications for Research Reactors” (Reference 10-1), an experiment at the U. of I. research reactor is defined as any operation, hardware, or target (excluding devices such as detectors, foils, etc.) that is designed to investigate nonroutine reactor characteristics or that is intended for an irradiation facility.

Presently, the only irradiation facilities planned for the research reactor are experimental wells situated externally to the reactor vessel (RV) for in-situ testing and monitoring of materials and/or instruments under irradiation conditions. Since these experimental wells are external to the RV and also do not penetrate the RV, they will not interfere with safe reactor shutdown or affect core reactivity or cooling. The exact number of wells, materials of construction, physical size and dimensions, and any required shielding or ventilation considerations will be provided with the Operating License Application, using the guidance stated in Section 10.2 of NUREG-1537.

As the U. of I. research reactor design matures, additional facilities may be incorporated to enhance experimental capabilities, and these will be finalized and described in detail in the Operating License Application.

10.3 EXPERIMENT REVIEW

The initial review of the experiment is conducted by U. of I. research reactor staff with technical expertise while assisting the experimenter in preparing the experiment request. The intent is to raise potential safety questions, analyze them during this review step, and identify appropriate mitigation measures to address any safety issues.

The experiment safety analysis includes, if applicable, analyses such as thermal, sample decomposition-pressure, experiment failure, loss of coolant to the experiment, failure of adjacent experiments, corrosion, and explosion. Specifically, ensuring that the experiment will meet the constraints imposed by “Experiments,” of the U. of I. research reactor Technical Specifications.

The Reactor Health Physics Manager reviews the experiment request to ensure that all necessary radiological control measures are performed during the experiment and that the experimenter possesses the experience and equipment to deal with the expected radiation levels. Radioactive waste generation and eventual disposal pathway are also stipulated. Their review follows the applicable requirements of the U. of I. research reactor Radiation Protection Program, which is written following the guidance of ANSI/ANS 15.11-2016 (R2021), “Radiation Protection at Research Reactor Facilities,” (Reference 10-2).

The Reactor Advisory Committee (RAC), as described in Chapter 12, then reviews the experiment request consistent with the guidance stated in Section 6.2.3 of ANSI/ANS 15.1-2007 (R2023), which includes determinations that proposed changes in equipment, systems, test, experiments, or procedures are allowed without prior NRC authorization per 10 CFR 50.59.

After all of the required experiment reviews have been completed, and all questions or concerns have been addressed and/or resolved, the Engineering Support and Operations Manager (Level 3 of the U. of I. research reactor organization structure – see Section 12.1.2.3) will approve the experiment.

Procedures associated with experiment review, which follow the guidance stated in Section 6.4 of ANSI/ANS 15.1-2007 (R2023), are discussed in Section 12.3 of this document, specifically the administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity.

Finally, controls over the design, fabrication, installation, and modification of experimental equipment to the extent that these impact safety-related items follow the U. of I. Quality Assurance Program (QAP) as stated in "Experimental Equipment," of "Quality Assurance Program Description for the University of Illinois Urbana-Champaign Research and Test Reactor" Topical Report (Reference 10-3).

10.4 REFERENCES

10-1 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1-2007 (R2023), "The Development of Technical Specifications for Research Reactors."

10-2 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.11-2016 (R2021), "Radiation Protection at Research Reactor Facilities."

10-3 U. of I. Topical Report IMRDD-MMR-22-03-A, Release 01, "Quality Assurance Program Description for the University of Illinois Urbana-Champaign Research and Test Reactor," dated August 14, 2023 (ADAMS Accession Number ML23167C137).

USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 11 - RADIATION PROTECTION AND WASTE MANAGEMENT

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
ALARA	As Low As is Reasonably Achievable
DOE	Department of Energy
DOT	Department of Transportation
FSAR	Final Safety Analysis Report
GTCC	Greater than Class C Radioactive Waste
HVAC	Heating, Ventilation and Air Conditioning
IHX	intermediate heat exchanger
LLRW	Low-Level Radioactive Waste
LEU+	Low-Enriched Uranium+
LWR	Light Water Reactor
NRC	Nuclear Regulatory Commission
NVALP	National Voluntary Laboratory Accreditation Program
PSAR	Preliminary Safety Analysis Report
RCCS	reactor cavity cooling system
RG	regulatory guides
RPP	Radiation Protection Program
RWP	Radiation Work Permit
TLD	Thermo Luminescent Dosimeters

CHAPTER 11 RADIATION PROTECTION AND WASTE MANAGEMENT

This chapter discusses and analyzes radiological consequences related to the normal operation of the reactor. Included are the principal discussions of the facility program to control radiation and expected radiation exposures resulting from the operation, maintenance, and use of the reactor. This chapter outlines the methods for quantitative assessment of radiation doses in the restricted and unrestricted areas; application of these methods to all applicable radiation sources related to the full range of operation; and the program and provisions for protecting the health and safety of all individuals present at the University of Illinois Urbana-Champaign (U. of I.) research reactor facility, the general public, and the environment.

11.1 RADIATION PROTECTION

The following sections provide information related to the Radiation Protection Program (RPP) for the U. of I. research reactor. These sections provide information for review by the U.S. Nuclear Regulatory Commission (NRC) staff as part of the Preliminary Safety Analysis Report (PSAR). The RPP as denoted in this section applies to all individuals working at or visiting the U. of I. research reactor, members of the general public, and includes the protection of the environment that could be affected by the operation of the reactor. U. of I. is responsible for implementation of the RPP and is committed to ensuring safe and efficient operation of the reactor facility such that it benefits the U. of I., the State of Illinois, and the public. This section addresses radiation protection as required in the applicable sections of 10 CFR 20, as well as 10 CFR 50.34. The reference materials in ANSI/ANS 15.11-2016 (R-2021), “Radiation Protection at research reactor Facilities” (Reference 11-1), and NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” were used as guidance in preparing this PSAR Chapter.

11.1.1 Radiation Sources

It is anticipated that any potential radioactive material generated in the U. of I. research reactor will likely come from the sources discussed in this section, which are summarized in Table 11-1. Inclusion in this list does not imply that a source has significant activity or health impacts. Table 11-1 lists the location of each radiation source, the primary radionuclides that generate radiation, the type of radiation emitted, and the degree to which the radionuclides are retained in the source location.

Table 11-1 Preliminary List of Radioactive Material Generation Sources

Location	Primary Radionuclide Source	Radiation Type	Retention of Radionuclides
Nuclear fuel	Heavy metals, fission products	Alpha, beta, gamma	Retained in TRISO and fuel pellet, with select mobile element diffusion
Startup neutron source	Cf-252 and its associated decay products, spontaneous fission products, and emitted neutrons	Neutrons, alpha, beta, gamma	Retained in sealed source capsule
B_4C bearing components: Burnable absorber, Control rod absorber, Core neutron shield	B-10 activation to H-3, C-13 activation to C-14	Beta	H-3 diffusion, C-14 retained in matrix
Core graphite	B-10 and Li-6 impurity activation to H-3, C-13 activation to C-14	Beta	H-3 diffusion, C-14 retained in matrix
Primary coolant	He-3 activation to H-3	Beta	Circulates in primary loop + helium clean-up system
Reactor internals	Co-59 activation to Co-60	Gamma	Retained in steel
Reactor vessel	Co-59 activation to Co-60	Gamma	Retained in steel
Structures and concrete in the Citadel Building	Various activation products, to be provided in the Operating License Application	Beta, gamma	Retained at source
Reactor cavity cooling system	H-2 activation to H-3, activation products of corrosion and impurities	Beta, gamma	Circulates in RCCS water
Secondary coolant (molten salt) in the intermediate heat exchanger	Na-23 activation to Na-24, activation products of corrosion and impurities	Beta, gamma	Circulates in secondary loop + salt clean-up system
Air in the reactor cavity	Ar-40 activation to Ar-41, N-14 and O-16 activation to C-14	Beta, gamma	Citadel building air controlled by HVAC pressure zones

A full source term and radionuclide transport assessment will be provided in the Operating License Application. A high-level description of the expected transport is provided in this section for the purpose of describing where radionuclides may accumulate during normal operations.

The fuel is based on the TRISO design which is discussed in detail in Section 4.2.1. Radionuclides generated by the fissioning of the U-235 content in the Low-Enriched Uranium+ (LEU+) fuel kernels would be similar to the fission product inventory found in a LEU-fueled Light Water Reactor (LWR).

During routine operations, potential diffusion or leakage of mobile radionuclides such as Kr-85m, Sr-90, Ag-110m, I-131, Cs-137 and Xe-135 from the fuel pellets may enter the primary helium coolant. Helium is circulated within the helium pressure boundary by the heat transport system to transfer heat to the secondary molten salt coolant through the intermediate heat exchanger (IHX). The helium pressure boundary limits release of the primary coolant and potential radionuclide content, although there is a potential for slow leakage from flange seals into Citadel building air volumes.

Activation of core components containing boron or lithium produces H-3 (tritium). These sources include the burnable poisons, control rod absorbers, core neutron shielding and trace amounts of impurities in the core graphite. H-3 generated in those components can potentially release into the primary coolant.

Anticipated or potential contaminants in the primary coolant are minimized by using high purity source helium, although the helium naturally contains a small amount of He-3. When activated by exposure to the reactor neutron flux, tritium is produced directly in the coolant.

During operation, radionuclides present in the primary coolant may accumulate in the helium purification system filters and be distributed throughout the primary loop via plate-out deposition and/or adsorption in materials such as graphite. This results in an equilibrium circulating activity in the primary loop. Discussion of primary coolant inventory control and sampling is provided in Section 5.2, Section 5.4 and Section 5.5.

The secondary molten salt coolant in the IHX is exposed to a low neutron flux, resulting in potential neutron activation directly in the coolant. Neutron activation products, inclusive of potential impurities and corrosion products, will primarily consist of the following isotopes: Na-24, K-40 and Mn-56, although some K-40 content is naturally occurring. Secondary coolant system monitoring and/or sampling are discussed in Section 5.3.4 and Section 5.3.5. Potential for secondary coolant contamination by primary coolant is discussed in Section 5.3.6.

Activated reactor components and structures in the Citadel building, including neutron flux detectors, are not a radiation safety concern during normal operations and maintenance. However, their isotopic composition and activities will be assessed if unplanned maintenance activities of the affected reactor components or systems with the affected reactor components or systems, and prior to decommissioning of the reactor facility. Further details will be provided in the Operating License Application.

Activation of the water in the reactor cavity cooling system (RCCS), inclusive of impurities and corrosion products, produces relatively small amounts of tritium and other radionuclides compared to other sources. Potential leakage from the RCCS in the Citadel building is collected in a sump and sent to the liquid radioactive waste system.

Generation of Ar-41 and C-14 may occur in areas where ambient air is activated by leakage of neutrons from the reactor core. Air in the Citadel building during normal operations, inclusive of potentially leaked primary coolant, is controlled by the nuclear plant heating, ventilation and air conditioning (HVAC) system and monitored as discussed in Chapter 9. Some radioactive condensate may be formed in the HVAC system and is sent to the liquid radioactive waste system.

More details on radionuclides, activities and distribution of airborne, liquid and solid sources will be provided in the Operating License Application.

11.1.2 Radiation Protection Program

The RPP as required in 10 CFR 20.1101 is designed such that radiological protection of the public and staff is the highest priority and ensures the greatest level of awareness by the U. of I. administration and reactor management. The reactor staff is expected to include a Reactor Health Physics Manger and some combination of professional health physicists and health physics technicians. These positions may be dedicated entirely to the U. of I. research reactor or may be shared with knowledgeable staff members on the radiation safety staff of the U. of I. Division of Research Safety at the discretion of U. of I. management. The Reactor Health Physics Manger will have direct access to senior reactor staff and upper university administration, should the need arise, to report and address serious radiation safety concerns.

Radiation detection and monitoring instrumentation allows for monitoring and assessment of any effluent radioactivity releases to the environment or contamination that may occur at the reactor facility. The radiation monitoring equipment is chosen based on its ability to detect any of the nuclear radiation that emanates from the fission reaction itself or any of the neutron activation products produced by the operation of the reactor.

The RPP consists of the following elements:

1. Initial training and periodic retraining for personnel frequenting or working in Radiation Controlled Areas;
2. Radiation safety training and the effects of ionizing radiation. Current regulatory dose limits;
3. Meaning of posting and labeling requirements and their proper use;
4. Access and egress controls and escort procedures;
5. Periodic Survey requirements and Assessments of Radiologically Controlled Areas;
6. Personnel dose monitoring [NVLAP (National Voluntary Laboratory Accreditation Program)] and periodic reviews and assessments of the monitoring results;
7. Effluent and personnel monitoring and equipment;
8. Use, storage, and transfer of radioactive materials;
9. Emergency response training;
10. Radiation worker rights and responsibilities;
11. Reactor maintenance radiological work oversight;
12. As Low As is Reasonably Achievable (ALARA) reviews and assessments
13. Radiation Work Permit Program, if required; and
14. Periodic Audits that assess the effectiveness of the RPP.

The RPP utilizes the applicable guidance of ANSI/ANS 15.11-2016 (R-2021) and NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation" (Reference 11-2), for specific areas of radiation safety.

11.1.3 ALARA Program

The U. of I. research reactor ALARA program is designed in accordance with ANSI/ANS 15.11-2016 (R-2021) and RG 8.10, Rev 2, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable" (Reference 11-3). Dose limit guidelines (goals) will be developed internally within the reactor facility to keep doses less than the federal regulatory limits as described in 10 CFR 20, Subpart C. Investigational levels may also be set which if exceeded would trigger an investigation by the Health Physics staff into the causes of the above normal doses such that staff can identify and recommend potential solutions to keep facility doses ALARA.

Examples of ALARA objectives include, but are not limited to, the following and may be accomplished with Health Physics and operational staff input and guidance:

- Informing the reactor staff of ALARA principles, goals, and objectives;
- Ensuring that staff are responsible for implementation of the ALARA program as they are the individuals qualified and most familiar with their individual processes and procedures;
- Performing radiation surveying and monitoring throughout the facility and making the results available to staff so that they can utilize the information in order to keep their doses ALARA;
- Minimization of radioactive waste generation in order to keep downstream waste handlers' doses ALARA;
- Providing calibrated radiation survey instruments to staff for their use in assessing their individual radiation work environments;
- Reviewing radioactive materials isotopic concentrations in liquid and gaseous effluents so that the appropriate individuals or groups can take steps in the minimization of the effluent sources;
- Ensuring that staff members perform periodic procedural reviews to maintain exposures ALARA; and
- Ensuring that all radioactive materials receipt, transfer, and disposal is performed with ALARA principles in mind.

Dose reduction may also be obtained by using planning activities of all parties involved—such as operations, maintenance, and health physics—for a special maintenance or repair activities that could involve individuals receiving greater than normal or expected doses.

The ALARA Program also sets guidelines for the release of gaseous and liquid effluents from the reactor facility. These guidelines will be set after the appropriate engineering controls are developed that minimize releases, where possible, prior to any additional human intervention. All guidelines will be set under the limits prescribed in 10 CFR Part 20, Appendix B or in the U. of I. research reactor Technical Specifications.

The ALARA Program will be approved by reactor management prior to its implementation and administered by the Health Physics staff.

11.1.4 Radiation Monitoring and Surveying

The radiation monitoring and survey programs will consider the guidance provided in the following regulatory guides (RG):

- RG 8.2, "Administrative Practices in Radiation Surveys and Monitoring," Revision 1;
- RG 8.4, "Personnel Monitoring Device-Direct-Reading Pocket Dosimeters," Revision 1;
- RG 8.7, "Instructions for Recording and Reporting Occupational Radiation Exposure Data," Revision 4;
- RG 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," Revision 1;
- RG 8.25, "Air Sampling in the Workplace," Revision 1; and
- RG 8.34, "Monitoring Criteria and Methods to Calculate Occupational Radiation Doses," Draft Revision 1.

Radiation monitoring and surveying at the U. of I. research reactor meets the requirements of and are conducted in accordance with 10 CFR 20.1501 and 20.1502. A facility effluent monitoring system will be utilized to assess any gaseous or particulate effluent leaving the reactor facility. The system will have the capability to monitor noble gases, radioiodines and any activation or fission products identified in this PSAR and future Final Safety Analysis Report (FSAR). The release limits will be in accordance with those specified in the U. of I. research reactor Technical Specifications, to be provided in the Operating License Application.

Monitoring and/or sampling is employed to ensure activation products within the helium gas primary coolant loop and the secondary system molten salt coolant loop are within expected parameters.

Radiation monitoring and surveying will be conducted routinely in designated areas of the facility to ensure that unwanted contamination or dose rates do not exist that would affect the exposure of general facility workers. Additionally, monitoring and surveying will be conducted prior to and during any activity involving the handling of radioactive materials or when individuals need to frequent or work in an area which is designated as "Radiation Area" or "High-Radiation Area." Radiation detection equipment will be available to staff members when working in these areas so they can independently assess their radiation environment and make the appropriate decisions related to their job tasks, consistent with the procedures established by the RPP.

Personnel monitoring will be performed by using either portal type or handheld radiation survey instruments or some combination thereof. Staff will be trained in their proper use, and the Health Physics Group will determine where they will be used most effectively.

Radiation monitoring equipment will consist of instrumentation capable of measuring any of the radiation expected to occur at the U. of I. research reactor including dose rate fields, particulate matter or gaseous or liquid emissions generated by the operation of the reactor or from the use of radioactive sources necessary for the operation of the reactor or Health Physics program.

11.1.5 Radiation Exposure Control and Dosimetry

Radiation exposure to members of the general public and reactor staff will be controlled to the greatest extent possible by utilizing engineered controls to minimize human interactions with activated or contaminated structures, systems, and components. These engineered systems and controls will provide the necessary means to supply appropriate radiation exposure reductions. When engineering controls are not possible or are not feasible, trained Health Physics staff members will be available to help evaluate potential exposures and to assist with the minimization of doses to the public, environment, and reactor staff.

Radiation dosimetry accredited by NVLAP will be provided to staff radiation workers as required by the U. of I. research reactor RPP and dosimetry and will be exchanged periodically (e.g., monthly, quarterly). The monitoring results will be disseminated to team leaders and all staff members. Information gained by personnel dosimetry results will be used by Health Physics and Reactor Operations staff to help keep facility exposures ALARA and set ALARA goals and targets.

Environmental radiation dosimetry will also be used for assessing public doses via the reactor environmental monitoring program. Dosimeters will be placed at varying distances from the reactor in selected compass quadrants and then analyzed quarterly to help assess any offsite radiological consequences resulting from the operation of the reactor. Environmental monitoring results will be used to assess public dose to members of the public in proximity to the reactor facility.

11.1.5.1 Controlled Area

Access to controlled areas (areas where access is controlled with respect to radiation safety) will be controlled by using administrative procedures that allow area access to only specifically trained staff members of the reactor facility including, but not limited to, operations, engineering, health physics and maintenance staff members. Should any facility high radiation area exist, access to the high radiation area will be controlled as required by the requirements of 10 CFR 20.1601. The need for access controls, including additional training, will be evaluated and implemented as necessary to provide the appropriate level of radiation protection for individuals accessing those areas.

Areas possibly having dose rates that meet the monitoring requirements of 10 CFR 20.1502 and thus contributing to the overall staff radiation dose include the reactor mechanical equipment room, areas within the Citadel building with proximity to the reactor vessel and primary helium cooling equipment. These areas will be monitored with area radiation monitors or dosimetry to assess the general radiation field environment so that additional engineered controls can be implemented to keep personnel doses ALARA.

11.1.6 Contamination Control

Potential contamination within the reactor building is expected to be limited to the Citadel building and areas where helium impurity activation products can escape the pressure vessels system. Another potential source of radioactive trace contamination is the secondary system molten salt loop resulting from the activation of the salt in the IHX and the resulting contamination within the liquid salt circulation system.

Personnel will be required to use radiation monitoring equipment upon entering and exiting potentially contaminated areas to ensure that contamination is not spread to other areas of the facility. The Health Physics staff will provide equipment and will ensure its operability and calibration status.

Contamination inside the reactor facility will be controlled to the greatest extent possible by utilizing engineered systems and controls to minimize human interactions within potential contaminated areas. These systems will provide radiological control features necessary to provide the appropriate radiation exposure (both external and internal) reductions. Additionally, detailed work procedures will be utilized that consider experience and lessons learned from previous maintenance activities that may have encountered contamination issues. A Radiation Work Permit (RWP) system will be available and utilized to establish acceptable dose and/or contamination levels when work is unique or when procedural deviations are required to perform the specified work. When engineering controls are not possible or are not feasible, qualified Health Physics staff members will be available to help evaluate contamination and exposure rates and to assist with the minimization of doses to the public and reactor staff.

11.1.7 Environmental Monitoring

Environmental radiation monitoring on and around the proposed reactor site will be conducted prior to the reactor's operation to assess ambient radiation levels on the University of Illinois campus area where the reactor will be sited. Additionally, water, soil and vegetation samples may be collected for background determination via radionuclide analysis in the areas around the reactor site, if deemed necessary.

Once reactor operations have commenced, environmental monitoring will include Thermo Luminescent Dosimeters (TLD) placed at strategic locations around the reactor complex to assess any increased dose rates that originate from the reactor. Additionally, soil, water and vegetation samples may be collected and analyzed to ensure that the impacts to the environment are minimal resulting from reactor operations.

11.2 RADIOACTIVE WASTE MANAGEMENT

11.2.1 Radioactive Waste Management Program

A Radioactive Waste Management Program will be developed for the U. of I. research reactor facility. It will contain procedures used to assess, collect, process and ship Low-Level Radioactive Waste (LLRW) to a licensed LLRW disposal site directly or through a radioactive waste processing broker. Also, any liquid or gaseous effluent materials produced by the reactor operation will be analyzed and quantified to ensure that the appropriate release limits of 10 CFR 20, Appendix B or the U. of I. research reactor Technical Specifications are not exceeded. If Part 20 release concentration limits of liquid effluent are expected to be exceeded via analysis, the liquid waste can be subject to a waste solidification process so that it can qualify for or Class A radioactive waste. Transportation of the waste is anticipated to be as a Low-Specific Activity waste and it will be buried as a Class A radioactive waste.

11.2.2 Radioactive Waste Control

Procedures and controls will be developed to minimize the generation of LLRW. Any LLRW generated is expected to be Class A. Procedures will be developed to identify and segregate the two waste streams to facilitate the processing and eventual shipment of the waste to the appropriate waste disposal site or for collection and processing by a licensed LLRW broker.

Radioactive waste storage areas will be controlled such that access is limited to only authorized individuals. Radiation monitoring of the area will be performed periodically to allow individuals frequenting the waste storage area to assess radiological conditions in keeping with ALARA principles.

Procedures will also be developed to collect and assess the concentrations and total quantities of radioactive materials that leave the facility either through liquid or gaseous pathways. A liquid effluent collections system may be used to collect liquid effluent prior to its monitoring and release. If necessary, liquid waste can be solidified via an acceptable solidification agent for treatment as Type A Radioactive waste. Additionally, a facility effluent monitoring system will be used to analyze and quantify any materials leaving the facility through the facility exhaust system.

11.2.3 Release of Radioactive Waste

Radioactive solid waste will only be released for packaging after it has been assessed for waste classification in accordance with 10 CFR 61. It will then be packaged in accordance with U.S. Department of Transportation (DOT) regulations and the accepting waste broker or waste site acceptance criteria for the specific waste stream being handled. Additionally, some contaminated radioactive waste containing short-lived isotopes may be allowed to decay to background levels to allow for disposal as normal refuse after surveying to ensure sufficient decay has occurred to meet free-release criterion. Procedures will be developed that allow for the assessment and categorization of such waste for eventual unrestricted release into the environment. Radioactive material released as an effluent will be assessed or monitored prior to release into the environment. Releases will be in accordance with 10 CFR 20.2003 and 10 CFR 20, Appendix B or the U. of I. research reactor Technical Specifications.

The potential to generate Greater than Class C Radioactive Waste (GTCC) is currently not known. Potentially reactor core components and associated reactor hardware which experiences the greatest neutron fluences may need to be classified as Class C Waste. Should this be the case, components with this waste classification will be disposed of in accordance with the LLRW waste site criteria in effect during license termination and decommissioning activities. Once more information becomes available, this information will be included in the Operating License Application.

The precedent for university research reactors is that the U.S. Department of Energy (DOE) maintains ownership of the nuclear fuel throughout the full lifecycle and takes responsibility for its disposal. The U. of I. is currently pursuing this approach and is actively engaged with the DOE University Fuel Services program. Therefore, spent fuel will be handled and shipped utilizing U.S. DOE requirements as they are the ultimate owner and disposal pathway for the spent fuel.

11.3 RESPIRATORY PROTECTION PROGRAM

At this time, it is not anticipated that a Respiratory Protection Program will be needed for the U. of I. research reactor. If a Respiratory Protection Program is found necessary, such a program will be developed and implemented in accordance with the guidance in NRC Regulatory Guide 8.15, “Acceptable Programs for Respiratory Protection” (Reference 11-4) to meet the appropriate requirements of 10 CFR 20.1701-1705.

11.4 REFERENCES

11-1 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.11-2016 (R2021), "Radiation Protection at Research Reactor Facilities."

11-2 NUREG-1736, "Consolidated Guidance: 10 CFR Part 20 - Standards for Protection Against Radiation," October 2001.

11-3 Regulatory Guide 8.10, Revision 2, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," August 2016.

11-4 Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," October 1976.

USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 12 - CONDUCT OF OPERATIONS

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

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March 31, 2026

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Term	Description
ALARA	as low as (is) reasonably achievable
ALIs	Annual Limits on Intake
DACs	Derived Air Concentrations
EALs	Emergency Action Levels
EC	Emergency Coordinator
ED	Emergency Director
EP	Emergency Plan
EPZ	Emergency Planning Zone
FSAR	Final Safety Analysis Report
NEPA	National Environmental Protection Act
NPRE	Nuclear Plasma and Radiological Engineering
NRC	Nuclear Regulatory Commission
PSAR	Preliminary Safety Analysis Report
QAPD	Quality Assurance Program Description
RAC	Reactor Advisory Committee
RPP	Radiation Protection Program
SGI	safeguards information

CHAPTER 12 CONDUCT OF OPERATIONS

This chapter describes the conduct of operations for the University of Illinois Urbana-Champaign (U. of I.) research reactor facility. It includes the administrative aspects of the facility, emergency planning, physical security, quality assurance, reactor operator requalification, and the startup plan. Administrative aspects of facility operations are organization, review and audit functions, organizational aspects of radiation safety, facility procedures, required actions in case of facility Operating License or Technical Specification violations, or certain events or observations, reporting requirements, and recordkeeping.

12.1 ORGANIZATION

This section describes the organizational structure, functional responsibilities, levels of authority, and interfaces for establishing, executing, and verifying the organizational structure concerning facility operation. The organizational structure includes internal and external functions including interface responsibilities for multiple organizations. The organizational aspects of the Radiation Protection Program (RPP), the facility safety program, staffing, and selection and training of personnel are also discussed in this section.

12.1.1 Structure

The organizational structure for facility operations is shown in Figure 12-1.

12.1.2 Responsibility

The University of Illinois Board of Trustees is the entity with legal responsibility for holding the Construction Permit and the facility Operating License.

All functional positions in the organizational structure have responsibility for the development, implementation and compliance of policies and safe operation of the reactor facility, adherence to all requirements of the Operating License and Technical Specifications, and the protection of the health and safety of the general public, facility staff, and the environment. Additionally, all research reactor staff have stop-work authority with respect to safety, which will be detailed in facility policies and procedures.

Responsibilities for the key functional positions in the organizational structure are described in the following subsections, consistent with Figure 12-1. Responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

12.1.2.1 Head of the Department of Nuclear, Plasma, and Radiological Engineering (Level 1)

The Head of the Department of Nuclear, Plasma, and Radiological Engineering (NPRE) is responsible for directing the research mission, the quality and effectiveness of all programs and dedicating university resources necessary to ensure that all research, education and service are conducted in accordance with applicable federal, state and local regulations and accreditation requirements as well as providing the necessary university resources for safe operation of the facility.

12.1.2.2 Reactor Director (Level 2)

The Reactor Director reports to the Head of the Department of NPRE and is responsible for establishing the policies that minimize radiation exposure to the public and to radiation workers, providing direction for engineering and technical support, and ensuring that the requirements of the Operating License and Technical Specifications are met. The Reactor Director is advised by the Reactor Advisory Committee of the U. of I. research reactor.

12.1.2.3 Engineering Support and Operations Manager (Level 3)

The Engineering Support and Operations Manager reports to the Reactor Director and is responsible for day-to-day engineering support activities, design engineering, engineering configuration management, engineering administration, modifications and their implementation, facility design configuration control, document control/records management, facility engineering, system engineering, system testing, and technical support. The Engineering Support and Operations Manager is the design authority for the facility and is also responsible for ensuring the safety analysis is current and amend it as needed.

During operations, the Engineering Support and Operations Manager is responsible for facility activities, including operations, maintenance, and startup/preoperational testing.

12.1.2.4 Reactor Health Physics Manager (Level 3)

The Reactor Health Physics Manager reports to the Reactor Director and is responsible for the implementation of the RPP and the “as low as is reasonably achievable” (ALARA) program, monitoring worker doses, and calibration of health physics instrumentation. They also have responsibilities pertaining to the radiological aspects of potential emergencies, as described in Appendix 12A of this chapter. The Reactor Health Physics Manager maintains a line of communication with the Head of the Department of NPRE to facilitate the escalation of topics requiring executive level disposition. The Reactor Health Physics Manager has the responsibility and authority to interdict or terminate licensed activities that it believes are unsafe. The Reactor Health Physics Manager maintains a line of communication with the University Division of Research Safety to request additional radiological safety resources at the research reactor facility, if needed.

The University Division of Research Safety provides oversight in campus laboratory, biological, chemical, laser, radiological, and electrical safety, handling incident response (spills, injuries, near misses), training researchers, ensuring compliance with regulations, and managing hazardous materials to prevent accidents and maintain a secure research environment.

12.1.2.5 Reactor Operations Staff (Level 4)

The Reactor Operations Staff, consisting of U.S. Nuclear Regulatory Commission (NRC)-licensed reactor and senior reactor operators, reactor operations trainees, and other non-licensed personnel. They report to the Engineering Support and Operations Manager and are responsible for the safe and efficient operation of the U. of I. research reactor by conforming to applicable rules, regulations, Operating License and Technical Specification requirements, and operating procedures. Reactor and senior reactor operators are responsible for maintaining licensed qualification status.

12.1.2.6 Reactor Health Physics Staff (Level 4)

The Reactor Health Physics Staff report to the Reactor Health Physics Manager and are responsible for overseeing/managing the radiological aspects of research, as well as the training, and monitoring programs in order to protect personnel from radiation hazards and to assure compliance with federal, state, and U. of I. regulations.

12.1.3 Staffing

Sufficient resources will be provided in personnel and materials to safely conduct facility operations. Minimum staffing levels when the reactor is not secured, required contact lists, and events requiring the presence of a senior reactor operator at the facility will meet the requirements of the regulations and are consistent with Section 6.1.3 of ANSI/ANS 15.1-2007 (R2023), "The Development of Technical Specifications for Research Reactors" (Reference 12-1). Details will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(i).

12.1.4 Selection and Training of Personnel

A training program will be implemented and maintained for personnel performing or managing activities to support facility operation. ANSI/ANS 15.4-2016 (R2021), "Selection and Training of Personnel for Research Reactors" (Reference 12-2), will be used in the selection and training of personnel, as applicable, and will be described in the Operating License Application. Records of personnel training and qualification will be maintained.

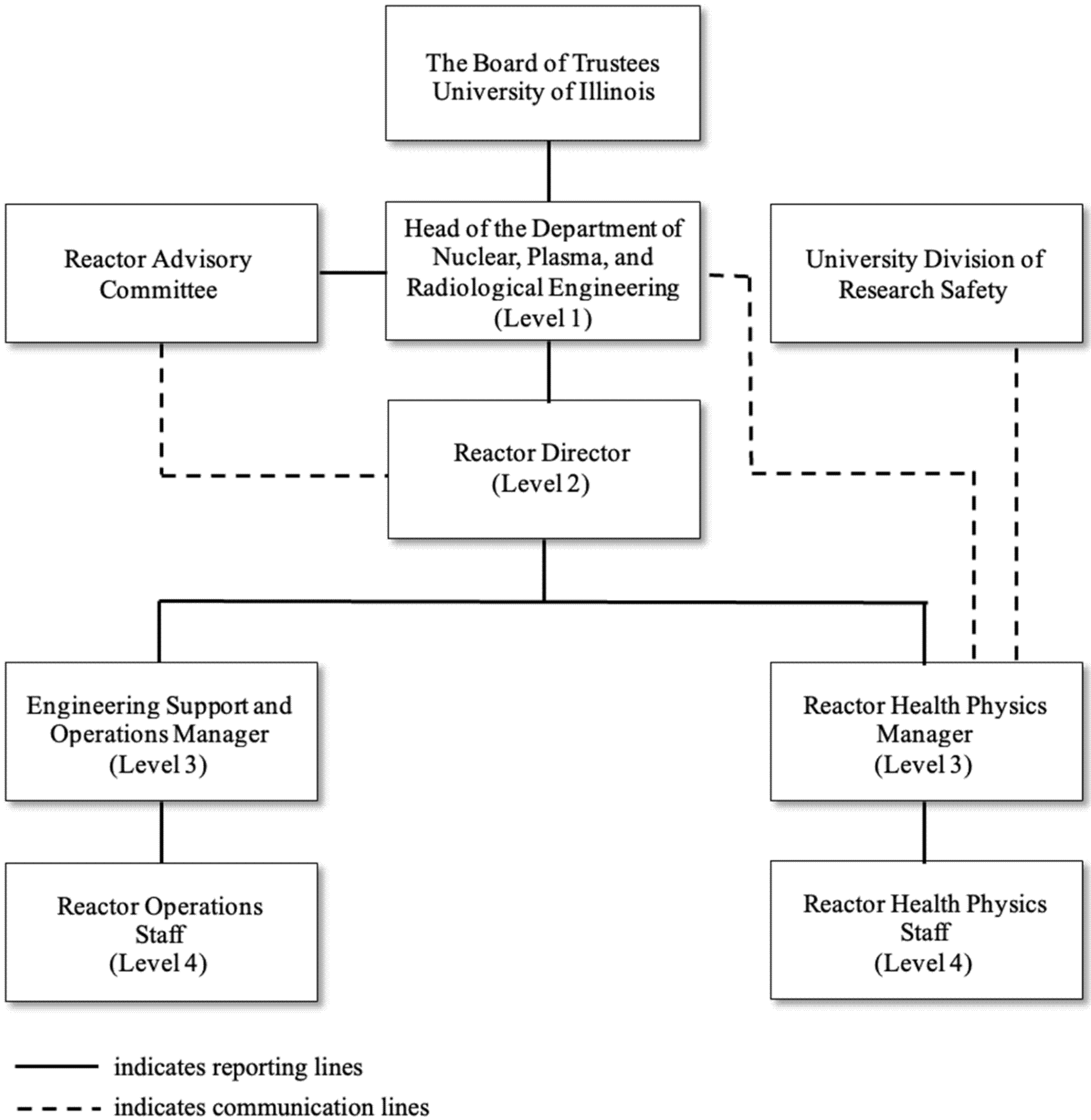
A description of the training program and the required minimum qualifications for facility staff will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(i).

The licensed operator training program, including the operator requalification training program, is addressed in Section 12.10.

12.1.5 Radiation Safety

Sufficient resources in terms of staffing and equipment are provided by the University Division of Research Safety to implement an effective RPP, consistent with guidance provided in ANSI/ANS 15.11-2016 (R2021), "Radiation Protection at Research Reactor Facilities" (Reference 12-3). Further details related to the authority of the RPP staff with respect to facility operations will be provided with the Operating License application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(i). The RPP is described in Section 11.1.2.

Figure 12-1 University of Illinois Urbana-Champaign Research Reactor Organizational Structure



12.2 REVIEW AND AUDIT FUNCTIONS

The U. of I. research reactor review and audit committee, referred to as the Reactor Advisory Committee (RAC), reports to the Head of the Department of NPRE and maintains an indirect reporting relationship with the Reactor Director and is responsible for independent oversight of certain activities to ensure the safe operation of the facility. The Head of the Department of NPRE ensures that the appropriate technical expertise is available for review and audit activities.

Committee activities are summarized and reported to the Head of the Department of NPRE and to the Reactor Director. The details of review and audit activities, who holds the approval authority, how it communicates and interacts with facility and University management will be described in the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(ii).

12.2.1 Composition and Qualifications

The composition and qualifications of the U. of I. research reactor RAC will follow the guidance stated in Section 6.2.1 of ANSI/ANS 15.1-2007 (R2023), and details will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(ii).

12.2.2 Charter and Rules

The charter and rules of the U. of I. research reactor RAC will include provisions for meeting frequency, quorums, use of subgroups, and minutes, will follow the guidance stated in Section 6.2.2 of ANSI/ANS 15.1-2007 (R2023), and details will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(ii).

12.2.3 Review Functions

Items that must be reviewed by the U. of I. research reactor RAC and the rules for distribution of reports or minutes will be consistent with the guidance stated in Section 6.2.3 of ANSI/ANS 15.1-2007 (R2023), and details will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(ii).

12.2.4 Audit Function

Items that must be audited by the RAC and rules for distribution of audit findings will follow the guidance stated in Section 6.2.4 of ANSI/ANS 15.1-2007 (R2023), and details will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(ii).

12.3 PROCEDURES

Operating procedures provide appropriate directions to ensure that the facility is operated safely and within the design basis and Technical Specification limits. Activities affecting safety are performed in accordance with approved implementation procedures. The level of detail in a procedure is dependent on the complexity of the task and considers the experience, education, and training of the users and the consequences of errors. Expectations for the use of procedures are documented and communicated to facility personnel.

The Technical Specifications require procedures for the following topics consistent with Section 6.4 of ANSI/ANS 15.1-2007 (R2023):

- Startup, operation, and shutdown of the reactor;
- Fuel loading, unloading and movement within the reactor;
- Maintenance of major components of systems that may have an effect on nuclear safety;
- Surveillance checks, calibrations, and inspections required by the technical specifications;

- Personnel radiation protection, consistent with applicable regulations or guidelines; procedures shall include management commitment and programs to maintain exposures and releases ALARA in accordance with the guidelines of ANSI/ANS 15.11-2016 (R2021);
- Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity;
- Implementation of required plans such as emergency or security plans; and
- Use, receipt, and transfer of by-product material.

A description of the facility procedures, including the review, approval, and changes processes, will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(vi) and 10 CFR 50.59.

12.4 REQUIRED ACTIONS

The U. of I. research reactor Technical Specifications will specify actions to be taken when certain events occur. These are actions to be taken in case of a safety limit violation, release of radioactivity from the site above allowed limits, and other events listed in ANSI/ANS 15.1-2007 (R2023). Details of required actions will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(vi).

12.5 REPORTS

The U. of I. research reactor Technical Specifications will specify the reporting requirements for the facility. These include annual routine operating reports, special reports for violations of safety limits, release of radioactivity from the site above allowed limits, and other events listed in Section 6.7 of ANSI/ANS-15.1-2007 (R2023). In addition, to ANSI/ANS-15.1-2007 (R2023), reports are made concerning permanent changes in the organization involving Level 1 or 2 personnel and significant changes in the transient or accident analysis as described in the Final Safety Analysis Report (FSAR).

Technical Specifications are described in Chapter 14 and will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(vi).

12.6 RECORDS

The U. of I. research reactor Technical Specifications will specify records required to be maintained, where and how they are maintained, and their length of retention, consistent with the guidance stated in Section 6.8 of ANSI/ANS-15.1-2007 (R2023). Records are grouped by retention period: 1) a period of at least five years (or for the life of the component involved if less than five years), and 2) the lifetime of the reactor facility. There are also record retention requirements for training records of NRC-licensed reactor and senior reactor operators. The Technical Specifications are described in Chapter 14 and will be provided with the Operating License Application, consistent with the requirements of 10 CFR 50.34(b)(6)(vi).

12.7 EMERGENCY PLANNING

In accordance with 10 CFR 50.34(a)(10), the specific information required in a Preliminary Safety Analysis Report (PSAR) by Appendix E.II of 10 CFR 50 is provided in Appendix 12A of this chapter. The Emergency Plan (EP) will be updated with the Operating License Application, consistent with the requirements applicable to a research reactor stated in 10 CFR 50.34(b)(6)(v). The EP will use the guidance provided in ANSI/ANS 15.16-2015 (2020), “Emergency Planning for Research Reactors” (Reference 12-4); Regulatory Guide 2.6, Revision 2, “Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities” (Reference 12-5); and NUREG-0849, “Standard Review Plan for Review and Evaluation of Emergency Plans for Research and Test Reactors” (Reference 12-6) to meet the applicable requirements of 10 CFR 50.54(q) for a research reactor.

12.8 SECURITY PLANNING

A description of the physical security plan for the facility will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(c) and will consider the guidance provided in Regulatory Guide 5.59, Revision 1, “Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance” (Reference 12-7). Security under the Operating License will meet the requirements of 10 CFR 50.54(p) as applicable to a research reactor facility with Category III material.

Development of a security plan requires an approved safeguards information (SGI) protection plan. The U. of I. Safeguards Information-Modified Handling Protection Plan was submitted and approved by the NRC on March 12, 2014 (Document No. IMRDD-MMR-23-01) (Reference 12-8).

Additionally, a Material Control and Accounting Program, consistent with the requirements stated in 10 CFR 74, will be established for the facility for the control and accounting of special nuclear material.

12.9 QUALITY ASSURANCE

The Quality Assurance Program Description (QAPD) for the design, construction, and operation of the U. of I. research reactor is based on ANSI/ANS 15.8-1995 (R2018), “Quality Assurance Program Requirements for Research Reactors” (Reference 12-9) and considers the guidance provided by Regulatory Guide 2.5, Revision 1, “Quality Assurance Program Requirements for Research and Test Reactors” (Reference 12-10).

The U. of I. research reactor QAPD was submitted as a Topical Report and approved and accepted by the NRC (Document No. IMRDD-MMR-22-03-A) (Reference 12-11).

12.10 OPERATOR TRAINING AND REQUALIFICATION

The NRC-licensed reactor operator and senior reactor operator training and requalification plan will be developed and implemented in accordance with 10 CFR 55 as it pertains to non-power facilities. The operating training and requalification plan will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(8) and 10 CFR 50.54 and will periodically be audited as discussed above in Section 12.2.4.

12.11 STARTUP PLAN

The startup plan will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(iii).

12.12 ENVIRONMENTAL REPORTS

An environmental assessment to ensure compliance with National Environmental Protection Act (NEPA) requirements for the construction and operation of the U. of I. research reactor will be performed. As part of the Construction Permit and Operating License Applications, the Environmental Report will be submitted separately to ensure NEPA compliance.

12.13 REFERENCES

12-1 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1-2007 (R2023), "The Development of Technical Specifications for Research Reactors."

12-2 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.4-2016 (R2021), "Selection and Training of Personnel for Research Reactors."

12-3 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.11-2016 (R2021), "Radiation Protection at Research Reactor Facilities."

12-4 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.16-2015 (R2024), "Emergency Planning for Research Reactors."

12-5 Regulatory Guide 2.6, Revision 2, "Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities," September 2017.

12-6 NUREG-0849, "Standard Review Plan for Review and Evaluation of Emergency Plans for Research and Test Reactors," October 1983.

12-7 Regulatory Guide 5.59, Revision 1, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance," February 1983.

12-8 University of Illinois at Urbana-Champaign – Approval of Safeguards Information Protection Plan (EPID: L-2023-NFN-0006), March 12, 2024 (ML24040A193).

12-9 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.8-1995 (R2018), "Quality Assurance Program Requirements for Research Reactors."

12-10 Regulatory Guide 2.5, Revision 1, "Quality Assurance Program Requirements for Research and Test Reactors," June 2010.

12-11 U. of I. Topical Report IMRDD-MMR-22-03-A, Release 01, "Quality Assurance Program Description for the University of Illinois Urbana-Champaign Research and Test Reactor," dated August 14, 2023 (ADAMS Accession Number ML23167C137).

APPENDIX 12A UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN RESEARCH REACTOR PRELIMINARY EMERGENCY PLAN

This preliminary Emergency Plan (EP) provides a discussion of the plans for addressing emergencies which may arise related to operation of the University of Illinois Urbana-Champaign (U. of I.) research reactor, as required by 10 CFR 50.34(a). As described in 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," these plans are generally in support of a Construction Permit Application. The EP to be submitted with the Operating License Application will contain sufficient information to ensure the compatibility of proposed emergency plans for the on-site area, which includes an Emergency Planning Zone (EPZ), with facility design features, site layout, and site location with respect to access routes and land use. The following items will be discussed in the Operating License EP:

1. Onsite and offsite organizations for coping with emergencies and the means for notification, in the event of an emergency, of persons assigned to the emergency organizations.

2. Contacts and arrangements made and documented with local, State, and Federal governmental agencies with responsibility for coping with emergencies, including identification of the principal agencies.

3. Protective measures to be taken within the site boundary and within each EPZ to protect health and safety in the event of an accident; procedures by which these measures are to be carried out (e.g., in the case of an evacuation, who authorizes the evacuation, how the public is to be notified and instructed, how the evacuation is to be carried out); and the expected response of off-site agencies in the event of an emergency.

4. Features of the facility to be provided for onsite emergency first aid and decontamination and for emergency transportation of on-site individuals to off-site treatment facilities.

5. Provisions to be made for emergency treatment at off-site facilities of individuals injured as a result of licensed activities.

6. Provisions for a training program for staff of the licensee, including those who are assigned specific authority and responsibility in the event of an emergency, and for other persons who are not staff of the licensee but whose assistance may be needed in the event of a radiological emergency.

7. A preliminary analysis that projects the time and means to be employed in the notification of State and local governments, U.S. Nuclear Regulatory Commission (NRC), and the public in the event of an emergency.

This preliminary EP follows the guidance provided in Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities," (Reference 12A-1) which endorses the latest version of ANSI/ANS 15.16, "Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities," (Reference 12A-2) and informed by NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors" (Reference 12A-3).

12A.1 INTRODUCTION

Appendix E to 10 CFR Part 50, “Emergency Planning and Preparedness for Production and Utilization Facilities,” establishes requirements for EPs to attain an acceptable state of emergency preparedness and to provide reasonable assurance that protective measures can and will be taken to protect the health and safety of workers and the public. This appendix provides the emergency planning related information required in Preliminary Safety Analysis Reports by 10 CFR 50.34(a)(10) and 10 CFR 50, Appendix E.II.

12A.1.1 Reactor Facility Description

The reactor facility is located on a site on the southwest corner of the Urbana-Champaign campus. A general description of the reactor facility is provided in Chapter 1 of the U. of I. research reactor Preliminary Safety Analysis Report (PSAR).

12A.1.2 Definitions

Table 12A-1 lists the terms and their definitions that are unique to the U. of I. research reactor EP.

Table 12A-1 Emergency Plan Terms and Definitions

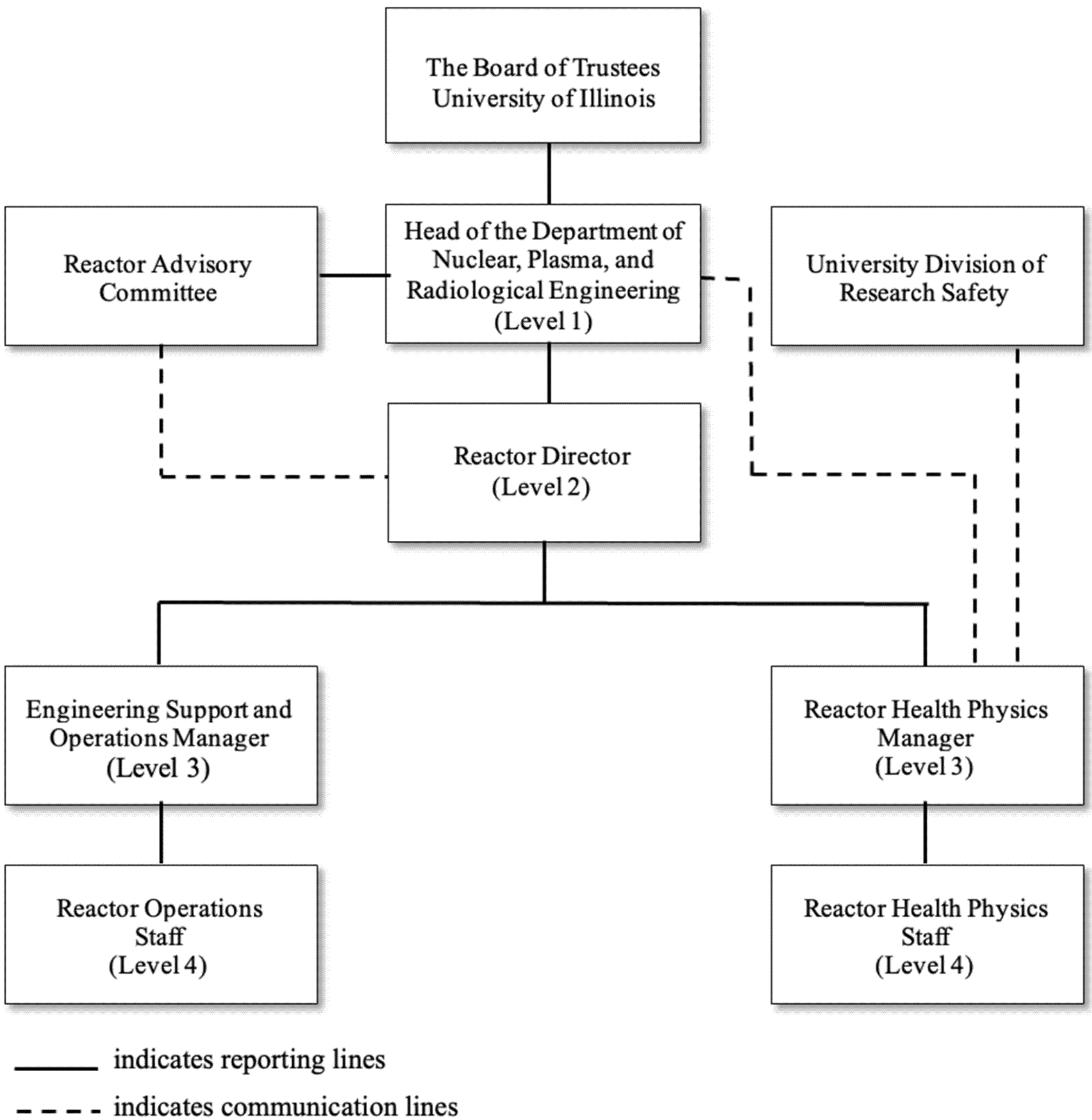
Term	Definition
emergency	An emergency is any situation that activates the U. of I. research reactor EP.
emergency action levels (EALs)	Specific instrument readings or observations, radiological dose or dose rates, or specific contamination levels of airborne, waterborne, or surface-deposited radioactive materials that are used in (a) establishing emergency classes and (b) initiating appropriate emergency measures.
emergency classes	Emergency classes are classes of accidents grouped by severity level for which predetermined emergency measures are taken or considered.
emergency plan (EP)	The U. of I. research reactor emergency plan provides the basis for actions to cope with an emergency. It outlines the objectives to be met by the emergency procedures and defines the authority and responsibilities to achieve such objectives.
emergency planning zone (EPZ)	Area for which off-site emergency planning is performed to assure that prompt and effective actions can be taken to protect the public in the event of an accident.
off-site	The geographical area that is beyond the site boundary.
on-site	The geographical area that is within the site boundary.
operations boundary	The outside walls of the U. of I. research reactor building. This area is where the Reactor Director has direct authority over all activities. The area within this boundary shall have prearranged evacuation procedures known to personnel frequenting the area.
protective action guides	Projected radiological dose or dose commitment values to individuals that warrant protective action following a release of radioactive material. Protective actions would be warranted provided the reduction in individual dose expected to be achieved by carrying out the protective action is not offset by excessive risks to individual safety in taking the protective action. The projected dose does not include the dose that has occurred prior to the assessment.
shall, should, and may	The word “shall” is used to denote a requirement; the word “should” is used to denote a recommendation; and the word “may” is used to denote a permission, neither a requirement nor a recommendation.
site boundary	The U. of I. research reactor site is bounded by E. Armory Avenue to the north and E. Gregory Drive to the south. The site boundary extends west to the outer edge of the Abbott Power Plant facilities and east to the perimeter of the Personnel Services Building, encompassing a section of S. Oak Street.
U. of I. research reactor	The deployment of Nano’s KRONOS MMR on the U. of I. campus.

12A.2 ORGANIZATIONS AND RESPONSIBILITIES

12A.2.1 Normal Organization Structure

The normal organization structure is described in Section 12.1 of the PSAR and shown below in Figure 12A-1. Facility administrative procedures shall provide the details of the normal U. of I. research reactor organization, including reporting relationships.

Figure 12A-1 U. of I. Research Reactor Normal Organization Structure (per Section 12.1 of the PSAR)



12A.2.2 Emergency Organization Structure

10 CFR Part 50, Appendix E.II.A, requires information regarding on-site and off-site organizations for addressing emergencies and the means for notification, in the event of an emergency, of persons assigned to the emergency organizations.

Since the actual personnel that staff the emergency organization will change over time, actual persons are not identified in the EP. The Emergency Director (ED) is the senior EP response-qualified individual on-shift at the onset of an emergency – or if no qualified person is at the facility, the senior person on the EP contact list that can first respond to the facility – is responsible for assessing and declaring an emergency and assuming command and control responsibilities following an emergency declaration. The EP contact list includes facility personnel by name and telephone number that are qualified to respond to an emergency. Upon declaration of an emergency, designated members of the normal staff fulfill corresponding roles in responding to the emergency. For example, health physics personnel undertake radiation protection activities; security personnel undertake security activities; engineering personnel focus on facility assessment and technical support for operations; and operations personnel focus on facility operations.

Additional personnel may be designated by U. of I. management as emergency responders providing special expertise deemed beneficial to the planned response. The individuals assigned as emergency response personnel are designated by U. of I. management based on the technical requirements and training requirements of the position. The primary responsibilities of key emergency response personnel are outlined below. The additional roles and responsibilities for emergency response personnel will be provided in the Operating License Application.

U. of I. shall provide the capability for 24-hour notification to university and non-university organizations including a primary backup means to accomplish the required communications. University organizations will be able to call in staff as needed during off-hours and for emergency support. Non-University emergency support organizations maintain their staffing as needed for emergency support.

Figure 12A-2 shows the U. of I. research reactor site layout as well as the site boundary. The U. of I. research reactor emergency organization structure is shown in Figure 12A-3.

Figure 12A-2 U. of I. Research Reactor Site Layout

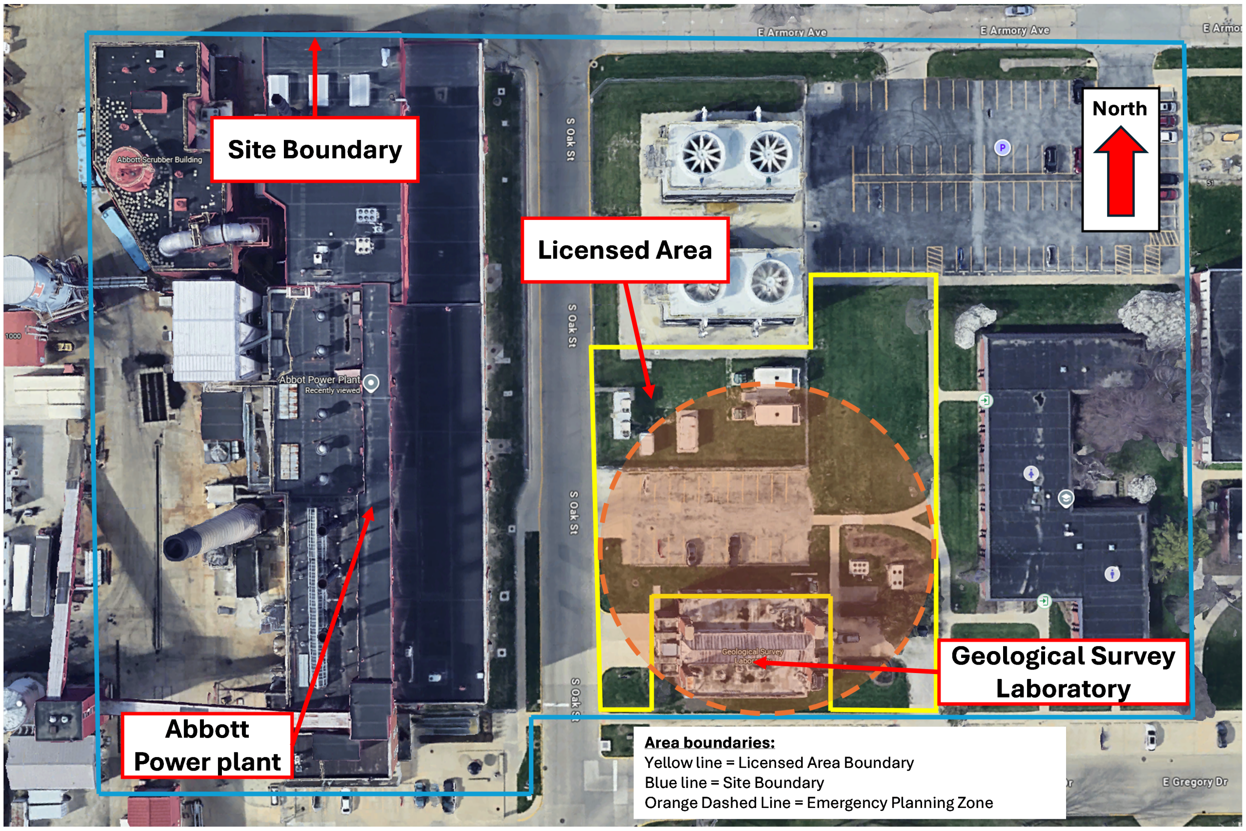
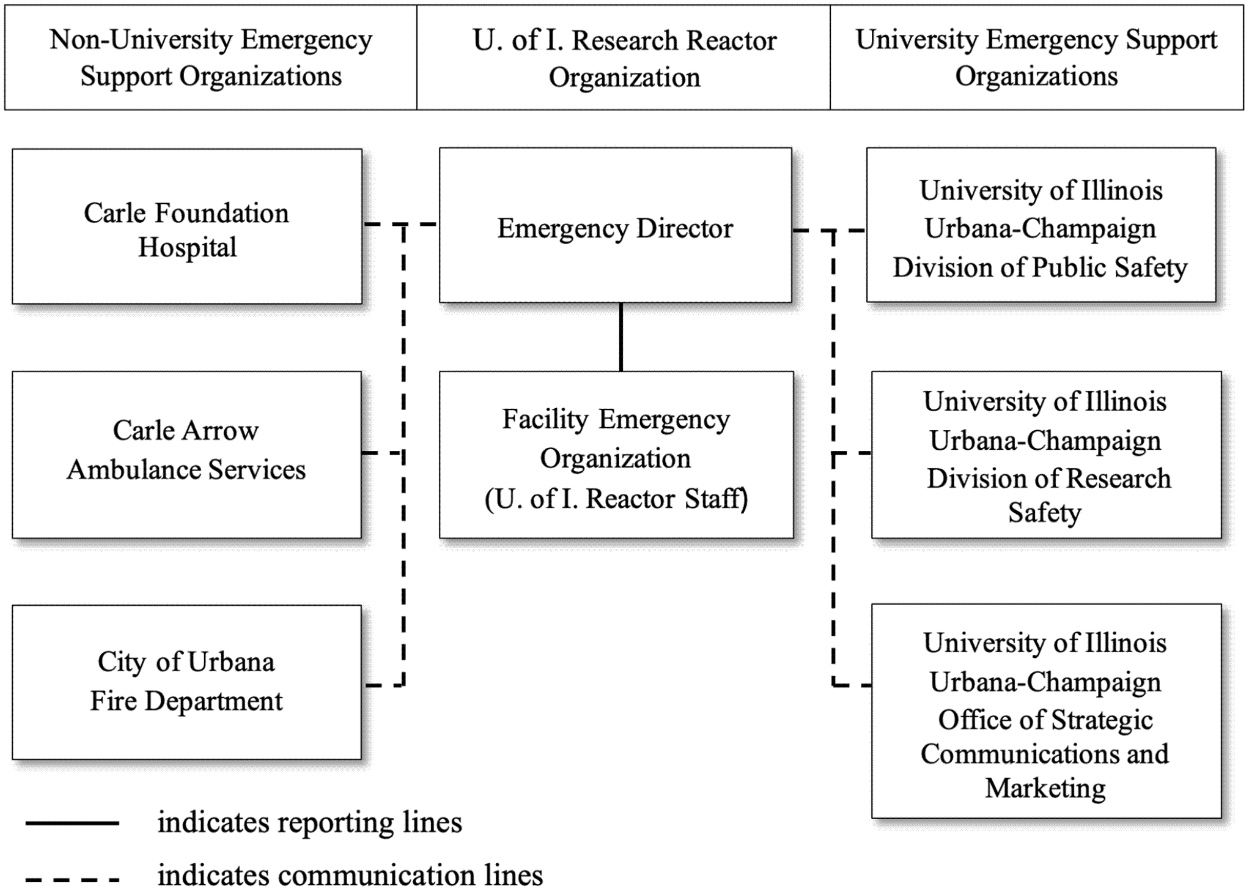


Figure 12A-3 U. of I. Research Reactor Emergency Organization Structure



12A.2.2.1 University of Illinois Research Reactor Organization

12A.2.2.1.1 Emergency Director

The ED shall be responsible for the overall direction in the event of an emergency. The senior EP response-qualified individual on-shift at the onset of an emergency, or if no qualified person is at the facility, the senior person on the EP contact list that can first respond to the facility will be the ED for that emergency. This individual will fulfill this role until duties are transitioned to the dedicated replacement.

The ED shall have the following direct responsibilities:

- Declaring and classifying the emergency;
- Directing emergency operations and ensuring proper implementation of the EP;
- Ensuring that any necessary NRC notifications are made in accordance with 10 CFR 50 Appendix E requirements;
- Authorizing volunteer emergency workers to incur radiation exposures in excess of normal occupational limits;
- Terminating the emergency and initiating recovery operations;

- Assessing conditions in the facility after termination of the emergency to determine the proper course of further recovery actions;
- Authorizing a facility or area evacuation;
- Authorizing reentry into the facility (or portion thereof) that required evacuation during the emergency;
- Establishing and coordinating recovery/re-entry efforts;
- Evaluating the cause(s) of the emergency and recommending corrective actions before returning the facility to a normal operating status;
- Coordinating emergency response actions with the off-site emergency support organizations; and
- Requesting augmented support, as required.

12A.2.2.1.2 Emergency Coordinator

The Emergency Coordinator (EC) shall be appointed by the ED to assist them, as needed, and to ensure a record of the events during and following the emergency is maintained.

The EC shall have the following direct responsibilities:

- Evaluating the need for evacuation activities and providing that recommendation to the ED;
- Accountability of individuals following a facility or area evacuation; and
- Maintaining a roster of all individuals released from the site by the ED.

12A.2.2.1.3 Reactor Health Physics Manager

The Reactor Health Physics Manager, as identified in Figure 12A-1 as part of the normal organization structure, shall be responsible for the radiological aspects of the emergency. The Reactor Health Physics Manager has the responsibility for radiological assessments on-site and off-site, which includes determining radiation and contamination levels, and using this information to determine where isolation is required.

The Reactor Health Physics Manager shall have the following direct responsibilities:

- Performing radiological assessments on-site and off-site, which includes determining radiation and contamination levels, and relaying this information to the ED;
- Supervising access to isolated areas to minimize the spread of contamination and exposure of individuals;
- Evaluating public and personnel doses received during the incident;
- Assessing subsequent potential doses and recommending protective actions, as appropriate; and
- Assisting the ED and helping determine the course of further action.

12A.2.2.2 University Emergency Support Organizations

12A.2.2.2.1 University of Illinois Urbana-Champaign Division of Public Safety

The U. of I. Division of Public Safety provides both police services as well as emergency management. Police services have the responsibility of providing and supporting security measures during an emergency, such as crowd and traffic control, and for responding to security incidents at the U. of I. research reactor. If a security incident occurs, police services shall respond and notify the ED if the incident is valid. A transition of the ED to a broader emergency response team may then occur in accordance with the EP. Police services perform assessments and response actions to security incidents with necessary assistance by the ED in accordance with the U. of I. research reactor Physical Security Plan. Responsibilities of emergency management include active shooter, campus building emergency action plans, emergency response guides, etc. The Physical Security Plan will be provided as part of the Operating License Application.

12A.2.2.2.2 University of Illinois Urbana-Champaign Division of Research Safety

The U. of I. Division of Research Safety shall be responsible for assisting the ED and the Reactor Health Physics Manager in the radiological and industrial safety aspects of the emergency.

12A.2.2.2.3 University of Illinois Urbana-Champaign Office of Strategic Communications and Marketing

The U. of I. Office of Strategic Communications and Marketing shall be responsible for relaying the necessary information about the emergency situation to the news media and the public.

12A.2.2.3 Non-University Emergency Support Organizations

10 CFR Part 50, Appendix E.II.B, requires information regarding contacts and arrangements made and documented with agencies with responsibility for coping with emergencies, including identification of the principal agencies. This section describes the authorities, responsibilities, and support functions of Federal, state, county, and local governmental agencies in an emergency situation. The information presented here pertains to any class of emergency.

Arrangements with the Carle Foundation Hospital, Carle Arrow Ambulance Services, and the City of Urbana Fire Department will be obtained, documented, and included in the Operating License Application to ensure a clear understanding of the emergency support responsibilities of each organization. U. of I. has conducted early discussions with these entities to inform them of the plans to locate the U. of I. research reactor on the Urbana-Champaign campus and have obtained preliminary documentation from them to indicate their awareness of their potential responsibilities in the EP to be submitted as part of the Operating License Application. Copies of those documents are included as Attachments to this Appendix.

12A.2.2.3.1 Carle Foundation Hospital

Carle Foundation Hospital, which is located within 2.3 miles (3.7 km) from the reactor facility, is a regional care hospital in the City of Urbana. Carle Foundation Hospital will be the provider of services and facilities to individuals at the U. of I. research reactor who have been injured, exposed, or contaminated for whom the hospital's services and facilities are necessary or appropriate.

Carle Foundation Hospital has standard operating procedures for dealing with radiological emergencies, including contaminated patients. U. of I. is working with Carle Foundation Hospital to develop appropriate response plans for potential incidents at the U. of I. research reactor and a letter documenting the engagement can be found in the Attachments to this Appendix.

12A.2.2.3.2 Carle Arrow Ambulance Services

Carle Arrow Ambulance Services will provide transport of patients from the U. of I. research reactor site to off-site medical services. U. of I. is working with the Ambulance Service to develop appropriate response plans for potential incidents at the U. of I. research reactor and a letter documenting the engagement can be found in the Attachments to this Appendix.

12A.2.2.3.3 City of Urbana Fire Department

The City of Urbana Fire Department will provide assistance during emergencies involving actual or potential fire, explosions, or injuries. Moreover, response incidents involving hazardous materials will be developed to incorporate emergency response and hazardous/materials/rescue units as may be appropriate. U. of I. is working with the City of Urbana Fire Department to develop appropriate response plans for potential responses at the U. of I. research reactor and a letter documenting the engagement can be found in the Attachments to this Appendix.

12A.2.2.3.4 U.S. Nuclear Regulatory Commission

Notification procedures (e.g., telephone, electronic messaging, written reports, etc.) will be implemented as required. The response provided by the NRC is described in NUREG-0728, "NRC Incident Response Plan." The NRC is the Coordinating Agency/Lead Federal Agency for incidents that occur at fixed facilities or activities licensed by the NRC.

12A.3 EMERGENCY CLASSIFICATION SYSTEM

This preliminary EP describes several classes of emergency situations covering the spectrum of emergency conditions that involve the alerting or activating of progressively larger segments of the emergency support organizations. To provide for improved communications between U. of I. and federal, state, and local agencies and organizations, the most severe accidents are standardized in four (4) classes of emergency conditions that group the accidents according to the severity of off-site radiological consequences. This preliminary EP includes all four (4) standard classes that may be appropriate for dealing with accident consequences for the U. of I. research reactor. Each emergency class is associated with particular Emergency Action Levels (EALs) and with particular immediate actions to provide the appropriate graded response. In order of increasing severity, the four (4) standard emergency classes are described in qualitative terms in the following subsections.

12A.3.1 Notification of Unusual Events

Notification of Unusual Events may be initiated by either man-made events or natural phenomena that can be recognized as creating a significant hazard potential that was previously nonexistent. There is usually time available to take precautionary and corrective steps to prevent the escalation of the accident or to mitigate the consequences should it occur. No releases of radioactive material requiring off-site responses are expected.

Although the situation may not have caused damage to the reactor, it may warrant an immediate shutdown of the reactor or interruption of nonessential routine functions.

Situations that may lead to this classification include:

1. threats to or breaches of security, such as bomb threats or civil disturbances directed toward the reactor;

2. natural phenomena such as tornados in the immediate vicinity of the reactor, hurricanes, or earthquakes felt in the facility; and

3. facility emergencies such as prolonged fires, fuel damage indicated by high coolant fission product activity, or high off-gas activity.

One or more elements of the emergency organization are likely to be activated or notified to increase the state of readiness as warranted by the circumstances.

12A.3.2 Alert

Events leading to an Alert would be of such radiological significance as to require notification of the emergency support organizations and its response as appropriate for the specific emergency situation. Under this class, it is unlikely that off-site response or monitoring would be necessary. Substantial modification of reactor operating status is a highly probable corrective action. Protective evacuations or isolation of certain areas within the operations boundary or within the site boundary may be necessary.

Situations that may lead to this classification include:

1. severe failure of one of the fuel coating layers or of fueled experiments where containment boundaries exist to reduce releases or less severe cladding failures in situations where fission products are not well contained; and

2. significant releases of radioactive materials as a result of experiment failures.

12A.3.3 Site Area Emergency

A Site Area Emergency may be initiated when events such as major damage of fuel or fuel coating layers and actual or imminent failure of other physical barriers containing fission products in reactor fuel or fueled experiments have occurred, and projected off-site radiological consequences exceed Section 12A.4.3 action levels.

No credible accidents attributable to the U. of I. research reactor or its operations are postulated that can cause emergency conditions at or beyond the site boundary for this classification. However, the ED retains the right to declare this class if necessary. Monitoring at the reactor site boundary is conducted to assess the need for off-site protective actions. Protective measures on-site may be necessary. An armed attack directed towards or occurring at the reactor facility may result in a Site Area Emergency.

12A.3.4 General Emergency

A general emergency may be initiated by accidents that result in an uncontrolled release of radioactive material into the air, water, or ground to the extent that protective actions off-site may be necessary.

No credible accidents attributable to the U. of I. research reactor or its operation are postulated that can cause radiological emergency conditions at or beyond the site boundary for this classification. The EPZ shall be within the site boundary and clarified in the Operating License Application. However, the ED retains the right to declare this class if necessary. Loss of physical control of the reactor facility may result in a General Emergency.

12A.4 EMERGENCY ACTION LEVELS

Emergency Action Levels (EALs) may be based on airborne Effluent Concentrations (EC), which are listed in 10 CFR 20, Appendix B, Table 2 (Reference 12A-4), at the ventilation exhaust point and other on-site parameters for which dose rates and radiological effluent releases at the site boundary can be projected. The radiation dose levels specified below are considered adequate for the credible accidents associated with the operation of research reactors and the specified action levels provide reasonable assurance that protective measures associated with the action levels can and will be taken.

The following subsections detail the EALs for the four (4) defined classes of emergencies. One or more elements of the emergency support organizations are likely to be activated or notified to increase the state of readiness as warranted by the circumstances. Although the situation may not have caused damage to the reactor, it may warrant an immediate shutdown of the reactor or interruption of nonessential routine functions. In situations where an EAL is not applicable, the ED retains the right to declare an emergency if warranted by other conditions.

12A.4.1 Emergency Action Levels for Notification of Unusual Events

Situations that may lead to this class include:

1. Actual or projected radiological effluent at the site boundary that is calculated (or measured) to result in either of the following conditions, both of which are based on an exposure of 24 hours or less:

    a. A deep dose equivalent of 0.15 mSv (15 mrem)

    OR

    b. A committed effective dose equivalent of 0.15 mSv (15 mrem) based on the following considerations:

        - For noble gases: 50 EC × 24 hours = 1.2×10^3 EC-hours ≈ 0.15 mSv (15 mrem)
        - For radionuclides other than noble gases: 100 EC × 24 hours = 2.4×10^3 EC-hour ≈ 0.15 mSv (15 mrem)

2. Report or observation of a severe natural phenomenon affecting the reactor site.

3. Receipt of a bomb threat affecting the reactor facility.

4. Credible security threat affecting the reactor facility.

5. Fire within the reactor facility that cannot be extinguished within 15 minutes.

12A.4.2 Emergency Action Levels for an Alert

Situations that may lead to this class include:

1. Actual or projected radiological effluent at the site boundary that is calculated (or measured) to result in either of the following conditions, both of which are based on an exposure of 24 hours or less:

    a. A deep dose equivalent of 0.75 mSv (75 mrem)

    OR

    b. A committed effective dose equivalent of 0.75 mSv (75 mrem) based on the following considerations:

        - For noble gases: 250 EC × 24 hours = 6.0×10^3 EC-hours ≈ 0.75 mSv (75 mrem)
        - For radionuclides other than noble gases: 500 EC × 24 hours = 1.2×10^4 EC-hour ≈ 0.75 mSv (75 mrem)

2. Actual or projected radiation levels at the site boundary of 0.2 mSv/hour deep dose equivalent (20 mrem/hour) for 1 hour or 1.0 mSv (100 mrem) to the thyroid (committed dose equivalent).

3. Security breach affecting the reactor facility.

12A.4.3 Emergency Action Levels for a Site Area Emergency

Situations that may lead to this class include:

1. Actual or projected radiological effluent at the site boundary that is calculated (or measured) to result in either of the following conditions, both of which are based on an exposure of 24 hours or less:

    a. A deep dose equivalent of 3.75 mSv (375 mrem)

    OR

    b. A committed effective dose equivalent of 3.75 mSv (375 mrem) based on the following considerations:

    - For noble gases: 1250 EC × 24 hours = 3.0×10^4 EC-hours ≈ 3.75 mSv (75 mrem)
    - For radionuclides other than noble gases: 2500 EC × 24 hours = 6.0×10^4 EC-hour ≈ 3.75 mSv (375 mrem)

2. Actual or projected radiation levels at the site boundary of 1.0 mSv/hour (100 mrem/hour) deep dose equivalent for 1 hour or 5.0 mSv (500 mrem) to the thyroid (committed dose equivalent).

3. An armed attack directed towards or occurring at the reactor facility.

12A.4.4 Action Levels for a General Emergency

The U. of I. research reactor PSAR indicates that no credible accidents attributable to the reactor or its operation are postulated to cause an emergency condition at or beyond the site boundary in excess of:

1. Sustained actual or projected radiation levels at the site boundary of 5.0 mSv/hour (500 mrem/hour) deep dose equivalent.

2. Actual or projected dose at the site boundary in the plume exposure pathway of 10 mSv (1 rem) (total effective dose equivalent) or 50 mSv (5 rem) to the thyroid (committed dose equivalent).

Loss of physical control of the reactor facility may result in a General Emergency and may be declared.

12A.5 EMERGENCY PLANNING ZONES

As part of emergency planning, U. of I. shall identify any radiological emergencies that may result in exposures exceeding 10 mSv deep dose (1 rem whole body) or 50 mSv (5 rem) to the thyroid and identify an appropriate Emergency Planning Zone (EPZ). The U. of I. research reactor is designed such that no credible accidents attributable to the reactor or its operation are postulated to require the EPZ to be extended beyond the site boundary. The EPZ will be defined in the Operating License Application.

12A.5.1 Determining an EPZ

The EPZ size depends on the distance at which the protective actions are calculated to be warranted and will be provided in the Operating License Application. A preliminary EPZ is shown on Figure 12A-2.

12A.6 EMERGENCY RESPONSE

Emergency response measures shall be identified for each emergency in the Operating License Application. These response measures shall be related to the emergency class and action levels that specify what measures shall be implemented.

12A.6.1 Activation of Emergency Organization

The method for activating the emergency organization shall be described in the Operating License Application. The EP shall specify the location(s) of current notification lists, specific actions to notify and mobilize the emergency organization, and the applicable off-site support organizations for each emergency class.

12A.6.2 Assessment Actions

The methods, systems, and equipment for gathering and processing information and data on which to base decisions to escalate or deescalate emergency response actions shall be described in the Operating License Application.

12A.6.3 Corrective Actions

The corrective actions for taking control of the emergency situation, protecting or providing aid to affected personnel, and mitigating the consequences of the emergency shall be described in the Operating License Application.

12A.6.4 Protective Actions

The EP submitted with the Operating License Application shall describe protective actions appropriate for the emergency classes as described above. It is anticipated that the EP may include the following:

1. conditions for either partial or complete on-site evacuation, evacuation routes, and primary and alternate assembly areas;

2. methods to assure personnel accountability and the segregation of potentially contaminated personnel;

3. protective measures and exposure guidelines for emergency personnel;

4. provisions for isolation and access control of facility areas to minimize exposures to radiation and the spread of radioactive contamination; and

5. the methods for monitoring radiation dose rates and contamination levels, both on-site and off-site, including provisions for transmitting collected information and data to the element of the emergency organization responsible for accident assessment.

12A.7 EMERGENCY FACILITIES AND EQUIPMENT

Descriptions of the emergency facilities, types of equipment, and their locations will be provided in the Operating License Application.

12A.7.1 Emergency Support Center

A facility or defined area within a facility shall be designated in the Operating License Application to serve as an Emergency Support Center from which emergency control directions shall be given. The Emergency Support Center shall be located to oversee operations effectively and shall be separated from actual activities to function efficiently.

12A.7.2 Assessment Facilities

Monitoring systems and laboratory facilities that are to be used to determine the need to initiate emergency measures as well as those to be used for continuing assessment shall be identified in the Operating License Application. These monitoring systems may consist of equipment such as radiological monitors, personnel monitoring, sampling equipment, earthquake sensors, fire and combustion product detectors, and process monitors that provide pertinent facility system or status information.

12A.7.3 First Aid and Medical Facilities

The measures that will be used to provide necessary assistance to persons injured or exposed to radiation will be identified in the Operating License Application. The capabilities for decontamination, administering first aid, and transporting injured personnel along with the arrangements for medical treatment shall be described. The following items shall be included:

1. capabilities for decontaminating personnel for their own protection and to prevent or minimize further spread of contamination;

2. first aid training and capabilities of the emergency support organizations;

3. arrangements for transporting injured personnel who also may be contaminated to medical treatment facilities;

4. arrangements for local hospital and medical services; and

5. assurance that hospital and medical services can provide the required services, and those persons providing them are available, prepared, and qualified to manage radiological emergencies; written agreements with respect to arrangements made for hospital and medical services shall be included.

12A.7.4 Communication Equipment

The systems of emergency communications that will be available to communicate instructions and information both on-site and off-site throughout the course of the emergency shall be identified in the Operating License Application. U. of I. shall establish reliable primary and backup means of communication (e.g., public telephone and radio) that are compatible with local off-site support groups.

12A.7.5 Contingency Planning

Contingency plans in the case an emergency render any of the above facilities or equipment unusable shall be established and described in the Operating License Application. This does not mean fully redundant dedicated backup facilities or equipment, but rather established agreements with alternate facilities from which the needed functions can be performed in the event of an emergency.

12A.8 RECOVERY

As part of the Operating License Application, this element of the EP shall describe the criteria for restoring the reactor facility to a safe status including reentry into the reactor facility or portions of the facility that may have been evacuated because of the accident. The operations to recover from most severe accidents may be complex and depend on the actual conditions at the facility; therefore, it is not practicable to plan detailed recovery actions for all conceivable situations.

12A.9 MAINTAINING EMERGENCY PREPAREDNESS

The elements necessary for maintaining an acceptable state of emergency preparedness shall be described in the Operating License Application. A description shall be provided of how the effectiveness of the EP will be maintained, including training, review, and update of the EP and associated implementing procedures along with maintenance and inventory of equipment and supplies that would be used in emergencies. Frequent coordination with the emergency support organizations shall also be maintained to ensure the necessary training and the efficient use of their capabilities.

12A.9.1 Training and Drills

The following shall be identified or described, as applicable, to demonstrate emergency preparedness in the Operating License Application:

1. programs to train and periodically retrain on-site personnel for participation in the EP and to provide specified training to on-site and off-site personnel who have specific emergency assignments;

2. on-site emergency drills to be conducted as action drills with each required emergency measure being executed as realistically as is reasonably possible, including the use of appropriate emergency equipment; and

3. provision for critiques of all drills, including timely evaluation of observer comments, correction of identified deficiencies, and revision of implementing procedures, as needed.

12A.9.2 Plan Review and Update

Reviewing, revising, and updating of the EP shall be described in the Operating License Application. This includes specifying the methods to ensure that changes and revisions are reviewed, approved, and distributed to appropriate elements of the emergency support organizations.

12A.9.3 Equipment Maintenance

Provisions to ensure operational readiness of emergency equipment and supplies, including required maintenance and calibrations, testing, and periodic inventory shall be described in the Operating License Application.

12A.10 REFERENCES

12A-1 Regulatory Guide 2.6, Revision 2, "Emergency Planning for Research and Test Reactors and Other Non-Power Production and Utilization Facilities," September 2017.

12A-2 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.16-2015 (R2024), "Emergency Planning for Research Reactors."

12A-3 NUREG-0849, "Standard Review Plan for Review and Evaluation of Emergency Plans for Research and Test Reactors," October 1983.

12A-4 10 CFR 20, Appendix B, "Annual Limits on Intake (ALIs) and Derived Air Concentrations (DACs) of Radionuclides for Occupational Exposure; Effluent Concentrations; Concentrations for Release to Sewerage."

APPENDIX 12B LETTERS OF ENGAGEMENT

611 W. Park Street, Urbana, IL 61801 | (217) 383-3311 | carle.org

Carle

December 17, 2025

Caleb S. Brooks, Director
Illinois Microreactor Demonstration Project
Grainger College of Engineering University of Illinois Urbana-Champaign
104 South Wright Street
Urbana, IL 61801

Dear Dr. Brooks,

The Illinois Microreactor Demonstration Project site, located within the Grainger College of Engineering at the University of Illinois in Urbana, IL, is within the area served by Carle Foundation Hospital.

This letter serves to recognize the cooperative effort between the University of Illinois and Carle Foundation Hospital and acknowledges that the University is working with Carle Foundation Hospital to develop appropriate response plans to an emergency incident at the Illinois Microreactor Demonstration Project facility.

In accordance with its capacity, licensure, and then-current capabilities, Carle Foundation Hospital will be the provider of services and facilities to individuals at the microreactor facility who have been injured, exposed, or contaminated for whom the hospital's services and facilities are necessary or appropriate.

Sincerely,

A Rinehart

Allen Rinehart MSN, RN, TNS, CEN, CENP
Vice-President – CFH Inpatient Operations
Carle Foundation Hospital
611 W Park St
Urbana, IL 61801

Carle
Arrow Ambulance

12/16/25

Caleb S. Brooks

Director, Illinois Microreactor Demonstration Project

Grainger College of Engineering

University of Illinois Urbana-Champaign

104 South Wright Street Urbana, IL 61801

Dear Dr. Brooks,

The Illinois Microreactor Demonstration Project site is within the area served by Carle Arrow Ambulance Services. Arrow Ambulance will provide transport of patients from the microreactor site to off-site medical services.

This letter serves to recognize the cooperative effort between the University of Illinois and Carle Arrow Ambulance Services and acknowledges that the University is working with Arrow Ambulance to develop appropriate response plans to an emergency incident at the Illinois Microreactor Demonstration Project site.

Patients will be transported to medical facilities as requested by patients or as directed. Emergency response plans for patient care and transport shall be executed while following Arrow Ambulance protocols and Standard Operating Procedures to include foreseeable incidents.

Sincerely,

Justin Stalter

Director

Carle Arrow Ambulance Services

Urbana Fire Department
400 South Vine Street
Urbana, IL 61801
(217) 384-2420
FAX (217) 384-2449

1-20-2026

Caleb S. Brooks
Director, Illinois Microreactor Demonstration Project
Grainger College of Engineering
University of Illinois Urbana-Champaign
104 South Wright Street
Urbana, IL 61801

Dear Dr. Brooks,

This letter serves to recognize the cooperative effort between the University of Illinois Urbana-Champaign and the Urbana Fire Department (UFD). This letter acknowledges that the University is working with the UFD to develop appropriate response plans to an emergency alert at the Illinois Microreactor Demonstration Project facility.

Future UFD response will be contingent upon the type of emergency and the information provided by the caller to the UFD. The University is working to provide UFD responders with knowledge and understanding of microreactor facility operations to assist in this effort.

Moreover, response to incidents involving hazardous materials will be developed to incorporate emergency response, including nearby fire apparatus and hazardous materials/rescue units as may be appropriate. Further, it is my understanding that the University plans to provide a facility staff member(s) who will also respond to emergencies to help provide assistance. The representative(s) from the University is expected to be knowledgeable of the facilities and the hazards, and have the ability to assist in coordinating necessary emergency response activities.

Sincerely,

Tal D Prendergast

Tal Prendergast
Interim Fire Chief
Urbana Fire Department

USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 13 - ACCIDENT ANALYSES

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
CCV	cross connection vessel
CRDU	Control Rod Drive Unit
DC	Direct Current
DLOFC	Depressurized Loss of Forced Cooling
ESFs	Engineered Safety Features
{{ }}a(4)	{{ }}a(4)
FHSS	Fuel Handling and Storage Systems
HPRS	helium pressure relief system
HPS	Helium Purification System
HTGR	high temperature gas cooled reactor
IAEA	International Atomic Energy Agency
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratory
MHA	Maximum Hypothetical Accident
NRC	Nuclear Regulatory Commission
ODSL	outer dense surface layer
OLA	Operating License Application
PIEs	postulated initiating events
PLOFC	Pressurized Loss of Forced Cooling
RCCS	Reactor Cavity Cooling System
RCS	Reactor Control System
RCSS	Reactivity Control and Shutdown System
RG	Regulatory Guide
RPS	Reactor Protection System
RV	Reactor Vessel
SSCs	Structures, Systems and Components
TEDE	total effective dose equivalent
TESS	Thermal Energy Storage System
TRISO	tri-structural isotropic
UPS	uninterruptible power supplies
VS	Vessel System

CHAPTER 13 ACCIDENT ANALYSES

This chapter provides information and analyses that demonstrate the safety of the University of Illinois at Urbana Champaign (U. of I.) research reactor to avoid undue risk to the health and safety of the public, workers, and the environment. The potential radiological consequences of Structures, Systems, and Components (SSCs) malfunctions and disturbances are considered and analyzed using conservative models and under limiting conditions. This highlights that no postulated event can lead to unacceptable radiological consequences to people or the environment.

The safety basis for the research reactor is simple. The core, as described in Chapter 4, is comprised of materials that are very thermally robust. The temperature margins between normal operation and safety limits are very large. During postulated events, the functional containment provided by the tri-structural isotropic (TRISO) fuel particles and ceramic fuel pellets, described in Section 6.2, results in the retention of the vast majority of radionuclides. Passive decay heat removal to the ultimate heat sink is provided by the Reactor Cavity Cooling System (RCCS), described in Section 6.3, and ensures that no event can result in temperatures that challenge functional containment nor the performance of safety functions by any of the SSCs. The fail-safe Reactor Protection System (RPS) ensures that safe shutdown is achieved and maintained under any conditions.

Postulated event sequences are identified in accordance with the Event Sequence Identification and SSC Safety Classification Methodology Topical Report (Reference 13-1) and guidance contained in NUREG-1537 (Reference 13-2, Reference 13-3). These events are bounded by the Maximum Hypothetical Accident (MHA), a hypothetical scenario for which conditions are conservatively defined to result in dose consequences that bound those from all other postulated events. The dose consequences of the MHA demonstrate the acceptability of the design with respect to regulatory dose limits for the public and workers. The acceptable dose criterion to a member of the public is defined in 10 CFR 50.34(a)(1)(i) (Reference 13-4) for non-power utilization facilities to be below 1 rem (10 mSv) total effective dose equivalent (TEDE) for the duration of the MHA. There is no accident-specific dose limit for workers for non-power utilization facilities. The dose to workers is evaluated against the maximum allowable annual occupational TEDE limit to facility staff defined in 10 CFR Part 20.1201(a)(1)(i) (Reference 13-5) as 5 rem (50 mSv) per year. Main Control Room habitability will be assessed for compliance with PDC-19 (Reference 13-6) as part of the Operating License Application (OLA).

Scenario definitions and analyses for the MHA and other postulated event groups are covered in Sections 13.1 and 13.2, respectively. The analyses show that the consequences of all postulated events are conservatively bounded by those of the MHA.

13.1 INITIATING EVENTS AND SCENARIOS

In accordance with the guidance in NUREG-1537, this section discusses the events postulated to assess, quantify and demonstrate the research reactor safety performance against the licensing basis. The events are grouped according to type and characteristics of the events. The postulated event categories are:

1. MHA

2. Insertion of Excess Reactivity

3. Depressurized Loss of Forced Cooling (DLOFC) with air ingress
4. Pressurized Loss of Forced Cooling (PLOFC)
5. Mishandling or Malfunction of Fuel
6. Experiment Malfunction (not applicable for the research reactor)
7. Loss of Normal Electrical Power
8. Internal and External Hazards
9. Mishandling or Malfunction of Equipment

The MHA is a hypothetical scenario that results in dose consequences that bound all other postulated event categories. The MHA is defined to encompass all conditions that can arise during any postulated event and that challenge the performance of the functional containment radionuclide barriers.

Figures of merit are defined to evaluate the acceptability of the research reactor response and consequences for each postulated event category. For the MHA, the figures of merit are dose consequences to the public and workers, which are evaluated against the U.S. Nuclear Regulatory Commission (NRC) dose criteria. The analysis of the MHA demonstrates that the maximum radiological consequences that could result from any postulated event meet the regulatory dose criteria for research reactors and that the design is therefore acceptable.

The dose consequences of all other postulated events must be bounded by those of the MHA. Not all event categories are analyzed quantitatively. Experiment malfunction is not applicable to the U. of I. Research Reactor because there is no in-core experiment. Other categories, such as loss of normal electrical power, internal and external hazards, and mishandling and malfunction of equipment, are qualitatively assessed to justify they are bounded by a different limiting postulated event.

For postulated events that are analyzed quantitatively, two figure-of-merit types can be evaluated to ensure that the MHA dose consequences are bounding: (1) dose consequences; and (2) surrogate metrics. Evaluating postulated event dose consequences offers straightforward comparison to ensure that the MHA is bounding but requires complex analysis. Therefore, where possible, surrogate metrics are used as figures of merit. The surrogate metrics for each postulated event are developed based on the applicable radionuclide release pathways and necessary SSC functionality. The figures of merit and their acceptance criteria are summarized in Table 13-1 and discussed in detail in Section 13.2.

Each figure of merit for a postulated event must be shown to meet acceptance criteria derived from the MHA conditions. For example, elevated fuel temperatures arising from a postulated event can result in diffusion of radionuclides past the functional containment barriers. Therefore, the time-temperature profile of the fuel is a figure of merit, with the acceptance criteria that it is bound by the time-temperature profile defined for the MHA. Alternatively, design limits ensure SSCs such as the fuel, core barrel, and Reactor Vessel (RV) can perform their safety functions. The functionality of these SSCs is presumed for the MHA; therefore, these design limits are figures of merit for the other postulated events. If all figures of merit for a postulated event meet the associated acceptance criteria, the dose from that event is bounded by the MHA.

Table 13-1 Figures of Merit and Acceptance Criteria for Postulated Events

Figure of Merit	Acceptance Criterion	Applicable Event(s)
Public exposure	Below 1 rem (10 mSv) TEDE per 10 CFR 50.34(a)(1)(i) (Reference 13-4)	• MHA
	Bounded by the MHA public TEDE	• Mishandling or Malfunction of Fuel
Worker exposure	Below 5 rem (50 mSv) TEDE per 10 CFR 20.1201(a)(1)(i) (Reference 13-5)	• MHA
	Bounded by the MHA worker TEDE	• Mishandling or Malfunction of Fuel
Peak fuel time-temperature profile	Generally bounded by the MHA fuel time-temperature profile	• Insertion of Excess Reactivity • DLOFC • PLOFC
Peak radial fuel temperature tilt	Below the threshold for increased risk of thermal stress failure {{ }}a(4)	• Insertion of Excess Reactivity
Peak fuel graphite time-temperature	Generally bounded by MHA graphite time-temperature profile	• Insertion of Excess Reactivity • DLOFC • PLOFC
Peak reflector graphite time-temperature	Generally bounded by MHA reflector time-temperature profile	• Insertion of Excess Reactivity • DLOFC • PLOFC
Peak Reactor Vessel temperature	Below the ASME BPVC Section III, Division 5, Subsection HB limit for low-alloy steel (540°C) (Reference 13-7)	• Insertion of Excess Reactivity • DLOFC • PLOFC
Peak core barrel temperature	Below the limit set in NRC Regulatory Guide (RG) 1.87 Rev. 2 for {{ }}a(4) covered by ASME BPVC Section III, Division 5, Subsection HG (725°C) (Reference 13-8)	• Insertion of Excess Reactivity • DLOFC • PLOFC

Section 14.1 of this document proposes two Safety Limits that pertain to fuel and RV temperatures. The peak fuel temperature Safety Limit protects against the risk of significant TRISO particle failure during postulated events. Past safety tests have shown negligible failures below 1600°C (Reference 13-9). This Safety Limit is encompassed by the figure of merit on peak fuel time-temperature profile because the MHA fuel time-temperature profile is set well below the applicable limit. The Safety Limit for RV temperature is directly related to the figure of merit on peak RV temperature.

The methodology for identifying and screening postulated initiating events (PIEs) that can lead to postulated events has been approved by the NRC (Reference 13-1). The three categories of PIEs for a high-temperature gas-cooled reactor (HTGR) like the U. of I. Research Reactor are: (1) pipe breaches;

(2) non-pipe breach transient events; and (3) internal and external hazards. An initial list of PIEs was determined based on historical operational experience, past regulatory documents, and expert judgement. For each event category, this section reports the initial list of PIEs, identifies the limiting postulated event scenario and explains how other postulated events are precluded by design.

With one exception, the considered postulated events are assumed to occur during full power operation. The exception is the limiting postulated mishandling or malfunction of fuel event, which would occur during refueling under shutdown conditions. The list of PIEs and the limiting postulated event for each event category will be updated to include PIEs that occur in other reactor modes and incorporate License and Technical Specification limitations. This will be included in the Operating License Application (OLA).

Postulated event consequences may be mitigated by non-safety related structures, systems, and components (SSCs), if available, for reactor control, heat removal, and confinement. However, consistent with NUREG-1537, only the performance of the safety related SSCs is credited. The performance of credited safety-related SSCs also assumes the most constraining single failure of any active components. This conservative approach provides additional confidence that the postulated events are effectively bounded by the MHA. SSC safety classifications are provided in Section 3.6.

For all postulated events, a safe and stable state is established when:

1. The core is subcritical and long-term reactivity control is assured.

2. Decay heat is being removed and long-term cooling is assured. Figure of merit temperatures for the relevant SSCs fuel, graphite, core barrel, and RV are stable or steadily decreasing.

13.1.1 Maximum Hypothetical Accident

Per NUREG-1537, the MHA is a hypothetical postulated event scenario analyzed under conservative conditions that results in dose consequences that bound all other postulated events. This section describes the MHA and the conservative assumptions applied to ensure that the dose consequences bound all other postulated events.

For HTGRs like the U. of I. research reactor, all PIEs that affect the reactor during full-power operation will ultimately result in one of two conditions: PLOFC or DLOFC. In both scenarios, forced flow of primary coolant is lost either due to the PIE or as part of the reactor response, and the reactor must rely on passive removal of fission and decay heat. Postulated PLOFC and DLOFC events are explicitly analyzed as their own event categories, but similar conditions also result from the other event categories that affect the reactor, including: insertion of excess reactivity; loss of normal electrical power; internal and external hazards; and mishandling and malfunction of equipment, to the extent that it affects the reactor during operation. As such, PLOFC and DLOFC conditions are representative end states for all full-power postulated event categories.

The PLOFC condition may be caused by a PIE, such as loss of cooling, or as a result of the reactor response to another PIE, such as reactivity insertion or loss of normal electrical power. In PLOFC conditions, the pressure boundary is not affected by the PIE and radionuclides diffusing out of solid structures (e.g., fuel and graphite) accumulate in the primary coolant. During normal operations and postulated events, the primary coolant may leak slowly out of the pressure boundary, resulting in some of the radionuclides contained in the primary coolant escaping the pressure boundary. If the primary coolant pressure increases beyond the set point, the helium pressure relief system (HPRS) valves, described in Section 4.3, will open to reduce the primary coolant pressure, causing a release of radionuclides; these valves close when the primary coolant pressure has sufficiently decreased, limiting the extent of the release. However, as long as the pressure boundary has not failed during the PLOFC event, there is no energetic release of primary coolant to drive radionuclides from the system to the environment. As a result, events that result in PLOFC conditions are not likely to cause significant dose consequences.

DLOFC conditions may arise from failures of the primary coolant pressure boundary caused by different PIEs, including: sudden mechanical failure of the Vessel System (VS); or internal and external hazards, such as earthquakes. DLOFC events result in an energetic release of primary coolant, including radionuclides contained in it. The sudden depressurization also lifts and entrains some of the radionuclides that have been deposited around the primary coolant system. As such, the DLOFC event is more likely than a PLOFC event to result in radionuclide release and dose consequences. However, depressurization would be complete well before significant amounts of radionuclides have time to diffuse out from the fuel. While radionuclide diffusion would continue for as long as the fuel temperature remains elevated, there would be no major driving force to transport those radionuclides from the reactor to the environment.

Given the importance of the PLOFC and DLOFC conditions to all full-power postulated event categories, the MHA for the U. of I. research reactor is defined to combine the most limiting aspects of PLOFC and DLOFC conditions and thereby challenge the performance of the functional containment radionuclide barriers. The MHA conditions are equivalent to those experienced during a DLOFC event that is analyzed with very conservative assumptions. In particular, the assumption that allows for the combination of PLOFC and DLOFC challenges to functional containment is that all radionuclides diffusing out of the fuel and graphite the reactor, for the entire duration of the MHA after the initial blowdown, are immediately released to the atmosphere. This represents a permanent driving force transporting radionuclides from the reactor core to the environment. The full set of MHA assumptions and their implications is given in Section 13.2.1.

The MHA analysis focuses on the assessment of dose consequences. The dynamic response of the reactor core to depressurization is not explicitly modeled. Instead, the temperatures during the MHA are defined conservatively to establish bounding dose consequences. Fuel and graphite temperatures are key drivers of the dose consequences during most postulated events and constitute one of the main differentiators of postulated event severity. Long periods of elevated fuel temperature increase diffusion of radionuclides through the TRISO and fuel pellet functional containment barriers. They also cause tritium contained in the graphite to be released. The conservative definition and analysis of the MHA ensure that the dose consequences bound all other postulated events. The temperatures used for the MHA are adopted as the acceptance criteria for the figures of merit by which the other event categories are assessed.

13.1.2 Insertion of Excess Reactivity

The limiting insertion of excess reactivity event is the continuous withdrawal of the highest worth control rod of the Reactivity Control and Shutdown System (RCSS), described in Section 4.2. The event could be initiated by a failure of the Reactor Control System (RCS) or of the Control Rod Drive Unit (CRDU), or as a result of a human-factor error. The control rod withdrawal would result in a positive reactivity insertion and, in turn, in a rapid increase of the reactor power. The RPS would detect the overpower and trip the reactor, rapidly inserting all other control rods via gravity and stopping the helium (primary coolant) circulator. The RPS is described in Section 7.4. After a reactor trip, decay heat is passively transferred from the reactor core to the RCCS (described in Section 6.3). The RCCS is assumed to operate in its passive mode, which relies only on safety-related components.

The analysis of the limiting insertion of excess reactivity event is detailed in Section 13.2.2.

The limiting postulated event bounds other possible insertion of reactivity events, including:

- <u>Events caused by overcooling</u>

  Overcooling of the reactor core could potentially be caused by circulator or cold molten salt pump overspeed. The magnitude and rate of positive reactivity insertion caused by these postulated events is expected to be lower than that caused by control rod withdrawal at full power because: (1) the extent of overcooling is limited, either by the maximum circulator overspeed {{ }}[a(4)] or cold salt temperature (275°C), whereas the removal of a control rod can proceed until the system responds; and (2) overcooling leads to small reactivity changes over the whole core, whereas control rod withdrawal may have strong local effects, potentially increasing the thermal stress on the fuel. Analysis of overcooling events will be included in the OLA.

- <u>Events caused by physical shifting of fuel blocks, reflectors, or shields</u>

  Because the core geometry is constrained by the safety-related core barrel system, the reactivity insertion caused by these postulated events is lower in magnitude than that caused by control rod withdrawal.

The following reactivity events are precluded by design:

- <u>Simultaneous withdrawal of multiple control rods</u>

  Each of the seven control rods, part of the RCSS, is individually controlled and actuated. Hardware and software interlocks prevent movement of more than one control rod at a time, precluding simultaneous withdrawal of multiple control rods. The design basis of the RCSS is described in Section 4.2. The design basis of the RCS is described in Chapter 7.3.

- <u>Control rod ejection</u>

  Control rod ejection requires a complete fracture of the interface between the RV and CRDU that would result in the ejection of the CRDU and the attached control rod. All vessels that are a part of the reactor pressure boundary are designed in accordance with ASME BPVC Section III, Division 5, Subsection HB (Reference 13-7), and are regularly inspected and maintained, as described in Section 4.3. Brittle fracture would be required for a sudden ejection of the CRDU assembly, but is precluded due to compliance with ASME design, fabrication and maintenance rules. Additionally, the carbon steel ductile-to-brittle fracture transition temperature is significantly lower than the anticipated temperatures in the RV. Ductile fracture of the CRDU or RV, which would be the primary mode of failure under the anticipated conditions, would result in leakage of the primary coolant rather than a detachment of the CRDU from the RV.

- Control rod withdrawal followed by failure to shut down

  The RPS and RCSS are safety related and designed with sufficient reliability and redundancy to rely on their safety function to shut down the reactor.

- Reactivity insertion events followed by loss of passive RCCS function

  The passive functionality of the RCCS is safety related and designed with sufficient reliability and redundancy to credit its ability to remove decay heat, thereby ensuring a safe and stable state can be achieved and maintained.

13.1.3 Depressurized Loss of Forced Cooling

Postulated DLOFC events are defined by the magnitude and location of the pressure boundary breach. Breaches can range from small, slow leaks to large, sudden breaks. Larger breaches are less likely than smaller breaches but can have more significant consequences.

The limiting DLOFC event is defined as an instantaneous breach in the largest pipe connected to the pressure boundary. The pipe connects the Vessel Systems (VS) to the Helium Purification and Primary Coolant Make-Up Systems, described in Sections 5.4 and 5.5, respectively. Following the breach, the primary coolant can exit the pressure boundary into the Intermediate Heat Exchanger (IHX) cavity and through the depressurization pathway to the environment, which is described in Section 3.5.3. The loss of primary coolant results in a quick decline in primary coolant pressure. Once the low pressure setpoint is reached, the RPS trips the reactor, resulting in the insertion of all the control rods and circulator shutdown. Depressurization continues until the system reaches atmospheric pressure. With the loss of forced cooling, the fuel and core materials heat up, and heat is transferred from the reactor core to the RCCS. The RCCS is assumed to operate in its passive mode, which relies only on its safety-related components.

After the depressurization, residual air in the reactor cavity may enter the pressure boundary and cause oxidation of core structures and components. Air must diffuse from the breach through the primary loop to reach the core. The fuel consists of TRISO particles within a SiC matrix, which is not vulnerable to oxidation at the temperatures expected for the research reactor and tends to form a passive, protective oxide layer that prevents further oxidation (Reference 13-9). However, the graphite in the core can oxidize. The main consequences of graphite oxidation are the loss of strength and potential release of tritium. The rate and extent of oxidation is limited due to the type and location of the breach.

The analysis of the limiting DLOFC event is detailed in Section 13.2.3.

The limiting DLOFC event is expected to bound other postulated DLOFC events, including:

- Smaller DLOFC events

  DLFOC events that are caused by smaller breaches are expected to be less severe. The primary driver for reduced consequences is a reduction in the dose consequence contribution of the liftoff of plated-out fission products. A slower depressurization results in lower helium velocities and lower shear forces on the surfaces in contact with the primary coolant, resulting in less liftoff.

- Spurious or inadvertent actuation of the HPRS valves

  Any opening of the HPRS valves (described in Section 4.3) is bounded by the limiting DLOFC because: (1) the rate of depressurization through the HPRS valves is lower than through the breach assumed in the limiting DLOFC event, resulting in lower primary coolant velocities and therefore lower shear forces that could lift off plated-out radionuclides from surfaces in contact with the primary coolant.

The following DLOFC events are precluded by design:

- Chimney breaks

  A chimney break involves simultaneous rupture of the top and bottom of the RV, enabling "chimney-like" buoyancy-driven airflow through the reactor core. This event requires multiple concurrent failures and is prevented by the robust design and regular inspection of the VS, as described in Section 4.3. Preliminary analysis of pipe breach PIEs demonstrates that concurrent breaches in the VS are unlikely.

- Double-guillotine breaks of the cross-connection vessel (CCV)

  A double guillotine break of the CCV (a full circumferential break of both the CCV and hot gas duct) could create a pathway for air to circulate naturally through the reactor core after depressurization. This event is prevented by the robust design and regular inspection of the CCV and vessels generally, as described in Section 4.3. The CCV is also seismically qualified, and there are no moving components or operator actions during operation that would contribute to CCV failure. Preliminary analysis of pipe breach PIEs demonstrates that large breaks of the cross duct are unlikely.

- Loss of coolant inventory followed by failure to shut down

  The RPS and RCSS are safety related and designed with sufficient reliability and redundancy to rely on their safety function to shut down the reactor.

- DLOFC followed by loss of passive RCCS function

  The passive functionality of the RCCS is safety related and designed with sufficient reliability and redundancy to credit its ability to remove decay heat, thereby ensuring a safe and stable state can be achieved and maintained.

- <u>DLOFC followed by large-scale oxidation by air</u>

  Large-scale oxidation requires: (1) a pathway and driving force for circulation through the core; and (2) a large supply of air. Air ingress will be discussed in detail in the OLA, but the effects of oxidation are anticipated to be minor. Large-scale oxidation is precluded by:

  - Limiting the possibility of creating circulation pathways; by preventing chimney and guillotine breaks, the likelihood for large draft forces is minimized.
  - Limiting the volume of air available to cause oxidation; no pathway exists for notable amounts of air to enter the Citadel building. The amount of air expected to be available is not sufficient to drive notable oxidation.

- <u>DLOFC followed by large-scale molten salt ingress</u>

  The IHX, described in Section 5.2, is designed to prevent large scale failure. However, in the event of a breach of the boundary between the primary and secondary coolants, the helium can escape through the secondary coolant piping system. After the depressurization, the molten salt may be able to enter the primary loop. A limited amount of molten salt in the primary loop would not affect reactor safety because: (1) it will stay liquid or freeze at the temperatures in the primary loop; (2) it does not cause exothermic chemical reactions with the materials in the core; and (3) it is a neutron absorber. Any molten salt that drains into the primary loop would accumulate in the lowest parts of the system, which is the CCV. Because the CCV interfaces with the RV and IHX vessel and is designed to withstand forces resulting from the nominal internal helium pressure, forces resulting from the possible weight of the salt in the CCV would not challenge its mechanical integrity. Once the CCV is filled with molten salt, no additional salt would be able to enter the primary loop as gas would be trapped in the RV with nowhere to escape. Larger scale flooding could occur if the cold molten salt pump continues running, but this is considered highly unlikely due to: (1) the possibility for the non-safety related Supervisory Control System or simple operator action to trip the cold molten salt pumps; and (2) the expected closure of the molten salt {{ }}[a(4)] valves at the inlet and outlet of the IHX when helium is detected in the Thermal Energy Storage System (TESS). If the core were to flood with molten salt, the consequences would be manageable for the aforementioned reasons. Additionally, the RPS would generate a low-pressure trip reactor scram.

13.1.4 Pressurized Loss of Forced Cooling

The limiting PLOFC event is defined as a spurious helium circulator trip followed by a failure of the RCS to initiate a reactor shut down. The loss of forced primary coolant flow causes an initial increase in fuel and core temperatures, and the resulting negative reactivity insertion from passive reactivity feedback shuts down the reactor. The RCCS is assumed to operate in its passive mode, which relies only on its safety-related components. In response to the reactor shutdown, the RPS enables the "neutron flux in shutdown mode" trip, described in Section 7.4.3.

Eventually, the decay of xenon, results in the core reactivity increasing progressively over time. Eventually, this increase in reactivity leads to a fission power increase. The RPS detects the increase in fission power when the measured neutron flux exceeds a very small fraction {{ }}[a(4)] of the full power value. The RPS then trips the reactor via the "neutron flux in shutdown mode" trip. The control rods insertion provides ample negative reactivity to offset all temperature and xenon effects, ensuring shutdown is maintained.

The analysis of the limiting PLOFC event is detailed in Section 13.2.4.

The limiting PLOFC event is expected to bound other postulated PLOFC events, including:

- Loss of normal electrical power event

  A loss of normal electrical power event will result in helium circulator trip along with immediate control rod drop via gravity. Therefore, the limiting PLOFC event results in a larger amount of heat to be handled, and therefore additional challenge to the identified figures of merit. Additionally, the immediate insertion of control rods results in immediate shutdown with no potential for re-criticality.

- Loss of secondary heat sink

  Loss of secondary heat sink involves a flow stoppage of the molten salt secondary coolant in the TESS, potentially due to a spurious valve closure, failure of the cold molten salt pump, or a breach in the cold molten salt piping. The event causes a reactor trip on high inlet temperature and the HPRS will actuate to relieve accumulated pressure. However, the event concludes before the reactor sees elevated helium temperatures and the near-immediate scram results in a rapid decrease in reactor power. As such, the limiting PLOFC event results in a larger amount of heat to be handled, and therefore additional challenge to the identified figures of merit.

- Abnormally high core bypass flow

  Unexpectedly high levels of primary coolant bypass flow around the fuel channels would increase temperatures in the fuel. Because the reactor has large operational and safety margins on fuel temperature, as discussed in Section 4.2, excess bypass flow is not anticipated to challenge the identified figures of merit. This will be quantified in the OLA.

- Fuel channel blockage

  Blockage of a fuel channel would limit or stop coolant flow in that channel, and its heat would be removed by coolant flowing in adjacent channels. Consequences of fuel channel blockage during normal operation will be accommodated by detection capabilities on primary coolant circulating activity and helium cleanup. If a fuel channel is blocked and failed fuel is suspected, the reactor will be shut down and the primary coolant cleaned by the Helium Purification System (HPS), described in Section 5.4. If a channel is blocked but no excess radioactivity is detected, then consequences do not exceed the acceptable bounds for circulating activity and no mitigation is necessary.

The following loss of coolant events are precluded by design:

- PLOFC followed by failure to shut down

  The RPS and RCSS are safety related and designed with sufficient reliability and redundancy to rely on their safety function to shut down the reactor.

- PLOFC followed by loss of passive RCCS function

  The passive functionality of the RCCS is safety related and designed with sufficient reliability and redundancy to credit its ability to remove decay heat, thereby ensuring a safe and stable state can be achieved and maintained.

13.1.5 Mishandling or Malfunction of Fuel

Various PIEs can result in a mishandling or malfunction of fuel. Fuel is handled between cycles as part of refueling operations by the Fuel Handling and Storage Systems (FHSS). Refueling must be performed with care because the TRISO layers and pellet matrix comprise the functional containment necessary for retention of radionuclides.

The limiting postulated event for mishandling or malfunction of fuel is the dropping of a spent fuel block while it is being lifted from the reactor core. Refueling occurs at or near atmospheric pressure, after the fuel has cooled down enough to allow safe handling {{ }}a(4) accesses the core via the control rod penetrations in the head of the RV. It latches to the handling socket of a graphite block, lifts it above the core, {{ }}a(4) In the limiting mishandling of fuel event, the fuel block is lifted from the second-to-bottom layer of the core and dropped from the highest point permissible {{ }}a(4) (above the upper layer of the core), falling from a significant height onto the fuel blocks still in the core. The fuel block drop results in the dropped and impacted graphite blocks breaking and releasing their fuel pellets. While unlikely due to their weight and strength, some of the pellets are assumed to be damaged as a result of the impact. The damage to the pellets results in a release of radionuclides into the RV. During refueling, {{ }}a(4) It is assumed that all radionuclides released from the damaged fuel into the RV gas volume can escape into the environment.

The analysis of the limiting mishandling of fuel event is covered in Section 13.2.5.

The limiting mishandling of fuel event is expected to bound other possible mishandling or malfunction of fuel events, including:

- Overheating of spent fuel blocks during handling

  Even with no strong, forced cooling during refueling, thermal stress in the fuel and graphite will remain very low because heat generation during refueling is limited to decay heat. Temperatures are sufficiently low that thermally driven diffusion of radionuclides from intact or damaged fuel is negligible.

- Breakage of a spent fuel block within the {{ }}a(4) during handling

  It is postulated that mishandling of spent fuel blocks by any system part of the FHSS could result in minor shocks or impacts, potentially resulting in damage to the graphite block. Events that occur while the fuel is handled by the FHSS are expected to be bounded by the limiting event because: (1) more material is at risk in the limiting event (two spent fuel blocks affected vs. one); and (2) the impact from the drop in the limiting event is much larger than that from any other possible event within the FHSS.

- <u>Operation of the reactor with incorrectly loaded fuel</u>

  All fuel blocks have the same fuel loading, but the burnable poisons are tailored to the general core region where the fuel block is intended to be loaded. Incorrect loading will be precluded by: (1) {{ }}[a(4)] and (2) administrative controls throughout the entire lifecycle of the fuel elements. Should the reactor be operated with an incorrectly loaded fuel element, the resulting power distribution could affect reactor performance, but it would be detected during startup testing and calibration procedures. If not detected during startup or at any point during normal operation, higher-than-normal power peaking would occur, potentially shortening the fuel cycle length. The increased power peaking would remain within the bounds of the fuel qualification envelope.

The following fuel malfunction and mishandling events are precluded by design:

- <u>Shine exposure during refueling</u>

  The design of the reactor and the FHSS provide radiation shielding for workers present during refueling operations. Refueling will be performed remotely and workers will oversee the process from a safe location. Shine events during refueling can therefore be excluded because they require multiple failures: (1) unlikely failure of shielding systems; and (2) failure of a worker to follow the refueling procedure, which will rely on access interlocks and be informed by substantial radiation monitoring.

- <u>Criticality events during refueling</u>

  No mishandling of fuel could result in a criticality event. The most reactive core configuration is that in which all fuel and reflector blocks are in their respective locations. Removal of even the top reflector block to begin refueling results in a decrease in reactivity. Although control rods are removed one at a time during refueling, the remaining control rods have sufficient worth to ensure subcriticality as fresh fuel is loaded.

13.1.6 Experiment Malfunction

The U. of I. research reactor will not include any in-core experiments. Therefore, experiment malfunction is not applicable to the reactor safety case.

Proper physical and administrative controls will ensure that ex-core experiments do not affect reactor safety and that any potential malfunction could not result in consequences that are more severe than those of the MHA. As described in Section 12.3 and consistent with Section 6.4 of ANSI/ANS 15.1-2007 (R2023) (Reference 13-10), the Technical Specifications require procedures for administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity. This is captured in the corresponding Limiting Conditions of Operations (3.8) reported in Table 14-1 of Section 14. An overview of the experiment review process that helps prevent failure and malfunction is provided in Section 10.3. A detailed description of the facility procedures, including the review, approval, and changes processes, will be provided with the Operating License Application, consistent with the requirements stated in 10 CFR 50.34(b)(6)(vi) (Reference 13-4) and 10 CFR 50.59 (Reference 13-11).

13.1.7 Loss of Normal Electrical Power

If normal electrical power is lost for any reason, all electricity-powered systems, including the TESS and active circulation of the RCCS, will be without power. The electrical power systems, described in Chapter 8, include uninterruptible power supplies (UPS) to enable the controlled shutdown of the reactor, the continuous operability of select systems, and the transition to diesel backup power. All safety-related SSCs are designed to be fail-safe without reliance on electrical power, and the electrical power systems do not perform any safety function.

The following will occur after normal electrical power is lost:

- Non-safety related systems will be de-energized and shut down. This includes the TESS and the active function of the RCCS.
- Immediately following the loss of normal electrical power, the UPS and Direct Current (DC) batteries provide non-safety related backup power to selected facility loads such as the RPS, Instrumentation & Control (I&C) systems, emergency lighting, critical communications, the fire alarm system, security system, and monitoring systems.
- Backed by the DC batteries, the RPS facilitates the safe shutdown of the reactor. The RPS responds to the loss of normal electrical power by tripping the reactor. The trip de-energizes the CRDUs, allowing the rods to insert under gravity, and de-energizes the helium circulator, stopping the flow of primary coolant. The resulting increase in reactor temperature further contributes to reactor shutdown via the negative temperature reactivity feedback mechanism. Importantly, both shutdown mechanisms can be performed passively to achieve and maintain a safe shutdown state without the need for the RPS trip signal if all electrical power is lost. Section 8.3.2.1 describes the design basis for the UPS and battery systems.
- Within a few seconds, backup diesel generators start and supply electrical power to select systems. Section 8.3 describes the design basis for the backup generators.

From this point forward, the reactor is in a PLOFC event condition with control rods inserted as a result of the RPS trip. The inserted control rods maintain shutdown, thereby achieving the safety function of reactivity control. Decay heat is passively transferred from the core to the RCCS, achieving the heat removal function. The RCCS is assumed to operate in its passive mode, which relies only on its safety-related components.

The loss of normal electrical power event is bounded by the PLOFC event described in Section 13.1.4. In a loss of normal electrical power, the RPS immediately trips the reactor by inserting the control rods and stopping the helium circulator, resulting in a rapid shutdown. The decrease in power from the RPS trip is faster than in the limiting PLOFC event described in Section 13.1.4. Therefore, the amount of heat that must be managed and would otherwise challenge the key figures of merit is greater for the limiting PLOFC event.

13.1.8 Internal and External Hazards

The research reactor is designed so that SSCs relied upon to perform safety functions are located below grade in the Citadel building. The Citadel building provides physical protection against external hazards, such as meteorological loads, external flooding, and seismic events. The Citadel also contains systems to mitigate the consequences of internal hazards such as internal fire and flood.

Chapter 3 discusses the design of the Citadel building and the means by which it protects those SSCs associated with Engineered Safety Features (ESFs) from internal and external events. Although internal and external hazards may be initiators for other events, their effects on the reactor are expected to be bounded by other events because the ESFs are physically protected.

SSCs that are not protected from internal or external hazards could be affected and potentially release their radioactive inventory during an event. The limiting event for this category is assumed to be a design basis earthquake event that causes a small-breach DLOFC and affects equipment that is not seismically qualified. Although the VS are seismically qualified, it is assumed that penetrations for non-safety related instrumentation lines could fail, resulting in potentially one or more small breaches in the pressure boundary.

The consequences of the small-breach DLOFC are expected to be bounded by those of the limiting DLOFC event described in Section 13.1.3. The primary driver for reduced consequences is a reduction in the dose consequence contribution of the liftoff of plated-out fission products. A slower depressurization results in lower helium velocities and lower shear forces on the surfaces in contact with the primary coolant, resulting in less liftoff.

Additionally, the design basis earthquake could result in the failure of all systems containing radioactive material that are not qualified to maintain structural integrity in a design basis earthquake. Radionuclides might form during operation in the following systems and components that are not seismically qualified:

- RCCS water, excluding the water in the safety-related standpipes within the reactor cavity
- TESS molten salt
- Heating, Ventilation, and Air Conditioning System filters

The amount of radioactive material able to accumulate in these systems is limited by design, such that the consequences of their potential failure remain below all acceptable limits. Details about the specific limits for each of those systems and the applicable inspection and maintenance programs will be detailed in the OLA.

The limiting hazard event is bounded by the MHA because: (a) the small-breach DLOFC is bounded by the larger DLOFC discussed in Section 13.1.4, which in turn is bounded by the MHA with respect to all applicable figures of merit; and (b) the material at risk in the aforementioned systems will be limited by their design and operation. Therefore, the combined dose from the small-breach DLOFC and the releases from the aforementioned systems will not result in a total dose that exceeds that of the MHA.

13.1.9 Mishandling or Malfunction of Equipment

Mishandling or malfunction of equipment can lead to radioactive releases from systems or components that contain radioactive material. The mishandled or malfunctioning equipment may itself contain radioactive material at risk or its failure could lead to releases from other systems. This section considers the failure of systems and components that could lead to radioactive releases not explicitly covered by the other sections. All scenarios are bounded in severity and consequences by the MHA, ensuring no equipment mishandling or malfunction event could result in unacceptable dose consequences.

13.1.9.1 Operator Error during Helium Purification System Maintenance

The HPS removes radionuclides from the primary coolant to keep its activity within acceptable limits. As a result, the HPS accumulates those radionuclides and needs to be regularly maintained. The HPS resides in the Citadel building and is seismically qualified to ensure no failure would occur during and following a design basis earthquake. However, because the various components require periodic maintenance to remove accumulated radionuclides, it is postulated that operator error could lead to the release of the complete accumulated inventory from any one of the HPS components.

To ensure that this event is bounded by the MHA, it is imposed that the amount of radioactive material at risk for release in each component of the HPS must be maintained below amounts that would result in exceeding the doses anticipated for the MHA. The maintenance procedures for the HPS and radioactivity level monitoring through the Unit Monitoring Systems will ensure that the amount of radioactive material at risk for release in the HPS is satisfying these requirements.

13.1.9.2 Earthquake

A seismic event is assumed to be the limiting equipment malfunction event for systems and components that are not seismically qualified. Additional discussion on the consequences of a design basis earthquake and how they are bounded by the MHA is provided in Section 13.1.8.

13.1.9.3 Mishandling of Control Rod during Refueling

Control rods must be handled as part of refueling procedure, putting them at risk for mishandling. Mishandling of a control rod would not lead to a criticality event because it would only happen with all other control rods known to be fully inserted, providing significant reactivity shutdown margin. However, it is postulated that the control rod could be damaged during handling. The primary radionuclide at risk is tritium, which is produced by neutron absorption in B-10 of the B_4C absorber material. During refueling, the reactor is at near-atmospheric pressure and the FHSS interfaces closely with the RV to form an effective seal.

If a control rod were to be damaged during handling due to equipment malfunction, it is possible that some of the accumulated tritium could be released. However, the consequences of mishandling a control rod are expected to be bounded by the consequences of mishandling a fuel block because the control rod contains much less radioactive material. The limiting postulated mishandling of fuel event is discussed in Section 13.1.5 and analyzed in Section 13.2.5.

13.2 ACCIDENT ANALYSIS AND DETERMINATION OF CONSEQUENCES

This section presents analyses for the limiting postulated events for each event category described in Section 13.1, including the MHA. The MHA is intended to postulate conditions leading to consequences that are bounding those resulting from other postulated events. The analysis of the MHA and resultant consequences is covered in Section 13.2.1.

Sections 13.2.2, 13.2.3, and 13.2.4 present the analyses for the limiting postulated insertion of excess reactivity, DLOFC, and PLOFC events, respectively. Conservative assumptions are used to define the initial state of the research reactor SSCs at the onset of the postulated events. Those assumptions ensure that the most limiting conditions are analyzed and quantified. These assumptions will be refined for the OLA once all Technical Specifications have been defined. The system conditions arising from these limiting postulated events are captured by figures of merit, described in Table 13-1. The acceptance criteria in Table 13-1 ensure that dose consequences that could arise from the limiting postulated events are bounded by those of the MHA. Technical Specification limits have been selected based on the accident analyses and ensure that reactor operation as well as the postulated events remain within the analyzed conditions.

Section 13.2.5 covers the analysis of dose consequences arising from the limiting postulated mishandling of fuel event.

<u>*Mitigation of Postulated Events*</u>

The analyses performed for all limiting postulated events assume that safety-related SSCs and ESFs are available to mitigate the consequences of those events. The behavior of those SSCs and ESFs during postulated events, as driven by their design bases, are presented in the following paragraphs. Those behaviors are applicable to any and all postulated events. The event-specific sequences are described in the sections dedicated to each limiting postulated event considered.

<u>*Reactor Reverse and Reactor Trip*</u>

The RCS can detect off-normal conditions and initiate a reactor reverse, which involves inserting the control rods through controlling the stepper motor, described in Section 4.2.3.4, and slowing or stopping the circulator. The result of a reactor reverse is an immediate reduction of reactor power and eventual shutdown of the reactor. The RCS and reactor reverse provide defense in depth to limit the progression of initiating events in a controlled manner that does not necessitate a reactor trip. The RCS is described in Section 7.3 and the reactor reverse action is discussed in Section 7.3.3. Because the RCS is not safety related, it is not credited with responding to off-normal conditions occurring during the limiting postulated events.

The RPS, described in Section 7.4, is credited with detecting off-normal conditions and sending a trip signal to shut down the reactor. Upon detecting conditions established as a trip setpoint, the RPS sends a trip signal, which cuts electrical power to: (1) the CRDUs part of the RCSS; and (2) the helium circulator. This causes the control rods to quickly insert under gravity and stops the helium circulator. The RPS detection and actuation capabilities are automatic and do not rely on operator action.

The passive shutdown function of the RCSS is safety related, as described in Section 4.2.3. The control rods have sufficient worth to achieve and maintain safe shutdown of the reactor, as discussed in Section 4.5.

The helium circulator is described in Section 5.2. When it stops, forced flow of primary coolant is lost, and core temperatures rise. The negative temperature reactivity feedback mechanism, discussed in Section 4.5, is one of the two guaranteed means of shutting down the reactor. It is independent from the insertion of control rods and entirely passive.

Passive Cooling by the RCCS

The design of the RCCS is described in Section 6.3. Normally, the RCCS operates in its active cooling mode. Because the subsystems and components that enable active cooling within the RCCS are non-safety related, this active cooling function is not credited as part of the safety analyses once a postulated event is initiated. Instead, during postulated events, the RCCS is assumed to solely be able to operate in its passive mode, which relies only on the safety-related portions of the RCCS. The passive heat transfer pathway from the reactor to the RCCS in this mode is described in Section 4.6.

Similar to other passively safe reactor designs, the research reactor relies on passive heat removal via the RCCS that does not credit any operator intervention for a minimum of 72 hours, as described in Section 6.3. Therefore, the safety analyses for each limiting postulated event must demonstrate that a safe state is reached and maintained for a minimum of 72 hours following the initiation of the postulated event.

After that time, it is assumed that the operator is able to take simple and unambiguous actions to intervene and further mitigate the event, such as replenishing the RCCS water inventory.

Functional Containment

The fuel is credited for its role in providing functional containment, which is ensured by the TRISO particles and the outer dense surface layer (ODSL) of the fuel pellet. The TRISO kernel and coating layers provide four barriers that together retain a large fraction of radionuclides. The ODSL is the outer fuel barrier and retains within the fuel pellet the radionuclides that might have diffused out of the TRISO particles. The fuel design bases are described in Section 4.2.1. The role of the fuel in the functional containment approach is described in Section 6.2.

Consequence Analysis Methods

Consequence analysis is performed with a suite of models developed to determine the distribution of radionuclides during normal operation and releases from fuel and non-fuel sources during postulated events. The model performs atmospheric dispersion for calculating dose consequences using the equations described in NUREG CR-6331 Rev. 1 (Reference 13-12) and implemented in ARCON. Bounding use of the ARCON equations follows the guidance provided in Regulatory Guide (RG) 1.249 (Reference 13-13). The models and assumptions are described in Appendix 13A.

Additionally, topical reports detailing these methodologies, their V&V, and their application to the consequence analysis of the U. of I. research reactor will be prepared and submitted prior to the OLA.

<u>*System Analysis Methods*</u>

System analysis is performed with the Flownex Simulation Environment (Reference 13-14), a thermal-hydraulic system code that solves the cross-sectional averaged, one-dimensional, conservation of mass, momentum, and energy equations using the finite volume method for steady state and transient calculations. Flownex provides an integrated point kinetics model and specialized models for representing HTGRs. A Flownex model of the research reactor has been developed to calculate mass flows, temperatures, and pressures in the reactor SSCs during postulated events. The model is described in Appendix 13B.

Flownex is developed under an NQA-1 quality management system at M-Tech Industrial (Pty) Ltd (M-Tech). Software verification and validation (V&V) for general capabilities of Flownex is performed by M-Tech. V&V documentation regarding Flownex and its use in the safety analysis of the U. of I. research reactor will be prepared and submitted prior to the OLA.

13.2.1 Maximum Hypothetical Accident

The MHA is a hypothetical postulated event scenario analyzed under conservative conditions, resulting in dose consequences that bound all other postulated events, including limiting postulated events. The MHA is a rapid and full depressurization of the primary coolant loop analyzed with conservative radionuclide transport assumptions to produce more severe consequences than those expected DLOFC or PLOFC events, or any other postulated event that results in a DLOFC or PLOFC end state. The dynamic response of the reactor core during the MHA is not explicitly modeled; instead, the temperatures during the MHA are defined conservatively to establish bounding dose consequences.

This section describes the assumptions used to define the MHA, the responses of the safety related SSCs credited to mitigate the event, and the conservative analysis of consequences.

13.2.1.1 Initiating Events

The initiating event for the MHA is an instantaneous breach in the reactor primary coolant pressure boundary. The MHA analysis does not include modeling of the dynamic response of the SSCs or change in reactor conditions such as temperatures, pressures, mass flows, and reactivity. Therefore, specific details pertaining to the initiating event are not pertinent to the analysis and bear no impact on the outcome.

13.2.1.2 Sequence of Events and Systems Operation

A breach in the reactor primary coolant pressure boundary causes system depressurization. The type of breach and rate of depressurization are not explicitly defined; assumptions related to depressurization rate that affect the radiological consequences are discussed in Section 13.2.1.3. Primary coolant exits the pressure boundary into the reactor and IHX cavities, inside the Citadel building. The pressure in those cavities is relieved via the depressurization pathway described as part of the Pressure Zone 2 components in Section 9.1.5.2. The pathway {{

}}[a(4)] exhausted to the environment.

The RPS is credited with detecting the low-pressure condition and initiating a reactor trip. The reactor trip and shutdown sequence are described in the introduction to Section 13.2. The shutdown of the reactor achieves the safety function to control reactivity.

The loss of primary coolant causes the fuel and core graphite to heat up, but their temperatures eventually decrease as heat is passively removed from the reactor by the RCCS, as described in the introduction to Section 13.2. The passive heat transfer from core to the RCCS results in decreasing fuel and reactor system temperatures, achieving the safety function of heat removal.

The system spends 72 hours in this safe, stable state, with consequence mitigation only from the ESFs, all of which are passive features.

After 72 hours, the operator is credited for being able to take simple actions to further mitigate the consequences of the event, such as refilling the RCCS with water. The operator's actions will ensure that the same safe and stable state is maintained, and that the fuel and graphite temperatures continuously decrease for the remainder of the MHA.

The depressurization caused by the reactor coolant pressure boundary breach results in a short-duration release of the radionuclides that are contained in the primary coolant at the time of the MHA initiation. {{

}}$^{a(4)}$ Radionuclides contained in the air initially present in the reactor and IHX cavities are also released. After depressurization is complete, radionuclides released from the fuel and graphite over time by temperature-driven processes are assumed to be continuously released to the environment despite the absence of a driving force. This additional release of radionuclides continues for 30 days, the assumed duration of the MHA. The fuel is credited for its role in providing functional containment and thereby hindering the diffusion of radionuclides, as described in the introduction to Section 13.2. No credit is given to the attenuation of released activity by deposition in the Citadel building.

13.2.1.3 Analysis of Effects and Consequences

This section describes the high-level inputs and assumptions used in the consequence analysis of the MHA. Additional details are provided in the description of the consequence analysis model in Appendix 13A.

13.2.1.3.1 Input Parameters and Assumptions

Prior to the MHA, the reactor is assumed to be operating normally at full power. Normal operating conditions are described in Section 4.5.

The radioactive material at risk for release during the MHA includes the following sources:

- Radionuclides contained in the fuel
- Radionuclides circulating in the primary coolant
- Condensable fission products that deposit around the primary coolant loop
- Tritium generated from impurities in the core graphite during normal operation
- Activated air in the reactor and IHX cavities

The distribution of radionuclides during normal operation prior to the MHA is determined conservatively to maximize the potential radiological release during the MHA. Appendix 13A describes the determination of the normal operating condition in more detail. The following assumptions are employed:

- As discussed in Reference 13-9, acceptance of TRISO particles from manufacturing is statistical and therefore a fraction of the TRISO particles might have defects. As a result, this fraction of TRISO particles will not retain radionuclides as effectively. The TRISO defect fractions are set to bounding values (Table 2.2 in Reference 13-9). Additionally, a fraction of the TRISO particles may experience in-service failures during normal operations. {{ }}$^{a(4)}$ For the purpose of conservatively quantifying the inventory of radionuclides contained in the primary coolant and deposited on surfaces within the pressure boundary, it is assumed that the failed fraction of TRISO particles is already failed from when they are first loaded in the core (as opposed to in-service failure). {{ }}$^{a(4)}$
- Similarly, it is statistically possible that a fraction of the fuel pellet ODSL might have defects from manufacturing, which may reduce their radionuclide retention capabilities. For the MHA, it is conservatively assumed that a {{ }}$^{a(4)}$ of fuel pellets have their ODSL entirely failed from when they are first loaded in the core (as opposed to in-service failure), effectively allowing radionuclides to entirely bypass diffusion through the ODSL. {{ }}$^{a(4)}$ It is assumed that no additional ODSL failure occurs during normal operation or during any postulated event including the MHA, {{ }}$^{a(4)}$
- The silicon carbide matrix within the fuel pellets, inside the ODSL and surrounding all the TRISO particles, is assumed to provide no resistance to radionuclide transport. In reality, radionuclides would have to diffuse through it, but this is not credited as a means of mitigating release, therefore maximizing radionuclide release rate from the fuel.
- All fuel in the core is assumed to be at a constant temperature of {{ }}$^{a(4)}$ for the duration of normal operations, which is greater than the maximum fuel temperature calculated in the analysis described in Section 4.5. In reality, the temperatures will vary throughout the cycle and only a fraction of the fuel will experience the peak temperature for a shorter duration.
- The radionuclide inventory in the fuel is determined at the end of an assumed 3-year cycle for the nominal average discharge burnup {{ }}$^{a(4)}$
- The tritium inventory in the fuel element graphite block is determined at the end of an assumed 3-year cycle, resulting in the maximum inventory.

- The permanent radial reflector blocks, described in Section 4.2.2, are modeled as containing the highest tritium inventory expected from normal operations. {{

  }}a(4)
- Tritium produced by activation of trace He-3 in the primary coolant is the largest contributor to the circulating activity in the primary loop, and its production rate is conservatively quantified by assuming no depletion of the He-3 source.
- The circulating activity in the primary loop is evaluated crediting nominal HPS functionality and minimum deposition around the primary loop. The design basis for the HPS is discussed in Section 5.4. The removal rate of radionuclides from the primary coolant via deposition is set to a minimum value to maximize circulating activity. Note that this assumption and the one pertaining to deposition on surfaces lead to some double counting of radionuclides, further contributing to the conservatism of the MHA analysis.
- The inventory of deposited radionuclides around the primary coolant loop is evaluated after an assumed 40 years of operation with a maximum deposition rate, informed by literature (Reference 13-16).
- The activity in the air contained in the reactor and IHX cavities is evaluated after an assumed 40 years of continuous operation assuming no depletion of the source isotopes, and assuming losses only via radionuclide decay. {{

  }}a(4)

Additional conservative assumptions are also used to evaluate the releases of the radionuclides during the MHA. The initial release during the MHA is caused by the depressurization of the primary coolant from the pressure boundary into the reactor and IHX cavities. The depressurization rate is not explicitly modeled but is informed by preliminary analysis of pipe breach initiating events. The initial release includes radionuclides contained in the primary coolant, in the air present in the reactor and IHX cavities, and lifted off from deposited radionuclides around the primary coolant loop. Primary coolant and air from the reactor and IHX cavities are assumed to be released in full along with their entire radionuclide inventories. {{

}}a(4)

Throughout the entire MHA duration, assumed to be 30 days, radionuclides are continuously released from the fuel and graphite via diffusion processes. The time-temperature profiles assumed for the fuel and graphite during the MHA are shown in Figure 13-1 and tabulated in Table 13-2. The time-temperature profiles do not directly show the expected change in reactor conditions because the MHA analysis does not include modeling of the dynamic response of the SSCs. Rather, the profiles were developed to bound the maximum time-temperature profiles of all limiting postulated events, thereby ensuring that the releases from fuel and graphite during the MHA are bounding. These temperatures are applied to all elements across the entire core, effectively assuming that all fuel and reflector elements experience temperatures above those that might arise as a result of core thermal hydraulic evolution. This is a very conservative assumption because temperature-driven release processes are exponential or quadratic with temperature. {{

}}a(4) Additionally, only a minimal fraction of the fuel and graphite may experience peak temperatures approaching values used in the MHA.

Figure 13-1 Assumed Time-Temperature Profiles used for the MHA

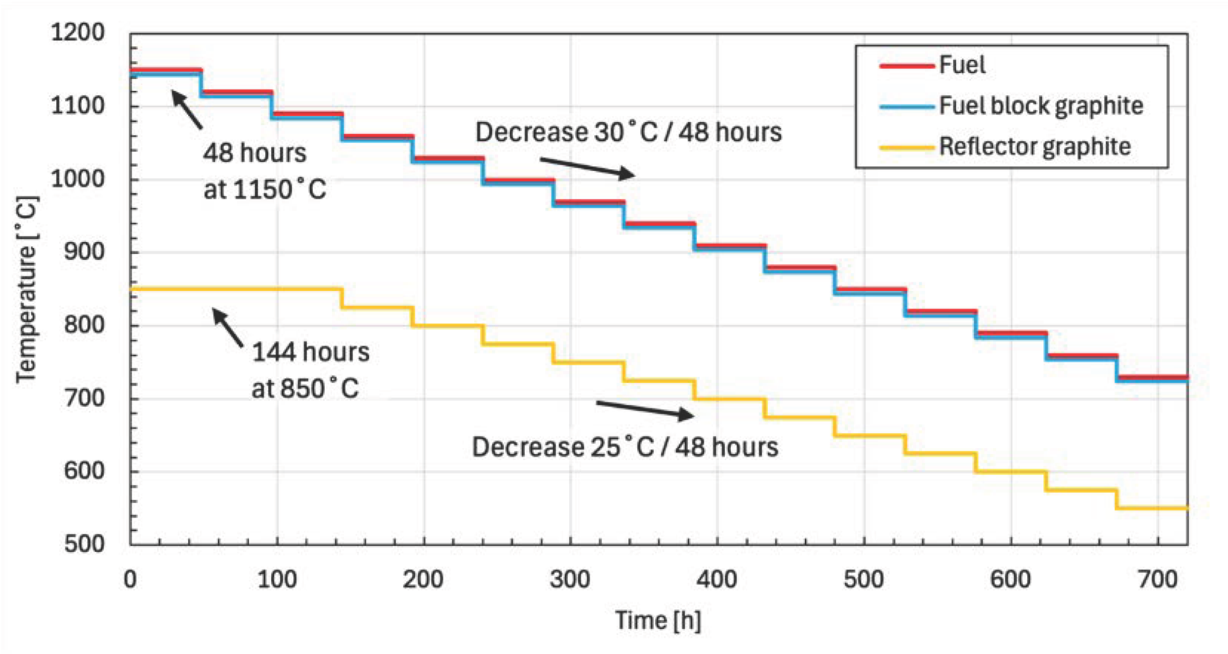

Table 13-2 Assumed Time-Temperature Values used for the MHA

Time		Fuel temperature [°C]	Graphite fuel block temperature [°C]	Time		Graphite reflector temperature [°C]
Start	End			Start	End	
0	48	1150	1150	0	144	850
48	720 (30 d)	Stepwise decrease: 1150-730 (30°C/48 h)	Stepwise decrease: 1100-650 (32°C/48 h)	144	720 (30 d)	Stepwise decrease: 850-550 (25°C/48 h)

Intact TRISO layers and pellet ODSLs act as diffusion barriers. International Atomic Energy Agency (IAEA) diffusion coefficients (Reference 13-17) are used for the TRISO particles. The particular set of IAEA diffusion coefficients used in the analysis are from the PARFUME code manual (Table 7-6 in Reference 13-18). The diffusion coefficients for TRISO SiC are used for the ODSL with a {{ }}[a(4)] safety factor applied directly to the coefficient values due to the use of different manufacturing techniques. No credit is taken for radionuclide retention by the fuel pellet matrix. Radionuclides escaping the TRISO particles join the fission products generated from dispersed uranium (discussed in Section 4.2.1.6.3), which is present inside the fuel pellets but outside the TRISO particles. Those radionuclides can then immediately begin diffusing through the fuel pellet ODSL because the holdup and retention offered by the SiC matrix is not credited.

It is assumed that every fuel pellet with a failed ODSL (assumed fraction {{ }}[a(4)]) releases those radionuclides that escape from TRISO particles or originate from the dispersed uranium. This is equivalent to assuming that all radionuclides contained inside a pellet, but outside of the TRISO particles, are instantaneously accumulating in the location of the ODSL failure site (e.g., crack). Statistically, this is an exceptionally conservative assumption.

Radionuclides, dominated by tritium, are generated from impurities contained in the graphite during irradiation at normal conditions and are subsequently released as a function of time-at-temperature.

Radionuclides released from the fuel and graphite during the entire MHA are assumed to be released directly and immediately to the environment. No credit is taken for time-based decay, precipitation, or deposition along the pathway between the core region and the environment. {{

}}[a(4)]

The reactor core spends 72 hours at elevated temperatures, with passive mitigation only from the ESFs. After that time, it is assumed that the operator is able to take simple and unambiguous actions to further mitigate the event, such as replenishing the RCCS water inventory.

The figures of merit for the MHA are the doses to the public and to workers. The dose to a member of the public for the duration of the accident is evaluated against the 1 rem (10 mSv) TEDE limit established in 10 CFR 50.34(a)(1)(i) (Reference 13-4). The worker dose is evaluated against the occupational TEDE limit of 5 rem (50 mSv) from 10 CFR Part 20.1201(a)(1)(i) (Reference 13-5). The dose conversion coefficients used are from FGR-11 (Reference 13-19) and FGR-12 (Reference 13-20).

The TEDE to a member of the public as a result of the MHA is evaluated assuming the exposed individual is 20 meters from the release point – the wall from the reactor building, well within the site boundary (described in Section 2.1), and remains there for the duration of the MHA. The calculation of TEDE includes exposures via inhalation, submersion, and groundshine. Inhalation and submersion result from the plume exposure. Groundshine is driven by exposure to the radionuclides deposited on the ground over the assumed 30-day duration of the MHA.

Atmospheric dispersion of the released radionuclides from the release point to a member of the public is evaluated with a model that draws on methods and guidance from NUREG CR-6331 Rev. 1 (Reference 13-12), RG-1.249 (Reference 13-13), and RG-1.194 (Reference 13-21). For conservatism, the model is applied in such a way to minimize dispersion, which in turn maximizes radionuclide concentration and the resulting dose to the individual. The model treats the release as originating at ground level to maximize near-field ground level concentrations. Vertical release velocity and plume buoyancy are not accounted for because they decrease near-field ground level air concentrations. Conservative meteorological conditions are applied rather than considering the site-specific meteorological data reported in Chapter 2. As discussed in Appendix 13A, the meteorological conditions are {{

}}[a(4)] the least-favorable X/Q for dose to an individual at the assumed distance.

The atmospheric dispersion model used is a point-source, straight-line Gaussian dispersion model. {{

}}[a(4)] This is conservatively applied over the entire period in a single direction, assuming that the individual is located in that exact direction, with constant worst-case weather conditions. Normal wind diffusion coefficients from RG-1.194 (Reference 13-21) are used with corrections for low wind speed and wake effects. A building area of 1e-2 m^2 is conservatively used as input such that the building wake effect is not credited.

The TEDE to a worker resulting from the MHA is calculated by assuming all radionuclides are released instantly and uniformly dispersed throughout the Maintenance Enclosure volume (described in Section 3.5), where a worker is exposed. The volume of primary coolant in the primary loop is {{ }}$^{a(4)}$ and the volume of the Maintenance Enclosure is {{ }}$^{a(4)}$ resulting in a dilution factor of {{ }}$^{a(4)}$ No credit is taken for reductions in activity concentration that might occur due to building leakage, HVAC filtration or radionuclide decay. The worker is assumed to remain in the Maintenance Enclosure for 10 minutes before leaving. It is anticipated that this exposure time is conservative and that realistically, the worker will be able to secure and clear the Maintenance Enclosure in much less time. The exposure time will be updated in the OLA when more detailed operational procedures have been developed. The calculation of TEDE to a worker includes exposures via inhalation and submersion.

13.2.1.3.2 Results

The MHA dose consequence to a member of the public is 0.57 rem (5.7 mSv). The calculated dose reflects significant conservatism in both the enabling assumptions and in the underlying methodology, ensuring the safety of the public and the environment. The maximum MHA dose is determined at 20 meters from the reactor building wall, whereas the shortest distance between the site boundary and the reactor building is over 25 meters. The MHA dose at the site boundary therefore satisfies the limit that the TEDE shall be less than 1 rem beyond the site boundary.

The MHA dose consequence to a worker who spends 10 minutes in the Maintenance Enclosure with the full inventory of released radionuclides is 1.0 rem (9.8 mSv). The MHA dose readily meets the occupational TEDE limit of 5 rem.

13.2.2 Insertion of Excess Reactivity

The limiting insertion of excess reactivity event is a spurious withdrawal of the highest worth control element at a hypothetically high rate when the core is operating at full power. This section describes the analysis of the limiting postulated insertion of excess reactivity event, including the sequence of events, system response, key assumptions, and the figures of merit used to ensure any possible consequences are bounded by the MHA consequences.

13.2.2.1 Initiating Events

During normal full power operation, the highest worth control rod is spuriously withdrawn due to a malfunction of the RCS. For conservatism, the control rod is assumed to be withdrawn at two times the maximum speed allowed by the CRDU motor and gears, which are described in Section 4.2.3.

This analysis is typically performed at the point in the operating cycle when the control rod worth is highest, usually when the control rods are the most inserted. {{

}}$^{a(4)}$ Additionally, the fuel has less time to dissipate the added heat, resulting in larger temperature gradients and thermal stress in the fuel.

13.2.2.2 Sequence of Events and Systems Operation

The control rod withdrawal results in a positive reactivity insertion, causing an increase of the reactor power.

Normally, the RCS detects the off-normal condition and initiates a reactor reverse to shut down the reactor. The details of the RCS are discussed in Section 7.3. However, because the RCS is not a safety-related SSC, it is not credited with detecting and shutting down the reactor, nor mitigating the postulated event.

Reactor power increases as the control rod is withdrawn. Fuel temperatures increase, resulting in a negative reactivity feedback. This negative reactivity feedback lags slightly behind the positive reactivity resulting from the control rod being withdrawn. As a result, the reactor power continues increasing until the RPS detects reactor overpower and trips the reactor. This occurs when the reactor power level reaches {{ }}$^{a(4)}$ of the nominal power. The RPS overpower trip reflects the corresponding Limiting Safety System Setting described in Section 14.1. Fuel temperatures at the time of the trip are higher than during normal operation.

The RPS is credited with detecting the overpower condition and initiating a reactor trip. The reactor trip and shutdown sequence are described in the introduction to Section 13.2. The withdrawn control rod remains stuck in the fully withdrawn position. The reactor is shut down by inserting the remaining control rods and powering down the circulator, thereby achieving the safety function to control reactivity.

After the trip, core temperatures rise briefly due to the loss of forced cooling. Decay heat is passively transferred from the fuel to the core graphite, then to the RV, and finally to the RCCS, as described in the introduction to Section 13.2. The establishment of passive heat transfer from the core to RCCS results in decreasing fuel and reactor system temperatures, achieving the safety function of heat removal. During the event, the RCCS is operating in its passive mode.

The research reactor spends 72 hours in this safe, stable state, with only the ESFs credited to mitigate consequences. All ESFs are passive features, as discussed in Chapter 6.

13.2.2.3 Analysis of Effects and Consequences

The limiting postulated insertion of excess reactivity event is analyzed using the Flownex model described generally in the introduction of Section 13.2 and detailed in Appendix 13B. This present section describes the event-specific inputs and assumptions, figures of merit, and resulting system evolution.

13.2.2.3.1 Input Parameters and Assumptions

Prior to the limiting postulated insertion of excess reactivity event, the reactor is assumed to be operating normally at full power. Normal operating conditions are described in Section 4.5. The reactor conditions prior to the initiation of the event are conservatively assumed to challenge system response.

- The power level is assumed to be {{ }}a(4) of the nominal power at the time of the initiating event. {{ }}a(4) The overpower condition increases the amount of decay heat that needs to be managed.
- The event occurs early in the operating cycle when the control rods are inserted to positions from which the withdrawal results in the highest positive reactivity insertion rate. The worth of the control rods is taken to be the maximum value that can occur throughout the operating cycle such as to bound the positive reactivity insertion associated with the control rod withdrawal.
- The reactivity coefficients are the least negative that occur throughout the cycle.

The event is initiated when the highest-worth control rod starts being withdrawn from its initial assumed position. The rate of withdrawal of the control rod movement is limited by design as described in Section 4.2.3, but it is conservatively assumed that the control rod is withdrawn at twice the maximum possible rate.

Reactor power increases while the control rod is withdrawn. The RPS trips the reactor when the power lever reaches the {{ }}a(4) overpower setpoint. The control rod withdrawal continues even after the trip, until it is fully withdrawn from the core. Conservatively, it is assumed that the control rod remains stuck in its fully withdrawn position for the remainder of the event; the other six control rods have sufficient worth to achieve and maintain safe shutdown of the reactor, as discussed in Section 4.5.

The RCCS active function is assumed to be unavailable upon initiation of the event. The RCCS passive mode, which relies only on safety related portions of the RCCS, remains functional.

The figures of merit for this event are listed below. If all figures of merit satisfy their corresponding acceptance criteria listed in Table 13-1, the consequences of this event are ensured to be bounded by the MHA.

- Fuel time-temperature profile is bounded by the MHA conditions.
- Fuel block and reflector graphite temperature are bounded by the MHA conditions.
- RV temperature remains within ASME BPVC Section III, Div 5, subsection HB (Reference 13-7) for accident conditions.
- Core barrel temperature remains within the limit set in NRC Regulatory Guide (RG) 1.87 Rev. 2 (Reference 13-8) {{ }}a(4) covered by ASME limit BPVC Section III, Div 5, subsection HG (Reference 13-7) for accident conditions.
- Peak radial fuel pellet temperature gradient is below the threshold {{ }}a(4)

Fuel, graphite, reflector, core barrel, and RV temperatures are calculated by Flownex, using the model described in Appendix 13B. The fuel, fuel block graphite, and reflector graphite thermal conductivities are conservatively set to their minimum expected values, accounting for temperature and irradiation effects.

The radial fuel pellet temperature gradient is evaluated as the difference between the maximum pellet temperature and minimum pellet surface temperature at a given axial and radial position in the core. The increase in the peak radial fuel pellet temperature gradient caused by the control rod withdrawal is evaluated analytically, using the change in peak power density caused by the control rod withdrawal. The peak power density resulting from the event is calculated by combining the following peaking factors:

- {{ }}a(4) corresponding to the core-wide power increase due to the control rod withdrawal (i.e., the {{ }}a(4) overpower that results in a reactor trip);
- {{ }}a(4) corresponding to the maximum pellet relative power density at any point in the cycle, obtained from the analysis described in Section 4.5;
- {{ }}a(4) a hypothetical factor accounting for a potential change of the power distribution in the core due to the control rod withdrawal.

The resulting maximum power peaking factor for the considered limiting postulated event is {{ }}a(4) The peak radial fuel pellet temperature gradient calculation is conservative because it overestimates the temperature gradient in the fuel pellet. The primary contributor to the conservatism is the assumption that the highest power pellet is coincident in space and time with the local peaking resulting from the control rod withdrawal.

A safe and stable state is established when:

- The core is subcritical and long-term reactivity control is assured.
- Decay heat is being removed and long-term cooling is assured; figure of merit temperatures are steadily decreasing.

13.2.2.3.2 Results

The spurious withdrawal of the highest worth control rod from its critical position results in the insertion of positive reactivity, causing an increase in reactor power. Figure 13-2 shows the fission and decay power reactivity changes, and fuel and graphite block temperatures for the first {{ }}a(4) after initiation of the event. The power increases until the {{ }}a(4) overpower limit is reached after {{ }}a(4) at which point the RPS trips the reactor, inserting the remaining control rods via gravity and cutting power to the helium circulator. The reactivity insertion from the withdrawn control rod outpaces the temperature reactivity feedback of the fuel until the remaining control rods are inserted. The insertion of the remaining control rods quickly brings the reactor subcritical and maintains the reactor in a shutdown state.

Figure 13-2 Power, Reactivity Changes, and Temperatures for the Early Portion of the Limiting Postulated Insertion of Excess Reactivity Event

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}}a(4)

The time-dependent figure of merit temperatures are shown in Figure 13-3. Due to the reactor power increasing and coolant flow being unchanged prior to the reactor trip, the peak and average fuel temperatures increase slightly. After the RPS trips the reactor and cuts power to the helium circulator, the peak and average fuel temperatures briefly decrease below normal operating temperatures as residual heat redistributed to the graphite but later increase as heat is retained in the core. Eventually, fuel temperatures decrease as the core transfers decay heat to the RCCS, which is functioning in its passive mode. Because reactivity is controlled and decay heat is removed, a safe state is achieved. None of the figure of merit temperatures exceed those defined for the MHA, ensuring that any consequences from the limiting postulated insertion of excess reactivity insertion event are bounded by those for of the MHA.

Figure 13-3 Figure of Merit Temperatures for the Limiting Postulated Insertion of Excess Reactivity Event

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}}$^{a(4)}$

The peak fuel radial temperature gradient arising from the overpower is calculated to be {{ }}$^{a(4)}$ below the {{ }}$^{a(4)}$ preliminary acceptance criteria in Table 13-1. {{

}}$^{a(4)}$ This result demonstrates that no additional fuel should fail as a result of the limiting postulated insertion of excess reactivity event.

13.2.3 Depressurized Loss of Forced Cooling with Air Ingress

The limiting DLOFC postulated event is an instantaneous breach in the reactor coolant pressure boundary within the IHX cavity, resulting in primary loop depressurization and air ingress. This section describes the analysis of this event, including the sequence of events, system response, key assumptions, and the figures of merit that ensure any possible consequences are bounded by the MHA potential consequences.

13.2.3.1 Initiating Events

The initiating event for the limiting DLOFC event is the instantaneous breach in the largest pipe connected to the pressure boundary. The pipe connects the VS to the HPS and Primary Coolant Make-Up System.

13.2.3.2 Sequence of Events and Systems Operation

The breach in the primary coolant pressure boundary causes primary coolant to depressurize as it exits the pressure boundary into the IHX cavity. The increasing pressure in the IHX cavity is relieved via the depressurization pathway, described in Section 3.5.3, allowing cavity air and primary coolant to be released to the environment along with the radionuclides they carry.

The loss of primary coolant results in a steady decline of the pressure within the pressure boundary. Because it is not safety related, the RCS is not credited with detection or mitigation of the event. Instead, the event progresses until the RPS low-pressure setpoint is reached. The RPS is credited with detecting the low-pressure condition and initiating a reactor trip. The reactor trip and shutdown sequence are described in the introduction to Section 13.2. The shutdown of the reactor achieves the safety function to control reactivity.

Depressurization continues until the pressure within the pressure boundary and that of the cavity are balanced, which is near atmospheric pressure due to the depressurization pathway of the Citadel building. After the depressurization and reactor trip, core temperatures rise briefly due to the loss of forced cooling. Decay heat is passively transferred from the fuel to the core graphite, then to the RV, and finally to the RCCS, as described in the introduction to Section 13.2. The passive heat transfer from the core to RCCS results in decreasing fuel and reactor system temperatures, achieving the safety function of heat removal.

After depressurization, it is possible that air could enter the reactor core and oxidize materials therein. The fuel pellets are made of SiC, which is not vulnerable to oxidation at the temperatures occurring during any postulated event. At lower temperatures, oxidation of SiC slowly forms a passive, protective oxide layer that prevents further oxidation. Graphite is vulnerable to oxidation above temperatures of about 600°C, and therefore graphite components in the core are at risk oxidation in case of air ingress. In case of the limiting postulated DLOFC event, the potential for air ingress is quite limited. {{

}}[a(4)] Detailed air ingress and oxidation assessments will be provided in the OLA.

The research reactor spends 72 hours in this safe, stable state, with only the ESFs credited to mitigate consequences. All ESFs are passive features, as discussed in Chapter 6.

13.2.3.3 Analysis of Effects and Consequences

The limiting postulated DLOFC event is analyzed using the Flownex model described generally in the introduction of Section 13.2 and detailed in Appendix 13B. This present section describes the event-specific inputs and assumptions, figures of merit, and resulting system evolution.

13.2.3.3.1 Input Parameters and Assumptions

Prior to the limiting postulated DLOFC event, the reactor is assumed to be operating normally at full power. Normal operating conditions are described in Section 4.5. The reactor conditions prior to the initiation of the event are assumed such as to define a limiting scenario with the greatest potential consequences.

- The power level is assumed to be {{ }}[a(4)] at the time of the initiating event. {{ }}[a(4)] The overpower condition increases the amount of decay heat that needs to be managed.
- The event occurs at the point in cycle when the power peaking occurs near the bottom of the core so that the highest possible peak fuel temperature is occurring.
- The reactivity coefficients are the least negative that occur throughout the cycle.
- Control rods are inserted to their critical positions, consistent with the reactor state assumed above that yields the most limiting power distribution.

The limiting postulated event is initiated when a full circumferential breach in the reactor primary coolant pressure boundary is instantaneously formed at the connection between the VS and the HPS and Primary Coolant Make-Up System, described in Sections 5.4 and 5.5, respectively. In response to the decrease in primary coolant pressure, the RPS trips the reactor at the low-pressure setpoint. The RPS trip causes the control rods to insert via gravity and the helium circulator to stop. Conservatively, it is assumed that the highest-worth control rod remains stuck in its initial position.

The RCCS active function is assumed to be unavailable upon initiation of the event. The RCCS passive mode, which relies only on safety-related portions of the RCCS, remains functional.

The figures of merit for this event are listed below. If all figures of merit are below their corresponding thresholds, the consequences of this event are ensured to be bounded by those of the MHA.

- Fuel time-temperature is bounded by the MHA conditions.
- Fuel block and reflector graphite temperatures are bounded by the MHA conditions.
- RV temperature remains within ASME BPVC Section III, Div 5, subsection HB (Reference 13-7) limit for accident conditions.
- Core barrel temperature remains within ASME BPVC Section III, Div 5, subsection HG (Reference 13-7) limit for accident conditions.
- Core barrel temperature remains within the limit set in NRC Regulatory Guide (RG) 1.87 Rev. 2 (Reference 13-8) {{ }}[a(4)] covered by ASME limit BPVC Section III, Div. 5, Subsection HG (Reference 13-7) for accident conditions.
- Peak radial fuel pellet temperature gradient is below the threshold {{ }}[a(4)]

Fuel, graphite, reflector, core barrel, and RV temperatures are calculated by Flownex, using the model described in Appendix 13B. The fuel, fuel block graphite, and reflector graphite thermal conductivities are conservatively set to their minimum expected values, accounting for temperature and irradiation effects.

A safe and stable state is established when:

- The core is subcritical and long-term reactivity control is assured.
- Decay heat is being removed and long-term cooling is assured; figure of merit temperatures are steadily decreasing.

13.2.3.3.2 Results

The breach in the reactor coolant pressure boundary results in a rapid loss of primary coolant. Depressurization occurs over a period of about {{ }}[a(4)] The primary coolant pressure, reactor power, reactivity changes, and fuel and graphite block temperatures are shown in Figure 13-4 for the first {{ }}[a(4)] after initiation of the event. The RPS detects the low-pressure condition and trips the reactor, inserting the control rods by gravity and stopping the helium circulator. This quickly brings the reactor into a strongly sub-critical state, which representative shutdown margins discussed in Section 4.5.2.5.

Figure 13-4 Primary Coolant Pressure, Reactor Power, Reactivity Changes, and Temperatures for the Early Portion of the Limiting Postulated DLOFC Event

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}}[a(4)]

The decrease in power throughout the limiting postulated DLOFC event means that the peak radial fuel pellet temperature gradient occurs at the initiation of the event, during normal operation. Therefore, the gradient remains well below the threshold {{ }}a(4)

The time-dependent figure of merit temperatures are shown in Figure 13-5. None of the figure of merit temperatures exceed those defined for the MHA, ensuring that any consequences from the limiting postulated DLOFC event are bounded by those for the MHA.

Figure 13-5 Figure of Merit Temperatures for the Limiting Postulated DLOFC Event

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}}a(4)

As a result of the reactivity control and decay heat removal, a safe state is shown to be established. The results shown above indicate that none of the figure of merit temperatures exceed those defined for the MHA, ensuring that any consequences from the limiting postulated DLOFC event are bounded by those for the MHA.

13.2.4 Pressurized Loss of Forced Cooling

The limiting postulated PLOFC event is a spurious trip of the helium circulator and subsequent shutdown of the reactor on negative temperature reactivity feedback alone. This section describes the analysis of the limiting postulated PLOFC event, including the sequence of events, system response, key assumptions, and the figures of merit that ensure potential consequences are bounded by those of the MHA.

13.2.4.1 Initiating Events

A spurious helium circulator trip stops the forced circulation of primary coolant. Normally, the RCS detects the loss of forced cooling by the reduction in helium flow rate and initiates a reactor reverse to shut down the reactor. Because the RCS is not safety related, it is not credited with detecting the loss of the primary coolant forced circulation.

13.2.4.2 Sequence of Events and Systems Operation

The spurious circulator trip results in a rapid loss of primary coolant flow through the reactor core. The RCS is not credited for detecting or mitigating the off-normal conditions.

The loss of forced cooling causes an increase in fuel and graphite temperatures. This results in a decrease of the reactor power due to the negative temperature reactivity feedback, leading to reactor shutdown. In response to the reactor shutdown, the RPS enables the “neutron flux in shutdown mode” trip, described in Section 7.4.3.

Following the reactor shutdown, decay heat is passively transferred from the fuel to the graphite, then to the RV, and finally to the RCCS, as described in the introduction to Section 13.2. The RCCS is assumed to be functioning in its passive mode, which relies only on safety-related portions of the RCCS. Because the RCCS active cooling mode relies on non safety-related portions of the RCCS, its function is not credited during the event. The passive heat transfer from the reactor core to RCCS eventually results in decreasing fuel and reactor system temperatures, achieving the safety function of heat removal.

The decay of xenon result in a progressive increase in core reactivity over time. Eventually, this increase in reactivity leads to an increase in fission power. The RPS detects the increase in fission power and trips the reactor via the “neutron flux in shutdown mode” trip when the measured neutron flux exceeds a very small fraction {{ }}$^{a(4)}$ of the full power value. The reactor trip and shutdown sequence are described in the introduction to Section 13.2. The shutdown of the reactor achieves the safety function to control reactivity.

The research reactor spends 72 hours in this safe, stable state, with only the ESFs credited to mitigate consequences. All ESFs are passive features, as discussed in Chapter 6.

13.2.4.3 Analysis of Effects and Consequences

The limiting postulated PLOFC event is analyzed using the Flownex model described generally in the introduction of Section 13.2 and detailed in Appendix 13B. This present section describes the event-specific inputs and assumptions, figures of merit, and resulting system evolution.

13.2.4.3.1 Input Parameters and Assumptions

Prior to the limiting postulated PLOFC event, the reactor is assumed to be operating normally at full power. Normal operating conditions are described in Section 4.5. The reactor conditions prior to the initiation of the event are assumed such as to define a limiting scenario with the greatest potential consequences.

- The power level is assumed to be {{ }}$^{a(4)}$ at the time of the initiating event. {{ }}$^{a(4)}$ The overpower condition increases the amount of decay heat that needs to be managed.
- The event occurs at the point in cycle when the power peaking occurs near the bottom of the core so the highest possible peak fuel temperatures is occurring.
- The reactivity coefficients are the least negative that occur throughout the cycle.

- Control rods are initially inserted to their critical positions, consistent with the reactor state assumed above that yields the most limiting power distribution.

The event is initiated when the helium circulator spuriously trips and shuts down. The forced flow of molten salt on the secondary side of the IHX is also assumed to stop upon initiation of the event. This is assumed to conservatively limit the amount of heat that can be dissipated through the secondary coolant.

After the initial shutdown of the reactor by the inherent temperature negative reactivity feedback, and subsequent cooling of the reactor core, fission power slowly re-increases. This is detected by the RPS, which trips the reactor. It is conservatively assumed that the highest-worth control rod remains stuck in its initial position.

The RCCS active function is assumed to be unavailable upon initiation of the event. The RCCS passive mode, which relies only on safety related portions of the RCCS, remains functional.

The key figures of merit are listed below:

- Fuel time-temperature is bounded by the MHA conditions.
- Fuel block and reflector graphite temperatures bounded by the MHA conditions.
- RV temperature remains within ASME BPVC Section III, Div 5, subsection HB (Reference 13-7) limit for accident conditions.
- Core barrel temperature remains within the limit set in NRC Regulatory Guide (RG) 1.87 Rev. 2 (Reference 13-8) {{ }}a(4) covered by ASME limit BPVC Section III, Div. 5, Subsection HG (Reference 13-7) for accident conditions.
- Peak radial fuel pellet temperature gradient is below the threshold {{ }}a(4)

Fuel, graphite, reflector, core barrel, and RV temperatures are calculated by Flownex, using the model described in Appendix 13B. The fuel, fuel block graphite, and reflector graphite thermal conductivities are conservatively set to their minimum expected values, accounting for temperature and irradiation effects.

A safe and stable state is established when:

- The core is subcritical and long-term reactivity control is assured.
- Decay heat is being removed and long-term cooling is assured; figure of merit temperatures are steadily decreasing.

13.2.4.3.2 Results

Figure 13-6 shows the power, reactivity changes, and fuel and graphite block temperatures for the first {{ }}a(4) after initiation of the limiting postulated PLOFC event. Initial heat up of the fuel and graphite due to the loss of forced cooling results in a negative reactivity insertion, making the reactor subcritical and quickly shutting it down. After shutdown, the xenon-135 concentration increases as its precursors decay, contributing additional negative reactivity.

Figure 13-7 shows the evolution of reactivity contributions over the first {{ }}[a(4)] of the transient. Although the fuel and fuel block graphite moderator provide negative reactivity contributions, xenon-135 decays with a half-life of 9.14 hours, eventually contributing to an increase of the reactivity. {{ }}[a(4)] the RPS detects it and trips the reactor, causing a rapid insertion of the control rods.

Figure 13-6 Power, Reactivity Changes, and Temperatures for the Early Portion of the Limiting Postulated PLOFC Event

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Figure 13-7 Reactivity Contributions for the PLOFC Event

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Figure 13-8 Figure of Merit Temperatures for the Limiting Postulated PLOFC Event

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}}a(4)

Temperatures rise as a result of the loss of forced cooling, but heat transfer to the RCCS, which is operating in its passive mode, eventually results in a steady decrease in temperatures. As a result of the reactivity control and decay heat removal, a safe state is shown to be established. Figure 13-8 shows that none of the figure of merit temperatures exceed those defined for the MHA, ensuring that any consequences from the limiting postulated PLOFC event are bounded by those for the MHA.

The decrease in power throughout the limiting postulated PLOFC event means that the peak radial fuel pellet temperature gradient occurs at the initiation of the event, during normal operation. {{

}}a(4) Therefore, the gradient remains well below the threshold {{
}}a(4)

13.2.5 Mishandling or Malfunction of Fuel

The limiting postulated mishandling of fuel event is the involuntary dropping of an irradiated fuel element while it is being removed from the reactor during refueling operations. The analysis does not include modeling of the dynamic response of the SCCs or change in reactor conditions such as temperatures, pressures, mass flows, and reactivity. Instead, the analysis focuses on the evaluation of consequences under conservative conditions and assumptions. This section describes the analysis of the limiting postulated mishandling of fuel event, including the sequence of events, key assumptions, and determination of dose.

13.2.5.1 Initiating Events

While removing the bottom layer of the irradiated fuel elements during refueling, a malfunction {{ }}[a(4)] causes it to involuntarily let go of the spent fuel element it is handling, dropping it from the maximum elevation permissible {{ }}[a(4)] The dropped fuel element falls down to the bottom layer of fuel elements where it impacts another spent fuel element. The drop compromises the structural integrity of the two affected graphite blocks as well as the ODSL of some of the fuel pellets contained in those blocks. This results in a release of radionuclides into the RV.

13.2.5.2 Sequence of Events and Systems Operation

The dropped fuel element falls from the maximum height {{ }}[a(4)] to the bottom layer of spent fuel elements. The main impact will be between the two graphite blocks. The energy from this impact will primarily be dissipated into the graphite through fracture. As a result, the graphite blocks will no longer restrain the fuel pellets. The fracture of the graphite blocks releases some of their accumulated tritium into the RV.

Some of the impact energy will be transferred to the fuel pellets contained within the two graphite blocks. The fuel pellets are much stronger than the graphite but, hypothetically, some could be damaged during the impact due to kinetic energy and interactions with other fuel pellets. The fuel damage is expected to be superficial, such as gouging, chipping, or cracks at the fuel pellet edges or at the mating interface between two pellets. It is assumed that {{ }}[a(4)] of fuel pellets in both blocks will experience a failure of the ODSL as a result of the impact, equivalent to failure of more than two pellets in every channel of the affected blocks.

Conservatively, no credit is given to radionuclide retention or holdup in the fuel pellet matrix. This is equivalent to assuming that the ODSL failures result in the disintegration of the entire ODSL and SiC pellet matrix. Failure of those pellets results in an instantaneous release of the entire inventory of radionuclides that have accumulated in the pellet matrix. The released radionuclides are instantaneously dispersed into the RV.

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}}[a(4)] It is assumed that no additional TRISO failure occurs due to the impact.

Radionuclides are assumed to escape from the RV through the interface {{ }}[a(4)] over a period of a few minutes. This assumption is conservative since refueling operations occur near atmospheric pressure and there is no or very little pressure differential to drive radionuclides from the RV to the environment. To reach the environment, radionuclides would have to leak through the seal between {{ }}[a(4)] and the RV cover head.

13.2.5.3 Analysis of Effects and Consequences

13.2.5.3.1 Input Parameters and Assumptions

This section describes the inputs and assumptions used in the consequence analysis of the limiting postulated mishandling of fuel event. Details about specific parameters and assumptions are provided in the description of the consequence analysis model in Appendix 13A.

The radioactive material at risk for release during the limiting postulated mishandling of fuel event includes the following sources:

- Radionuclides contained in the fuel
- Tritium present in the fuel element graphite block during normal operation

Prior to the refueling operation during which the limiting fuel mishandling event occurs, it is assumed that the reactor has been operated for a full 3-year cycle and the fuel has reached its highest burnup. The distribution of radionuclides for the limiting postulated mishandling of fuel event is determined conservatively to bound the potential radionuclide release:

- The radionuclide inventory in the fuel is determined at the end of an assumed 3-year cycle, resulting in the maximum fission product inventory. The two impacted fuel elements are assumed to have the maximum burnup occurring in the core at the end of a cycle {{ }}[a(4)]
- The initial condition assumes all pellets in the affected blocks are intact and have experienced no diffusion through the ODSL during normal operation. As a result, the fuel pellet contains the highest possible accumulated radionuclide inventory from {{ }}[a(4)] of burnup. Radionuclides that have diffused from the TRISO particles into the pellet matrix during normal operations are included.
- {{ }}[a(4)] of fuel cooling and radionuclide decay after shutdown are assumed before refueling starts.
- The tritium inventory in the graphite is determined at the end of an assumed 3-year cycle, resulting in the highest possible inventory.

As a result of the impact of the dropped fuel element, radionuclides in the SiC matrix of the affected fuel pellets ({{ }}[a(4)] of the fuel in both blocks) are immediately released into the RV. The sources of these radionuclides include in-service diffusion from intact TRISO, releases from failed TRISO, and radionuclides generated from dispersed uranium. This conservative assumption implies coincident locations between the fuel pellet failures and the radionuclides in the pellet matrix, with no credit given to the matrix or remaining ODSL.

Additionally, a fraction of the tritium contained in the graphite block is immediately released into the RV. The released fraction of tritium is assumed to be {{ }}a(4) which is conservative because: {{

}}a(4)

The radionuclide inventory is assumed to be released from the RV over a period of {{ }}a(4), without holdup by any confinement mechanisms. This assumption is conservative because: (1) there is no significant driving force to transport radionuclides from the RV to the environment; and (2) no credit is taken for retention by the RV, {{ }}a(4) or seal between them, despite the effectiveness of such technologies evidenced by operational experience of past HTGRs.

The figures of merit for the limiting postulated mishandling of fuel event are the doses to the public and to workers. These must be bounded by the public and worker TEDEs resulting from the MHA. The applied dose conversion coefficients are from FGR-11 (Reference 13-19) and FGR-12 (Reference 13-20).

The dose to a member of the public is evaluated assuming the exposed individual is 20 meters from the release point, well within the site boundary (described in Section 2.1), and remains there for the duration of the event, which is assumed to be 30 days. Radionuclide release is evaluated using a point-source, straight-line Gaussian dispersion model with release occurring at ground level. The model and assumed conditions are the same as that used to evaluate releases for the MHA, described in Section 13.2.1.3. The calculation of TEDE includes exposures via inhalation, submersion, and groundshine. Inhalation and submersion result from the plume exposure. Groundshine is driven by exposure time to the radionuclides deposited on the ground, assumed to be 30 days.

The dose to a worker at the facility is calculated using the same assumptions and methodology described in Section 13.2.1.3.

13.2.5.3.2 Results

The conservatively calculated dose consequence from the limiting mishandling of fuel postulated event to a member of the public is 0.06 rem (0.60 mSv). The dose meets the site TEDE limit of 1 rem (10 mSv) from 10 CFR Part 50.34(a)(1)(i) (Reference 13-4) and is bounded by the MHA public dose consequence.

The dose consequence from the limiting postulated fuel mishandling event to a worker who spends 10 minutes in the Maintenance Enclosure with the full inventory of released radionuclides is 0.08 rem (0.83 mSv). The dose meets the occupational TEDE limit from 10 CFR Part 20.1201(a)(1)(i) (Reference 13-5) and is bounded by the MHA worker dose consequence.

13.2.6 Experiment Malfunction

All experiment malfunction scenarios are bounded in severity and consequences by the MHA. As discussed in Section 13.1.6, the U. of I. research reactor will not include any in-core experiments. Additionally, proper physical and administrative controls will ensure that ex-core experiments do not affect reactor safety. The Technical Specifications require procedures for administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity. This is captured in the corresponding Limiting Conditions of Operations (3.8) reported in Table 14-1 of Chapter 14. Therefore, dedicated system response and consequence analysis for ex-core experiment malfunctions is not included.

13.2.7 Loss of Normal Electrical Power

As discussed in Section 13.1.7, the loss of normal electrical power is bounded by the limiting postulated PLOFC event described in Section 13.1.4 and analyzed in Section 13.2.4. Therefore, dedicated system response and consequence analysis for loss of normal electric power scenario are not included.

13.2.8 Internal and External Hazards

The limiting postulated event in this category is assumed to be a seismic event that causes a small-breach DLOFC and results in radionuclide releases from all SSCs that are not seismically qualified. Section 13.1.8 describes how this event is bounded by the MHA because: (a) the small-breach DLOFC is bounded by the larger DLOFC discussed in Section 13.1.4, which in turn is bounded by the MHA with respect to all applicable figures of merit; and (b) the material at risk in systems that are not seismically qualified are limited such that the additional dose caused by their release does not result in a total dose that exceeds that of the MHA. Therefore, dedicated system response and consequence analysis for external hazards are not included.

13.2.9 Mishandling or Malfunction of Equipment

All mishandling or malfunction of equipment scenarios are bounded in severity and consequences by the MHA. As discussed in Section 13.1.9, this is evidenced by: (1) limiting the accumulation of radionuclides in systems that are not seismically qualified; and (2) qualitative assessment of the severity of equipment failure postulated events against others postulated events already analyzed as part of the safety analysis. Therefore, dedicated system response and consequence analysis dedicated analysis for equipment failure scenarios are not included.

13.3 SUMMARY AND CONCLUSIONS

This chapter demonstrates the safety of the U. of I. research reactor to avoid undue risk to the health and safety of the public, workers, and the environment. It also provides evidence that the design meets the regulatory requirements of the NRC. The potential consequences of the MHA demonstrate the acceptability of the design with respect to regulatory dose criteria for the public and workers.

The MHA is a hypothetical, conservatively defined DLOFC scenario. The MHA bounding conditions are characterized by temperatures, because high temperatures challenge the functional containment provided by the fuel elements and TRISO particles. Dose consequences are analyzed under limiting conditions with conservative radionuclide transport assumptions and modeling approaches. The MHA potential consequences to the public and workers are shown to meet all applicable criteria. This demonstrates the effectiveness of the functional containment approach and the performance of the ESFs to ensure safety and mitigate potential consequences.

A comprehensive list of postulated events is described and grouped into the event categories prescribed by NUREG-1537. For each event category, a limiting postulated event is identified for quantitative analysis. The analysis results demonstrate the resilience of the research reactor and the performance of its ESFs which are all passive. Each limiting event is evaluated against consistent acceptance criteria to ensure the potential radiological consequences are bounded by those of the MHA.

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APPENDIX 13A DOSE CONSEQUENCE APPENDIX

This appendix provides a summary of the methodology and assumptions used to analyze the preliminary dose consequences resulting from the MHA. A refined assessment will be provided as part of the Operating License Application or prior to submittal in an appropriate regulatory submission (e.g., topical report).

Evaluating dose consequences involves determining the sources of radioactivity, radionuclide inventories within the sources and associated release mechanisms and rates based on the conditions assumed for the MHA. The total released radionuclide content from each source is determined by multiplying its accumulated inventory by its release fraction, which is evaluated to reflect the mechanisms and extent of release from a given source. The primary source of concern in any nuclear facility is the uranium-bearing nuclear fuel, which produces a large quantity of radionuclides from fission and activation of the heavy metal content. Non-fuel sources can also be activated to produce radionuclides. More detailed discussion on the different radionuclide production sources is provided in Section 11.1.1.

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Figure 13A-3 {{ }}a(4)
(reproduced from Reference 13A-14)

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APPENDIX 13B SYSTEM ANALYSIS METHODS

This section describes the methods used to analyze the response of the U. of I. research reactor to the limiting postulated insertion of excess reactivity, depressurized loss of forced cooling (DLOFC), and pressurized loss of forced cooling (PLOFC) events.

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USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 14 - TECHNICAL SPECIFICATIONS

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
LCOs	Limiting Condition for Operations
LSSS	Limiting Safety System Setting
RCCS	Reactor Cavity Cooling System
SARRDL	specified acceptable radionuclide release design limits
SL	Safety Limit
SR	Surveillance Requirements
SSC	System, Structure and Component
TESS	Thermal Energy Storage System

CHAPTER 14 TECHNICAL SPECIFICATIONS

14.1 INTRODUCTION

The format and content of the technical specifications are consistent with the guidance provided in American National Standards Institute ANSI/ANS 15.1 (Reference 14-1) and NUREG-1537 (Reference 14-2), and include:

- Definitions
- Safety Limits and Limiting Safety System Settings
- Limiting Conditions for Operations
- Surveillance Requirements
- Design Features
- Administrative Controls

In accordance with 10 CFR 50.34(a)(5), the variables and conditions that are expected to be subject to technical specification control for the research reactor facility are provided in Table 14-1. These variables and conditions are informed by the preliminary safety analyses described in Chapter 13 of this report.

The final technical specifications and parameter limits will be submitted with the Operating License Application, consistent with 10 CFR 50.34(b)(6)(vi) and will address the requirements in 10 CFR 50.36. Note that in the research reactor neither the reactor pressure boundary nor the Citadel building serve a radionuclide retention function under postulated events. Fission product retention is provided by the functional containment described in Section 6.2. Therefore, the language in 10 CFR 50.36 (c)(2)(ii), “significant abnormal degradation of the reactor coolant pressure boundary,” is not applicable and will be replaced by “significant abnormal degradation of the functional containment.” Per the University of Illinois Urbana-Champaign (U. of I.) topical report on “Applicability of Nuclear Regulatory Commission Regulations” (Reference 14-3), the reactor pressure boundary is not credited as part of the functional containment but will perform a defense in depth role in preventing radionuclide release and therefore helium leakage detection instrumentation will be retained within technical specifications to meet the intent of Criterion 1 of 10 CFR 50.36 (c)(2)(ii).

Table 14-1 Proposed Variables and Conditions for Technical Specifications

Section	Section Name	Specification	Basis
2.0	Safety Limits (SLs) and Limiting Safety System Settings (LSSSs) SLs are those limits on process variables that are necessary to reasonably protect the integrity of certain physical barriers that are credited to preclude a potential uncontrolled release of radioactivity. LSSSs are settings for automatic protective devices related to those variables having significant safety functions. These settings ensure that automatic protective action will correct the abnormal situation before a SL is exceeded. This table consists of the proposed subjects of SLs and LSSSs. These are provided below.		
2.1	SL	The fuel temperatures shall not exceed an upper bound under any conditions of operation.	The maximum fuel temperature SL is established to ensure fuel integrity based on temperatures assumed in the safety analysis.
	SL	The reactor vessel temperatures shall not exceed an upper bound temperature under any condition of operation.	The maximum reactor vessel temperature SL is the maximum temperature that can be permitted with confidence that vessel integrity will be maintained.
2.2	LSSS	Positive reactor period shall not go below a lower bound period during startup operations.	Limiting reactor period will prevent excessive positive reactivity in the core during conditions where power level and temperature changes are low.
	LSSS	The high-power trip function shall not exceed an upper bound limit as specified in the safety analysis.	Limiting the upper bound limit will ensure that the reactor will trip prior to challenging a SL.
	LSSS	The reactor outlet temperature shall not exceed an upper bound temperature under any condition of operation.	Limiting the reactor outlet temperature will ensure that the SLs are not exceeded and that the reactor will trip prior to reaching a SL.

Table 14-1 Proposed Variables and Conditions for Technical Specifications (Continued)

Section	Section Name	Specification	Basis
3.0	Limiting Conditions for Operations (LCOs) LCOs are derived from the safety analysis and are implemented administratively or by control and monitoring systems to ensure safe operation of the facility. The LCOs are the lowest functional capability or performance level required for safe operation of the facility. The proposed subjects of LCOs are provided below.		
3.1	Reactor Core Parameters	Core configuration excess reactivity shall remain below a maximum acceptable value.	The objective is to ensure that adequate shutdown margin can be maintained, and the consequences of a reactivity insertion event remain within the design basis.
		Shutdown margin shall remain above a minimum acceptable level.	The objective is to ensure that the shutdown margin is sufficient to maintain the reactor shutdown under all core conditions.
		Reactor power shall not exceed the licensed reactor power level.	The objective is to limit the maximum operating power to ensure that the SLs will not be exceeded.
		Reactor fuel burnup shall not exceed the fuel qualification limit.	The objective is to ensure that fuel burnup is limited to the level demonstrated by the fuel qualification to ensure fuel reliability.
		Reactivity coefficients are within limits over the allowable range of operation.	The objective is to ensure that reactivity coefficients during plant operations are within design basis to limit the severity of a reactivity transient, and to ensure the shutdown response by temperature feedback occurs within bounding values assumed in the safety analyses.
3.2	Reactor Control and Safety Systems	Reactor protection system is operable.	The objective is to specify the requirement to have an operable reactor protection system to ensure that the SLs will not be exceeded.
		Controls rods are operable.	The objective is to ensure that a sufficient number of control rods can be inserted within the time specified in the safety analyses to ensure the ability to shut down the reactor in manner consistent with the design basis.

Table 14-1 Proposed Variables and Conditions for Technical Specifications (Continued)

Section	Section Name	Specification	Basis
3.3	Coolant Systems	Primary coolant radioactivity remains within limits.	The objective is to ensure the concentration of radionuclides in the coolant remains consistent with safety analyses and that the functional containment of the fuel has not degraded beyond the allowable limit.
		Primary coolant purity remains within limits.	The objective is to ensure the concentration of coolant impurities that can degrade core components remains within limits specified in the design basis.
		The RCCS initial thermal state required for the passive cooling function is maintained within limits.	The objective is to keep the RCCS in a state that guarantees the passive decay heat removal function is available, ensuring that SLs are not exceeded.
		Molten salt coolant radioactivity remains within limits.	The objective is to ensure the concentration of radionuclides in the molten salt of the intermediate loop remains consistent with safety analyses and that the specified acceptable radionuclide release design limits (SARRDLs) are not exceeded.
		Reactor helium leakage detection instrumentation is maintained operable.	The objective is to ensure that a significant abnormal degradation of the reactor helium pressure boundary can be detected and indicated in the main control room.
		The RCCS water radioactivity remains within limits.	The objective is to ensure the concentration of radionuclides in the RCCS remains consistent with safety analyses and that the specified SARRDLs are not exceeded.
		Primary coolant pressure remains within limits.	The objective is to ensure that the primary pressure is maintained so that operational transients will not result in exceeding the design pressure of the reactor helium pressure boundary.
		The Helium Pressure Relief System is maintained operable.	The objective is to ensure that in case of postulated events, rising helium pressure can be mitigated before it starts challenging the design pressure of the reactor helium pressure boundary.
3.4	Confinement	N/A	N/A
3.5	Ventilation Systems	Ventilation systems pressure zones are maintained within design basis requirements during reactor operations.	The objective is to provide a dedicated monitored pathway for effluent releases during normal operations.

Table 14-1 Proposed Variables and Conditions for Technical Specifications (Continued)

Section	Section Name	Specification	Basis
3.6	Emergency Power	N/A	N/A
3.7	Radiation Monitoring Systems and Effluents	Radiation monitoring system is designed to be available during normal operating conditions as well as during postulated events.	The objective is to ensure that radioactivity in the facility effluents is measured against applicable limits.
3.8	Experiments	Experiments conducted shall adhere to reactivity impact limits and design and material limitations.	The objective is to prevent damage to the reactor SSCs or excessive release of radioactive materials due to experiments or experiment failure.
3.9	Facility Specific LCOs	The Citadel building concrete wall temperature remains within limits.	The objective is to ensure the structural integrity of the Citadel building to provide structural support to the reactor core, RCCS, and other safety related structures, systems, and components (SSCs).
4.0	Surveillance Requirements (SRs) SRs relating to test, calibration, or inspection, to assure the necessary quality of systems and components is maintained, that facility operation will be within SLs, and that LCOs will be met, will be provided in the Operating License Application.		
5.0	Design Features Design features include those features of the facility, such as materials of construction and geometric arrangements, which if altered or modified could have a significant effect on safety. Design features will be provided in the Operating License Application.		
6.0	Administrative Controls Administrative controls are the programmatic provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting, necessary to ensure operation of the facility in a safe manner. Administrative controls will be provided in the Operating License Application.		

14.2 OPERATING MODES

The operational modes for the research reactor are summarized in Table 14-2. Each operational mode is defined in terms of combinations of core reactivity, reactor power, and reactor pressure boundary status. These modes are described individually in the following subsections.

MODE 1: Power Operations

The reactor is critical and operating above {{ }}a(4) rated thermal power. The reactor is operating at its normal operating temperatures and pressure. In this mode the reactor is being cooled by forced circulation of the primary coolant and transferring energy to the Thermal Energy Storage System (TESS), described in Chapter 5. Reactor power is controlled {{ }}a(4) using a combination of helium circulator speed control, molten salt flow and control rod movement. The reactor is transitioned to MODE 1 from MODE 2 during power ascension when the reactor {{ }}a(4) MODE 1 can be exited as part of a controlled power reduction or shutdown to MODE 2 when reactor power goes below {{ }}a(4) or it can transition directly to MODE 3 because of a reactor trip initiated by the reactor protection system.

MODE 2: Startup Operations

In this mode, control rods are being withdrawn, and the reactor is typically transitioning between sub-critical and critical. In rare cases, the reactor may transition from MODE 1 to MODE 2 and then back to MODE 1, remaining critical throughout the transition. In MODE 2, the reactor temperature is between ambient conditions and normal operating temperature. Reactor pressure is between the minimum operating pressure and normal operating pressure. Reactor power will be between the source range of the neutron flux detectors and {{ }}a(4) rated thermal power.

In this startup mode, the control rods are withdrawn from their fully inserted position to a critical position that depends on fuel burnup history, core temperatures, and other reactivity impacts. The reactor transitions to MODE 2 from MODE 3 during a normal startup when control rods are withdrawn to raise k-eff to ≥0.99. Once the reactor is critical with a stable positive period, reactor power is allowed to continue to rise until the reactor reaches the point of adding heat, beyond which nuclear heat is used to reach normal operating conditions. In MODE 2, the helium circulator is typically at minimum speed and power is controlled using the control rods. Once the primary coolant temperature is high enough for TESS operation, the TESS is placed in service. MODE 2 is exited to MODE 1 when reactor power is raised above {{ }}a(4) during power ascension

To reach MODE 2 during a reduction in power or shutdown, the reactor is transitioned from MODE 1 when control rods are inserted from the critical position at {{ }}a(4) rated thermal power towards their fully inserted position, and reactor power is insufficient to maintain normal operating temperatures and pressure. The TESS will typically be in service until coolant temperature decreases below the minimum allowed for TESS operation. MODE 2 is exited to MODE 3 during normal shutdown or due to a reactor trip when k-eff is lowered to less than 0.99.

MODE 3: Shutdown

The reactor is sub-critical in this mode. At least 6 of the 7 control rods are fully inserted to ensure that k-eff remains less than 0.99 under all conditions. In this mode the reactor is at any temperature or pressure condition between ambient and operating conditions. The Reactor Cavity Cooling System (RCCS) removes decay heat and maintains reactor core, reactor cavity, and Citadel building concrete temperatures within acceptable limits. If coolant temperature is above the minimum needed for TESS operations, the TESS may also be used to remove decay heat and transfer heat from the primary coolant. MODE 3 is entered from MODE 2 during a controlled shutdown or from MODE 1 or MODE 2 due to a reactor trip. MODE 3 is entered from MODE 4 once the reactor pressure boundary is intact and can perform its pressure-retaining function.

MODE 4: Refueling

The reactor is subcritical in this mode. The reactor pressure boundary is open to allow refueling or in-vessel maintenance activities. Normally, the refueling equipment interfaces with the pressure boundary without exposing the opening directly. In this mode, k-eff is maintained below 0.99 using a combination of control rod insertions and removal of fuel and/or moderator elements. The RCCS removes decay heat and maintains core, vessel, and Citadel building temperatures within acceptable limits to perform refueling or maintenance. MODE 4 is entered from MODE 3 when the reactor pressure boundary is opened for maintenance or refueling and not relied upon to perform its pressure-retaining function.

Table 14-2 Operating MODES for Technical Specifications

Operating Mode		**Reactivity Condition (k-eff)**	**Core Thermal Power (a)**	**Primary Coolant Temperature (d)**
MODE 1	Power Operations	≥0.99	{{ }}[a(4)]	N/A
MODE 2	Startup Operations	≥0.99	{{ }}[a(4)]	N/A
MODE 3 (b)	Shutdown	<0.99	N/A	N/A
MODE 4 (c)	Refueling	<0.99	N/A	N/A

Notes:

(a) excludes decay heat

(b) reactor pressure vessel sealed

(c) reactor pressure vessel open

(d) primary coolant temperature is not a determining factor in defining the operating modes, but will be constrained for each mode through the LCOs and SLs relating to temperature of the fuel and vessel

14.3 REFERENCES

14-1 American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1-2007 (R2023), "The Development of Technical Specifications for Research Reactors."

14-2 U.S. Nuclear Regulatory Commission (NRC) NUREG-1537, "Guideline for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors."

14-3 U. of I. Topical Report IMRDD-MMR-22-04-A, Release 04, "University of Illinois Urbana-Champaign High-Temperature Gas-cooled Research Reactor: Applicability of Nuclear Regulatory Commission Regulations," dated July 30, 2024 (ADAMS Accession Number ML24208A143).

USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 15 - FINANCIAL QUALIFICATIONS

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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LIST OF ACRONYMS AND ABBREVIATIONS

Term	Description
DOE	Department of Energy
FOCD	Foreign Ownership Control or Domination
MMR	Micro Modular Reactor
NRC	Nuclear Regulatory Commission
NWPA	Nuclear Waste Policy Act

CHAPTER 15 FINANCIAL QUALIFICATIONS

This chapter provides the financial information which establishes that the University of Illinois (U. of I.) is financially qualified to own, construct, operate, and decommission the KRONOS™ Micro Modular Reactor (MMR), a Class 104(c) non-power research reactor. U. of I.'s financial information is provided in accordance with 10 CFR 50.33(d)(3), 10 CFR 50.33(f), and the implementing regulations regarding the Price-Anderson Act contained in 10 CFR 140. This information is consistent with the guidance provided in NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors."

15.1 FINANCIAL ABILITY TO CONSTRUCT A NON-POWER REACTOR

Pursuant to 10 CFR Part 50.33(f)(1), and in accordance with Appendix C to Part 50 and the guidance provided in NUREG-1537, Part 1, Section 15.1, "Financial Ability to Construct a Non-Power Reactor," the following information and appendices describe the financial resources and commitments of U. of I. to construct the proposed advanced non-power research reactor. The information is limited in scope to the financial qualifications associated with reactor construction and demonstrates that adequate funds are available to complete the project without reliance on future reactor-generated revenue.

U. of I. is a public, land grant university established in 1867, with a long-standing mission in education, research and public service. As an established institution with a stable financial base, U. of I. possesses the financial strength necessary to support capital-intensive research infrastructure projects, including the construction of a non-power research reactor. The proposed reactor will be used solely for research, education, training, and related activities and will not be operated for commercial power production.

The construction scope addressed by this financial qualification includes reactor facility structures, reactor and auxiliary systems, instrumentation and control systems, safety related systems, site preparation, and associated licensing and project management costs, as applicable. Table 1 provides cost estimations for other recent reactor applications.

The total estimated cost of the research reactor construction, with and without contingency, is provided in Appendix 15A.

As described in Section 1.7, U. of I. intends to enter into a contract with the Department of Energy (DOE) that commits DOE to collect and manage U. of I. high-level waste and/or spent nuclear fuel resulting from research reactor operations to comply with the Nuclear Waste Policy Act (NWPA) of 1982. The October 6, 2025 letter from DOE to U. of I. regarding active and good faith negotiations for a disposal contract is included as Appendix 15B.

Table 15-1 Order-of-Magnitude Cost Estimates of Recent Reactor Applications

Reactor System	Overnight Cost [$M]	Year of Cost Estimate	Inflation Factor[a]	Thermal Power Level [MWth]	2026 Overnight Cost Estimate [$M/MWth]
UT TRIGA	5.8[b]	1992	2.33	1	13.5
HFIR	12.0[c]	1960	11.06	85	132.7
Fort Saint Vrain	274.1[d]	1974	6.64	842	1820

[a] based on Consumer Price Index.

[b] Wehring, Bernard W. *Final Report: University Reactor Sharing Program (Project Period 9/30/92 to 9/29/94)*. Nuclear Engineering Teaching Laboratory, The University of Texas at Austin. January 1995.

[c] Cole, T. E. *High Flux Isotope Reactor: A General Description*. Oak Ridge National Laboratory, operated by Union Carbide Corporation for the U.S. Atomic Energy Commission. March 1960.

[d] Energy Information Administration. *Nuclear Power Plant Construction Activity 1986*. Office of Coal, Nuclear, Electric and Alternate Fuels, U.S. Department of Energy. July 1987.
Report No. DOE/EIA-0473(86).

U. of I, has identified and secured adequate funding to complete construction of the advanced research reactor through established agreements with our technology partner NANO Nuclear Energy Inc. (referred to hereafter as “Nano”). Two agreements have been executed and are provided as appendices:

1. ***Memorandum of Understanding Between The Board of Trustees of the University of Illinois and NANO Nuclear Energy Inc.***, executed December 2025, establishes the intentions of the university and company to deploy the company's KRONOS MMR on campus as an advanced research reactor. Key elements of the agreement related to the reactor construction are that the agreement, 1) establishes the framework for engagement between organizations, 2) identifies Nano. as the lead organization for financing the project, 3) identifies the University of Illinois as the lead organization for engagement with state and federal agencies for additional avenues to offset project costs, and 4) establishes a joint project delivery steering committee to provide oversite through the construction phase, including representatives from both organizations. No funding is committed by this agreement. This memorandum of understanding is included as Appendix 15C.

2. ***Sponsored Research Agreement Amendment No. 2 Between The Board of Trustees of the University of Illinois and NANO Nuclear Energy Inc.***, acquired by Nano on October 2024, has committed funding of this project from Nano to U. of I. through February of 2027 to cover the costs incurred by the university for 1) Licensing, 2) Workforce development, 3) design and safety analysis support, and 4) stakeholder engagement and communication. The Sponsored Research Agreement, Amendment No. 02, is included as Appendix 15D.

Given the nature of the collaboration with Nano established in these agreements and their expressed commitment to finance the project, a financial qualification letter from Nano providing details demonstrating their ability to finance reactor construction is included as Appendix 15A. Also included within the letter is a discussion on initial fuel procurement and core loading.

U. of I., through its established partnership with Nano, has demonstrated the financial ability to meet these costs. The construction costs represent a manageable portion of the University's overall capital budget and do not place undue financial strain on the institution. No portion of the construction funding is contingent upon revenue generated by reactor operations, user fees, or external service income.

Construction expenditures will be managed in accordance with established institutional financial controls and capital project management procedures, including: formal capital project approval and authorization processes, dedicated project accounting and cost tracking, procurement and contracting controls, periodic financial reporting to senior administration and governing bodies, and independent or state-level financial audits, as applicable. These controls will ensure that construction funds remain available throughout the project duration and are expended in a manner consistent with approved budgets.

Based on the information provided above and the Nano financial letter that is included as an enclosure to the Cover Letter of the Construction Permit Application, U. of I. satisfies the financial qualification guidance set forth in NUREG-1537, Part 1, Section 15.1, “Financial Ability to Construct a Non-Power Reactor.” Adequate financing strategies have been identified, are currently being developed, and are well within the capability of the project team.

15.2 FINANCIAL ABILITY TO OPERATE A NON-POWER REACTOR

U. of I. has reasonable assurance of obtaining the necessary funds to cover facility operation costs for the period of the Operating License. Operational cost estimates and financial resources will be provided to the U.S. Nuclear Regulatory Commission (NRC) in the Operating License Application, per 10 CFR 50.33(f)(2).

15.3 FINANCIAL ABILITY TO DECOMMISSION THE FACILITY

U. of I. has reasonable assurance that funds will be available to decommission the U. of I. research reactor facility in accordance with 10 CFR 50.33(k)(1). This information is to be submitted to the NRC in accordance with 10 CFR 50.75(d)(1) as part of the application for an Operating License.

15.4 FOREIGN OWNERSHIP, CONTROL, OR DOMINATION (FOCD)

U. of I. is the applicant for the Construction Permit and subsequent Operating License for the research reactor. U. of I. is an established, nonprofit educational institution with a principal place of business in Urbana, IL, and is not acting as an agent or representative of another person in filing the application. Within the limits of authority fixed by the Illinois constitution and laws, The Board of Trustees exercises final authority over the University and is the governing body of U. of I. across the three campus locations in Urbana-Champaign, Chicago, and Springfield, Illinois. The University of Illinois Board of Trustees consists of 13 members, 11 who have official votes. Nine are appointed by the Governor and approved by the Senate of the State of Illinois for terms of six years, and three student trustees (one from each university campus) are elected by referenda at their university campus for one-year terms. All members of the board are US citizens; therefore, ownership and control are not dominated by foreign entities or individuals. The principal officers of U. of I. include the President, Chancellor, Provost, and Senior Vice Chancellor for Research & Innovation. All of these officers are U.S. citizens.

15.5 NUCLEAR INSURANCE AND INDEMNITY

U. of I. is a nonprofit educational institution applicant under 10 CFR 140.71, and as such, is not required to provide nuclear liability insurance. The NRC will indemnify U. of I. for any claims arising from a nuclear incident under the Price-Anderson Act, Section 170 of the Atomic Energy Act of 1954, as Amended, and in accordance with the provisions of its indemnity agreement pursuant to 10 CFR 140.95, Appendix E, above $250,000 up to $500 million. Because U. of I. is not requesting to possess Special Nuclear Material as part of its Construction Permit Application, U. of I. will request the indemnity agreement as part of the Operating License Application. Per 10 CFR 50.54(w), U. of I. is not required to purchase property insurance. If such arrangements are made, however, the details will also be provided as part of the application for an Operating License.

APPENDIX 15A CONFIRMATION OF FINANCIAL QUALIFICATIONS FOR UNIVERSITY OF ILLINOIS RESEARCH REACTOR CONSTRUCTION

NANO Nuclear Energy Inc.

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March 31, 2026

Docket No.: 99902094

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

Subject: Confirmation of Financial Qualifications for University of Illinois Research Reactor Construction (Docket No. 99902094)

NANO Nuclear Energy Inc. ("NANO Nuclear") submits this letter to the U.S. Nuclear Regulatory Commission (NRC) in support of the Construction Permit application filed by the University of Illinois ("U. of I." or "the Applicant").

As the technology vendor and primary commercial partner for the deployment of the KRONOS™ Micro Modular Reactor, NANO Nuclear confirms its role to lead the financing of project costs associated with this deployment, pursuant to the guidance in NUREG-1537, Part 1, Chapter 15.

Strategic Partnership and Commitment

NANO Nuclear's financial commitment is formalized through two key agreements executed with U. of I., which collectively ensures that financing this project is well within the capability of the project team and aligns with the objectives of all stakeholders:

- Sponsored Research Agreement (Executed October 2024): This agreement has already committed funding from NANO Nuclear to cover initial licensing, workforce development, and design analysis costs through February 2027.
- Memorandum of Understanding (Executed December 2025): This agreement explicitly designates NANO Nuclear as the lead organization for financing project costs, including project development, site preparation, and construction. Furthermore, it establishes a collaborative framework where U. of I. leads engagement with state and federal agencies for additional funding avenues, ensuring a robust and diversified capital strategy.

Financial Capability of the Project Team

The Applicant has estimated the total cost for the project development, licensing, procurement, construction, and commissioning to be approximately {{ }}[a(4)]. The project team is positioned to raise this amount through multiple avenues, primarily led by NANO Nuclear's commercial strength and supplemented by institutional partnerships.

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Primary Avenue: NANO Nuclear Capital Resources

As the lead financier, NANO Nuclear possesses many options to ensure project delivery including, but not limited to, strong liquidity and access to capital markets:

- Liquid Assets (Possession of Funds): As of the close of the fiscal quarter ending December 31, 2025, NANO Nuclear maintains a cash and cash equivalents position of approximately $578 million [1]. This liquidity, bolstered by a $379 million private placement in October 2025, {{ }}a(4).
- Access to Capital Markets: Since our Initial Public Offering in May 2024, the company has raised over $600 million. With a market capitalization exceeding $1.2 billion, the company retains significant capacity for additional equity financing via private placements, registered direct overnight offerings or, At-The-Market (ATM) offerings.
- Further, the SEC has approved a universal shelf registration for $900 million [2], including a $400 million at-the-market (ATM) equity program, providing the company with the flexibility to raise additional capital on an opportunistic basis which can allocated to the company's projects. Debt Capacity: With a minimal current debt load (approximately $1.93 million), NANO Nuclear retains nearly its full debt capacity to issue green bonds or secure commercial credit facilities to support infrastructure capitalization.

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Secondary Avenues: Government & Institutional Engagement

Complementing NANO Nuclear's direct capital, the MOU outlines that U. of I. will lead engagement with external agencies to offset project costs and secure fuel services:

- Department of Energy (DOE): Pursuit of federal funding and fuel support programs established for university-based research reactors.
    - {{ }}[a(4)] Fuel costs have been identified with the support of those partners, and NANO Nuclear is prepared to support these costs if procurement through the DOE programs was to fail.
- State of Illinois: Engagement with state programs to support shared priorities in clean energy and workforce development.

Conclusion

The project team's multi-faceted approach to financing—led by NANO Nuclear's substantial cash reserves of approximately $578 million and supported by diverse avenues for raising additional capital—provides the "reasonable assurance" required by 10 CFR 50.33(f). The availability of these funds aligns with the project schedule and the strategic goals of both the University and NANO Nuclear.

Sincerely,

Nano Nuclear Energy Inc.

By: ______________________________

Name: James Walker

Title: Chief Executive Officer

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REFERENCES

[1] Nano Nuclear Energy Quarterly Report. UNITED STATES SECURITIES AND EXCHANGE COMMISSION Form 10-Q

[2] Nano Nuclear Energy Shelf Registration date March 13, 2026.

APPENDIX 15B CONFIRMATION OF ACTIVE AND GOOD FAITH NEGOTIATIONS FOR DISPOSAL CONTRACT FOR THE ILLINOIS MICROREACTOR PROJECT AT THE UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN

Department of Energy
Washington, DC 20585

October 6, 2025

VIA ELECTRONIC MAIL

Dr. Caleb Brooks
Professor
University of Illinois Urbana-Champaign
111C Talbot Laboratory
104 S. Wright St.
Urbana, IL 61801-2959
csbrooks@illinois.edu

Re: Confirmation of Active and Good Faith Negotiations for Disposal Contract for the Illinois Microreactor Project at the University of Illinois Urbana-Champaign

Dear Professor Caleb Brooks:

I am writing to affirm that the University of Illinois Urbana-Champaign and NANO Nuclear are actively and in good faith negotiating with the Secretary of Energy for a contract for the Illinois Microreactor Project under section 302(b) of the Nuclear Waste Policy Act of 1982, as amended ("NWPA").

Although section 302(b)(1)(A)(ii) of the NWPA assigns to the Secretary of Energy the function of making the above affirmation, section 304(b) of the NWPA further provides that the director of the Office of Civilian Radioactive Waste Management (OCRWM) "shall be responsible for carrying out the functions of the Secretary under [the NWPA], subject to the general supervision of the Secretary." In 2010, OCRWM was closed and the functions relating to the Standard Contract were assigned to the Office of the General Counsel. Those functions were later assigned by the General Counsel to my office.

DOE is reviewing the issue of the appropriate contract mechanism and will be in contact for further discussions.

Sincerely,

/s/ Kurt Espiritu

Kurt Espiritu
Acting Director
Office of Standard Contract Management
Office of the General Counsel

APPENDIX 15C MEMORANDUM OF UNDERSTANDING BETWEEN THE BOARD OF TRUSTEES OF THE UNIVERSITY OF ILLINOIS AND NANO

I ILLINOIS

Memorandum of Understanding
Between
The Board of Trustees of the University of Illinois
and
NANO Nuclear Energy Inc.

1 Purpose

This Memorandum of Understanding (MOU) sets forth the intent of The Board of Trustees of the University of Illinois, a body corporate and politic organized and existing under the laws of the State of Illinois, on behalf of the University of Illinois, Urbana-Champaign, through Sponsored Programs Administration, 1901 South First Street, Suite A, Champaign, IL 61820 ("University") and NANO Nuclear Energy Inc. with its principal engineering offices at 111 Windsor Dr, Oak Brook, IL 60523 ("Company") to collaborate on the development, construction, and operation of a nuclear reactor on University-owned land. The reactor will serve as a platform for research, training, and at-scale demonstration.

The primary goal of the project is to efficiently deploy a KRONOS MMR nuclear reactor on the University campus. This deployment is strategically aligned with the energy future of the state of Illinois, offering a practical pathway to achieve the ambitious carbon-reduction goals codified in legislation such as the Climate and Equitable Jobs Act (CEJA). Beyond demonstrating the commercial viability of the Company's technology, this project positions the University as a central hub for state-level energy innovation. The data and operational experience generated here will not only enhance research and education but will also serve as a vital resource for the state of Illinois as it navigates the transition to a zero-carbon economy.

2 Scope of Collaboration

The parties agree to collaborate on the following key activities:

- Design and Construction
  - The Company will provide the nuclear reactor design and lead financing of project costs, including, but not limited to, project development, site preparation, procurement of nuclear reactor components, on-site construction, commission and startup.
  - Company will engage and pay an outside firm for procurement and assembly of the nuclear reactor.
  - The University will lead engagement with Department of Energy programs and other programs to procure fuel services in line with established practices for university-based research reactors. The University will submit all necessary construction permits and approvals.
  - The University will lead engagement with the state of Illinois and pursue closer involvement from the state in areas of shared priorities.
  - The nuclear reactor will be constructed on land owned by the University.
- Ownership and Operation
  - Upon completion, the University will own and operate the nuclear reactor in compliance with requirements for a Class 104(c) license.
  - The mechanism and timing for the transfer of ownership from the Company to the University will be determined in a future agreement.
- Research/Development and Data
  - The University will lead research in nuclear technology utilizing the nuclear reactor.

- The University will have full ownership and control over all energy and non-energy products generated by the nuclear reactor.
  - The Company will have access to utilize nuclear reactor and project performance data, generated during construction, operation and maintenance, refueling, and decommissioning.
- Training and Workforce Development
  - The University will utilize the nuclear reactor to lead a workforce development center.
  - The nuclear reactor will serve as a training center for the Company's personnel and other stakeholders, in coordination with the University's academic and research programs.
- Maintenance and Lifecycle Support
  - The parties intend to collaborate on the maintenance of the nuclear reactor throughout its lifecycle, as determined in a future agreement

3 Ancillary Agreements

The parties acknowledge that additional agreements will be required to address specific activities. Each ancillary agreement will include appropriate terms and conditions tailored to the activity it governs.

4 Joint Project Delivery Steering Committee

The parties intend to establish a joint project delivery steering committee to provide oversite during deployment through construction phase. This committee will include representatives from both the Company and the University and will be responsible for promoting alignment on key activities contemplated under this MOU.

5 Joint Operations Steering Committee

Operating the deployed nuclear reactor with excellence throughout the lifetime of the project will be a top level goal. To ensure this is met, the parties intend to establish a joint operations steering committee. This committee will include representatives from both the Company and the University and will be responsible for advising major operational decisions during the project lifetime. Decisions that influence the continued operation of the nuclear reactor, such as re-licensing, long-term shutdown, or decommissioning will include input from the joint operations steering committee.

6 Confidentiality

In connection with activities under this MOU, if either party discloses information, it considers proprietary and confidential ("Confidential Information"), such information shall be clearly identified as confidential at the time of disclosure. If disclosed in writing, it must be marked as confidential. If disclosed orally or visually, it must be identified as confidential at the time of disclosure and confirmed in writing to the recipient within fifteen (15) days.

The receiving party agrees to maintain such Confidential Information in strict confidence and not to use it for any purpose outside the scope of this MOU. This obligation of confidentiality and non-use shall remain in effect for a period of three (3) years from the date of disclosure, unless otherwise agreed in writing by the parties.

If University receives a request under the Illinois Freedom of Information Act or a request by legal process or administrative order to disclose Confidential Information, University will use reasonable efforts to provide prompt notice to Company and will reasonably cooperate with Company to protect any Company Confidential Information.

7 Intellectual Property

The parties acknowledge that intellectual property may be developed during activities conducted under this MOU. Ownership and rights to use such intellectual property will be defined in ancillary agreements, as referenced in Article 3.

In the absence of an applicable ancillary agreement:

- Intellectual property created solely by one party shall be owned exclusively by that party.
- Intellectual property created jointly by both parties shall be jointly owned.

Nothing in this MOU shall be construed to grant either party any rights or interests in the other party's background intellectual property, except to the extent necessary to carry out the activities contemplated under this MOU.

8 Compliance with Laws and Regulations

Each Party shall, at all times during the term of this MOU, comply with all applicable federal, state, and local laws, regulations, ordinances, and codes governing the design, construction, operation, and maintenance of the nuclear reactor and any related activities conducted under this MOU.

9 General Matters

9.1 Use of Names

The parties agree that each party may use factual information regarding the existence and purpose of the relationship that is the subject of this MOU for legitimate business purposes, to satisfy any reporting and funding obligations, or as required by applicable law or regulation without written permission from the other party. In any such statement, the relationship of the parties shall be accurately and appropriately described. Neither party will use the name of the other in any form of advertising or publicity without the express written permission of the other party. Company shall seek permission from University by submitting the proposed use, well in advance of any deadline, to the Associate Chancellor, Office of Strategic Communications and Marketing, University of Illinois; Email stratcom@illinois.edu.

9.2 Notices

Any notice given under this MOU will be in writing and will be effective upon receipt evidenced by: (a) personal delivery; (b) confirmed receipt of email; (c) return receipt of postage prepaid registered or certified mail; or (d) delivery confirmation by commercial overnight carrier. All communications will be sent to the addresses set forth below or to such other address designated by a party by written notice to the other party in accordance with this section:

University: **For technical matters:**
Caleb Brooks
Nuclear, Plasma and Radiological Engineering
111C Talbot Laboratory
104 S. Wright Street,
Urbana, IL 61801
Email: csbrooks@illinois.edu

For contractual matters or official notices:
University of Illinois
Director Pre-Award, Sponsored Programs Administration
1901 South First Street
Champaign, IL 61820-7406
Telephone: (217) 333-2187
Email: spa@illinois.edu

Company: NANO Nuclear Energy Inc
111 Windsor Dr,
Oak Brook, IL 60523
Email: florent@nanonuclearenergy.com

10 9. Non-Binding Nature

This MOU reflects the mutual understanding and intent of the parties but does not constitute a legally binding agreement. Binding obligations will be set forth in subsequent ancillary agreements, as referenced in Article 3.

11 Term and Termination

This MOU shall remain in effect for the duration of the project or until terminated by either party with sixty days (60) written notice to the other party. This MOU may be extended or amended by mutual consent of the parties, confirmed in a written amendment signed by authorized representatives of each party.

The rest of the page is intentionally left blank

Each party represents that the individuals signing this MOU have the authority to sign in the capacity indicated.

Signed:

FOR THE UNIVERSITY OF ILLINOIS URBANA-CHAMPAIGN

12/8/25

Rashid Bashir, Dean Date
Grainger College of Engineering

12/8/25

Susan A. Martinis Date
Senior Vice Chancellor for Research and Innovation

12/15/2025

Charles L. Isbell, Jr., Chancellor Date

NANO NUCLEAR ENERGY INC.

Jay Jiang Yu Date
Founder, Executive Chairman and President

James Walker Date
Chief Executive Officer

FOR THE BOARD OF TRUSTEES OF THE UNIVERSITY OF ILLINOIS

Paul N Ellinger

12/16/2025

Paul N. Ellinger, Comptroller Date

JRobinson

Comptroller Delegate: Julie Robinson, Senior Associate Director

RUSH MOU; Fully Executed; Award 124848

Final Audit Report 2025-12-16

Created:	2025-12-16 (Central Standard Time)
By:	Robin Beach (beach@illinois.edu)
Status:	Signed
Transaction ID:	CBJCHBCAABAALjjMbKgBTXLwakDC8jxtFKo_12toxWak
Number of Documents:	1
Document page count:	5
Number of supporting files:	0
Supporting files page count:	0

"RUSH MOU; Fully Executed; Award 124848" History

Document created by Robin Beach (beach@illinois.edu)
2025-12-16 - 1:55:24 PM CST- IP address: 128.174.41.234

Document emailed to Julie Robinson (jrobnsn@illinois.edu) for signature
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Agreement viewed by Julie Robinson (jrobnsn@illinois.edu)
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Julie Robinson (jrobnsn@illinois.edu) authenticated with Adobe Acrobat Sign.
2025-12-16 - 2:37:08 PM CST

Document e-signed by Julie Robinson (jrobnsn@illinois.edu)
Signature Date: 2025-12-16 - 2:37:08 PM CST - Time Source: server- IP address: 128.174.41.198

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APPENDIX 15D SPONSORED RESEARCH AGREEMENT AMENDMENT NO. 2 BETWEEN THE BOARD OF TRUSTEES OF THE UNIVERSITY OF ILLINOIS AND NANO

SPONSORED RESEARCH AGREEMENT AMENDMENT NO. 02

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EXHIBIT A
STATEMENT OF WORK
Licensing of Micro Modular Nuclear Reactor for Siting at UIUC

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Amendment to Exhibit B: Budget

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USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 16 - OTHER LICENSE CONSIDERATIONS

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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CHAPTER 16 OTHER LICENSE CONSIDERATIONS

16.1 PRIOR USE OF REACTOR COMPONENTS

The University of Illinois Urbana-Champaign (U. of I.) research reactor will be constructed of new and appropriately qualified structures, systems, and components to conduct facility operations. Discussions regarding used systems and components are not applicable to the facility.

16.2 MEDICAL USE OF NON-POWER REACTORS

The U. of I. research reactor will not contain equipment or facilities associated with direct medical administration of radioisotopes or any other radiation-based therapies; therefore, U. of I. will not be applying for a Class 104(a) license as described in 10 CFR 50.21. Hence, discussions involving medical use of the facility are not applicable.

USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 17 - DECOMMISSIONING

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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CHAPTER 17 DECOMMISSIONING

A decommissioning report for the University of Illinois Urbana-Champaign (U. of I.) research reactor will be provided with the application for the Operating License consistent with 10 CFR 50.33(k) and will address the content requirements in 10 CFR 50.75(d)(2). Section 15.3 will describe the financial assurances for the availability of funding to support decommissioning.

17.1 POSSESSION-ONLY LICENSE AMENDMENTS

This section relates to a possession-only license amendment and is not applicable to the construction and operation phases of the U. of I. research reactor facility.

USNRC Project No. 99902094

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 18 - HIGHLY ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS

Revision 0

submitted by

THE UNIVERSITY OF ILLINOIS

Illinois Microreactor RD&D Center

in collaboration with

NANO Nuclear Energy Inc.

to

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

Division of Advanced Reactors and Non-Power Production and Utilization Facilities

March 31, 2026

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CHAPTER 18 HIGHLY ENRICHED TO LOW-ENRICHED URANIUM CONVERSIONS

The University of Illinois Urbana-Champaign (U. of I.) research reactor fuel consists of tristructural isotropic fuel particles using low-enriched uranium Plus. The reactor facility will not perform fuel conversion activities nor does it utilize highly enriched uranium fuel that is enriched to 20% or more in the isotope U-235, as described in 10 CFR 50.2. Therefore, this chapter and the requirements in 10 CFR 50.64 are not applicable to the facility.