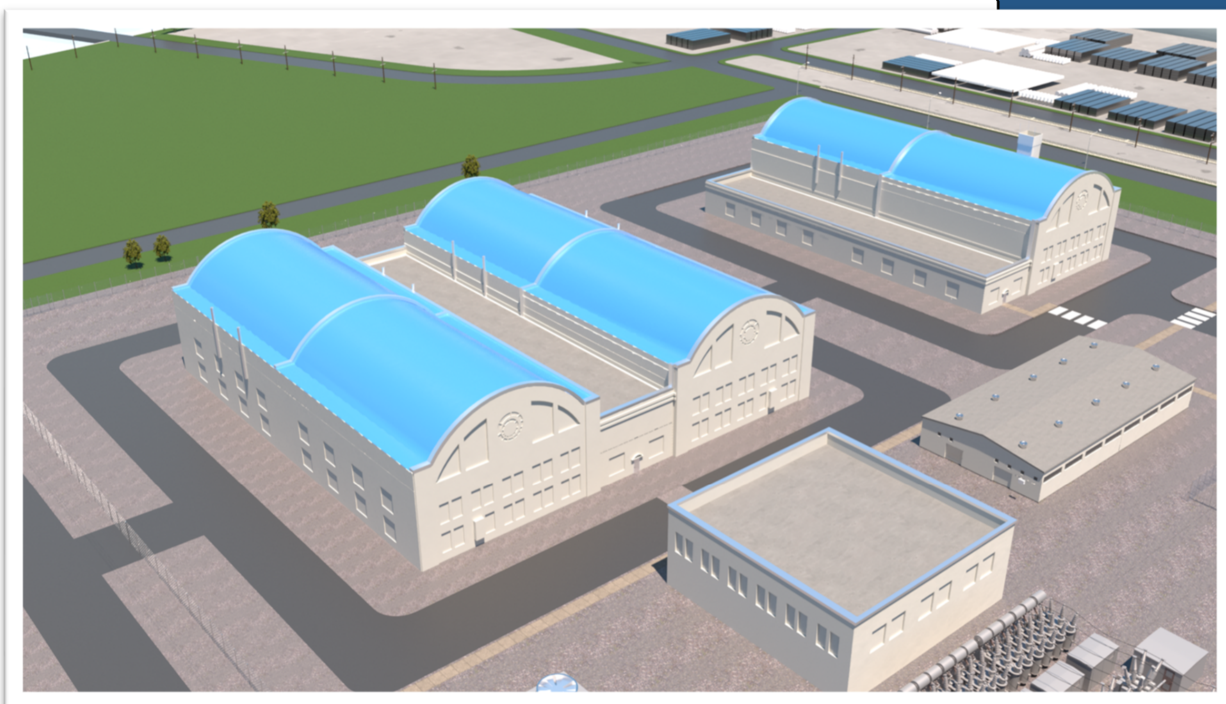
Enclosure 2

Preliminary Safety Analysis Report (Non-Proprietary)

Kairos Power



Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

H2-PSAR-000001

Revision 0

July 2023

Master Table of Contents

CHAPTER 3
DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

CHAPTER 4
REACTOR DESCRIPTION

CHAPTER 5
HEAT TRANSPORT SYSTEM

CHAPTER 6
ENGINEERED SAFETY FEATURES

CHAPTER 7
INSTRUMENTATION AND CONTROL SYSTEMS

CHAPTER 8
ELECTRIC POWER SYSTEMS

CHAPTER 9
AUXILIARY SYSTEMS

CHAPTER 10
EXPERIMENTAL FACILITIES AND UTILIZATION

CHAPTER 11
RADIATION PROTECTION PROGRAM AND WASTE MANAGEMENT

CHAPTER 12
CONDUCT OF OPERATIONS

CHAPTER 13
ACCIDENT ANALYSIS

CHAPTER 14
TECHNICAL SPECIFICATIONS

CHAPTER 15
FINANCIAL QUALIFICATIONS

CHAPTER 16
OTHER LICENSE CONSIDERATIONS

CHAPTER 17
DECOMMISSIONING AND POSSESSION-ONLY LICENSE AMENDMENTS

CHAPTER 18
HIGHLY ENRICHED TO LOW ENRICHED URANIUM CONVERSION

List of Acronyms

AAI	accidental aircraft impact
ACI	American Concrete Institute
AEC	Atomic Energy Commission
AGR	advanced gas reactor
AGR	advanced gas-cooled reactor
AISC	American Institute of Steel Construction
AL	analytical limit
ALARA	as low as reasonably achievable
ALOHA	Areal Locations of Hazardous Atmospheres
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASER	Oak Ridge Reservation Annual Site Environmental Report
ASHRAE	American Society of Heating, Refrigeration, and Air Conditioning Engineers
ASTM	American Society for Testing and Materials International
ATS	automatic transfer switch
BCS	base control system
BLEVE	boiling liquid expanding vapor explosion
BPS	backup power system
BUMS	burn up measurement sensor
C&C	components and cladding
CAMEO	Computer-Aided Management of Emergency Operations
CAPARS	computer-assisted protective action recommendation system
CCS	chemistry control system
CCWS	component cooling water system
CEO	chief executive officer
CEUS	Central and Eastern United States
CFD	computational fluid dynamics
CONUS	continental United States
CR-ESP	Clinch River early site permit
CRN	Clinch River Nuclear
CVSZ	Central Virginia Seismic Zone
CWS	chilled water system
DAW	dry active waste
DHRS	decay heat removal system
DOE	Department of Energy
DRS	design response spectra
EA	environmental assessment
EAB	exclusion area boundary
ED	emergency director
EOF	emergency operations facility
EPA	Environmental Protection Agency
EPZ	emergency planning zone
ESC	emergency support center
ESCS	equipment and structural cooling system

ESF	engineered safety feature
ESP	early site permit
ESPA	early site permit application
ESRI	Environmental Systems Research Institute
ETSZ	Eastern Tennessee Seismic Zone
ETTP	East Tennessee Technology Park
FAA	Federal Aviation Administration
FDT	fire dynamics tools
FEMA	Federal Emergency Management Agency
FIS	flood insurance study
FMEA	failure mode and effects analysis
GIS	geographical information system
GMC	ground motion characterization
GMPE	ground motion prediction equations
GPS	global positioning system
HALEU	high assay low enriched uranium
HEPA	high efficiency particulate air
HEU	highly enriched uranium
HRR	heat rejection radiator
HRS	heat rejection subsystem
HSI	human / system interface
I&C	instrumentation and control
IBC	international building code
IDLH	immediately dangerous to life or health
IEC	International Electrotechnical Commission
IGS	inert gas system
IHTCS	intermediate heat transport control system
IHTS	intermediate heat transport system
IHX	intermediate heat exchanger
IMS	inventory management system
INL	Idaho National Laboratory
ISG	interim staff guidance
ISP	intermediate salt pump
ISRS	in-structure response spectra
ISV	intermediate salt vessel
KP-FHR	Kairos Power Fluoride Salt-Cooled High Temperature Reactor
LCO	limiting condition for operation
LEL	lower explosive limit
LFL	lower flammability limit
LFRS	lateral force resisting system
LPZ	low population zone
LSSS	Limiting Safety System Setting
LWR	light water reactor
MAR	radioactive material at risk for release
MCC	motor control center
MCE_R	maximum considered earthquake
MCR	main control room
MHA	maximum hypothetical accident

mph	miles per hour
MSRE	molten salt reactor experiment
MSS	material surveillance system
MWFRS	main wind-force resisting system
NAVD	North American Vertical Datum
NFPA	National Fire Protection Association
NGA	next generation attenuation
NIOSH	National Institute for Occupational Safety and Health
NMSZ	New Madrid Seismic Zone
NOAA	National Oceanic and Atmospheric Administration
NRC	Nuclear Regulatory Commission
NSHMP	national seismic hazard mapping project
OL	Operating License
ORGDP	Oak Ridge Gaseous Diffusion Plant
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
PAC	protective action criteria
PAG	Protective Action Guide
PBR	pebble bed reactor
PCS	plant control system
PDC	principal design criteria
PEM	pebble extraction machine
PHSS	pebble handling and storage system
PHTCS	primary heat transport control system
PHTS	primary heat transport system
PIRT	phenomena identification and ranking table
PM	plant manager
PMF	probable maximum flood
PMP	probable maximum precipitation
PMWP	probable maximum winter precipitation
PPE	personal protective equipment
PSAR	preliminary safety analysis report
PSHA	probabilistic seismic hazard analysis
psi	pounds per square inch
psid	pounds per square inch differential pressure
psig	pounds per square inch gauge pressure
PSP	primary salt pump
QAPD	quality assurance program description
QM	Quality Manager
RAHS	reactor auxiliary heating system
RB	reactor building
RBHVAC	Reactor Building heating, ventilation, and air conditioning
RCACS	reactor coolant auxiliary control system
RCAS	reactor coolant auxiliary systems
RCS	reactor control system
RCSS	reactivity control and shutdown system
REMP	radiological environmental monitoring program
RF	release fraction

RG	regulatory guide
RLME	repeated large magnitude earthquakes
RMIS	remote maintenance and inspection system
ROSP	remote onsite shutdown panel
RP	radiation protection
RPS	reactor protection system
RS	reactor system
RSO	radiation safety officer
RTMS	reactor thermal management system
RTS	reactor trip system
RV	reactor vessel
RVSS	reactor vessel support system
SA	spectral acceleration
SARRDL	specified acceptable system radionuclide release design limit
SDC	seismic design category
SF	scale factor
SFCS	spent fuel cooling system
SL	safety limit
SNM	special nuclear material
SPT	standard penetration test
SR	surveillance requirement
SS	shift supervisor
SSAR	site safety analysis report
SSC	structures, systems, and components
SSHAC	senior seismic hazard analysis committee
SSI	soil-structure interaction
STEL	short term exposure limit
TEDE	total effective dose equivalent
TEMA	Tennessee Emergency Management Agency
TMS	tritium management system
TMS-HRR	heat rejection radiator tritium capture system
TMS-IGS	inert gas system tritium capture system
TMS-IHTS	intermediate heat transport system tritium capture system
TNT	trinitrotoluene
TRISO	tri-structural isotropic
TSC	technical support center
TVA	Tennessee Valley Authority
TLV	threshold limit value
TWA	time-weighted average
UCO	uranium oxycarbide
UEL	upper explosive limit
UHRS	uniform hazard response spectra
UHS	ultimate heat sink
UPS	uninterruptible power supply
USACE	United States Army Corps of Engineers
USGS	United States Geological Survey
WTP	Water Treatment Plant

Kairos Power

Chapter 1

The Facility

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

None

List of Figures

None

CHAPTER 1 THE FACILITY

1.1 INTRODUCTION

Kairos Power LLC (Kairos Power), the applicant, is requesting approval for a Construction Permit for a two unit 35 MWth non-power reactor facility, known as Hermes 2, to be located within the East Tennessee Technology Park near Oak Ridge, Tennessee. The Hermes 2 reactor facility (the facility) is expected to be licensed as a non-power reactor under Title 10 of the Code of Federal Regulations (10 CFR) Part 50 "Domestic Licensing of Production and Utilization Facilities," specifically 10 CFR 50.21(c). Kairos Power is a privately held company that was created for the purpose of commercializing and deploying the Kairos Power fluoride salt-cooled, high temperature reactor (KP-FHR) technologies. The purpose of the non-power reactor facility is to test and demonstrate the key technologies, design features, and safety functions of the KP-FHR technology and its structures, systems, and components (SSC) for a two-unit facility, including electrical power production. The facility will also provide data and insights for the safety analysis tools and computational methodologies used for the design and licensing of a KP-FHR commercial power reactor. Note that for instances where the term "reactor" is used in this document for Hermes 2 it would apply to both reactors unless otherwise stated.

This Preliminary Safety Analysis Report (PSAR) is submitted in accordance with the provisions of 10 CFR 50.34(a) in support of the construction permit application.

The PSAR meets the requirements in 10 CFR 50.34(a) and 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," as augmented by the Final Interim Staff Guidance Augmenting NUREG-1537, Part 1, "Guidelines for Preparing and Reviewing Applications for Licensing Non-Power Reactors: Format and Content for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors," on October 17, 2012. Specific content in this PSAR has been organized and formatted specifically to describe the KP-FHR technology and to align with the design architecture of the non-power reactor facility. Balance of plant power generation systems are not addressed in NUREG-1537. Section 9.9 of this application provides the discussion of these systems.

An overview description of the inherent and passive safety features of the facility are addressed in Section 1.2.2. Details regarding the site geographical location and the surrounding areas are presented in Chapter 2.

1.2 SUMMARY AND CONCLUSIONS ON PRINCIPAL SAFETY CONSIDERATIONS

The KP-FHR is an advanced reactor technology developed in the United States over the last decade. The technology follows from Department of Energy (DOE) sponsored research and development at universities and national laboratories. The fundamental concept is the combination of Tri-structural Isotropic (TRISO) particle fuel coupled with a molten fluoride salt coolant. This combination results in a high temperature, low-pressure reactor system with robust inherent safety characteristics. The combination of extremely high-temperature-tolerant fuel and low-pressure, single-phase, chemically stable reactor coolant removes entire classes of potential fuel-damage scenarios, greatly simplifying the design and reducing the number of safety systems. The intrinsic low pressure of the reactor and associated piping, along with the fission product retention provided by the TRISO fuel, enhances safety and eliminates the need for low-leakage, pressure retaining containment structures. Additionally, the design relies on passive decay heat removal and does not need an emergency core cooling system (ECCS) for decay heat removal or replacement of coolant inventory.

The major plant systems are the reactor system (RS), the primary heat transport system (PHTS), intermediate heat transport system (IHTS), and the decay heat removal system (DHRS). The RS is described in Chapter 4, the PHTS and IHTS are described in Chapter 5, and the DHRS is described in Chapter 6. Other associated plant support systems are described in Chapter 7 (instrumentation and control), Chapter 8 (electrical), and Chapter 9 (auxiliary systems), which includes a discussion of balance of plant power generation systems.

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 KP-FHR design relies on a functional containment approach to meet the siting regulations in 10 CFR 100.11(a) for dose limits. The functional containment represents an engineered safety feature of the reactor and is implemented principally by the high temperature TRISO particle fuel. The fuel utilizes a carbon matrix coated particle fuel, similar to that developed for high-temperature gas-cooled reactors, in a pebble-based fuel element. Coatings on the particle fuel have been demonstrated to provide retention of fission products to design temperatures in excess of 1600°C. The fuel design and performance are discussed further in Section 4.2.

The reactor coolant also provides a secondary functional containment role and is a chemically stable, low-pressure molten fluoride salt coolant. The mixture consists of an enriched lithium fluoride (LiF) and beryllium fluoride (BeF_2) salts in a ratio of approximately 2:1. The Molten Salt Reactor Experiment (MSRE) program and the subsequent operation of the MSRE nuclear reactor utilized this Fluoride-Lithium-Beryllium based salt as an effective nuclear coolant for both the primary coolant (which had dissolved fuel) and the intermediate coolant (which was clean coolant) (Reference 1). Furthermore, there has been significant research into the stability and compatibility of this coolant in fission and fusion energy applications since the operation of the MSRE (Reference 2). The Hermes reactor operates with a low (near atmospheric) overpressure in the reactor vessel head space. The reactor coolant is further described in Section 5.1.

Fission product retention and control in the reactor facility relies on a functional containment strategy (comprised of the TRISO fuel and Flibe coolant) as a means of preventing significant radionuclide release to the environment during normal operations and postulated events. The functional containment is described in Section 6.2. The analysis of postulated events which address the siting limits in 10 CFR 100.11(a) is described in Chapter 13.

Flibe coolant, while chemically stable, contains potentially toxic constituents including beryllium. The reactor building and ventilation system function as a confinement to manage and control beryllium hazards but are not credited for mitigation of radiological releases during postulated events.

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 limits of 10 CFR 20.1201 and 10 CFR 20.1301 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 limits of 10 CFR 100.11(a). The radiation protection program which addresses the limits in 10 CFR 20 as described in Chapter 11 will be provided in the application for an Operating License consistent with 10 CFR 50.34(b)(3).

1.2.2 Inherent and Passive Safety Features

The KP-FHR technology includes a number of inherent safety features:

- The design includes a functional containment provided by the design of the TRISO fuel layers and by the reactor coolant, which function as barriers that control the spread of radionuclides. The functional containment approach is described in Section 6.2.
- The fuel design includes TRISO layers comprised of highly refractory pyrocarbon and silicon carbide materials that can withstand high temperatures providing margin to failure in transient conditions. The fuel design is discussed in Section 4.2.
- The primary heat transport system operates at near-atmospheric pressures, reducing the potential for energetic releases of reactor coolant. The reactor coolant is discussed in Section 5.1.
- The reactivity coefficients are a key inherent safety feature of the reactor system. The fuel, moderator, and coolant temperature reactivity coefficients are all negative. The coolant void coefficient is also negative. The reflector coefficient is positive but is small. The core design is described in Section 4.5.
- The facility is designed to be able to execute safety functions, including the removal of decay heat, without reliance on electrical power in response to a postulated event. The evaluation of postulated events is described in Chapter 13.
- The reactor vessel and other safety-related components are located within a seismically isolated, safety-related structure that provides protection to safety-related SSCs during the design basis earthquake and provides protection from other design basis natural phenomena events such as high winds, tornadoes, and external flooding. Non-safety related SSCs with the potential to cause unacceptable interactions with safety-related SSCs are either prevented from adverse interaction by protective barriers, seismically mounted to remain in place during the design basis earthquake, or located a sufficient distance away to preclude interaction. The building structural design and design considerations from natural phenomena events are described in Chapter 3.
- Radiological shielding is used to minimize occupational exposures in normally occupied areas of the facility from radioactive materials. The biological shield around the reactor vessel is described in Section 4.4. The radiation protection program as described in Section 11.1 will be provided in the application for an Operating License consistent with 10 CFR 50.34(b)(3).
- Ventilation systems are designed such that the flow is from areas of no or low contamination zones to those plant process areas potentially containing higher concentrations of radioactive and beryllium containing materials. The ventilation systems are described in Chapter 9.

1.2.3 Design Features and Design Bases

The principal design criteria (PDC) for the facility SSCs are described in Section 3.1 and are based on those specified in the NRC-approved Kairos Power Topical Report, KP-TR-003-NP-A (Reference 3). The system-related sections throughout this SAR describe how the design bases, including the PDC, are satisfied.

As noted above, the reactor design relies on a functional containment approach, rather than a low-leakage, pressure-retaining containment structure and reactor coolant pressure boundary that is typically used for light water reactors (LWRs) to control the release of fission products. The functional containment approach is to control radionuclides primarily at their source within the TRISO coated fuel particle under normal operations and postulated events, without reliance on active safety features or on operator actions. The functional containment relies primarily on the multiple barriers within the TRISO fuel particle layers to ensure that the dose at the site boundary (from postulated accidents) meets regulatory limits. Additionally, for the fuel in the reactor core, the reactor coolant serves as an additional barrier providing retention of most fission products that could escape the fuel particle barriers. This additional retention barrier is a key feature of the enhanced safety and reduced source term. To enable fission product retention in the fuel particle and the reactor coolant, the reactor vessel maintains the fuel pebbles in the reactor core submerged in the coolant.

The SSCs in the facility are assigned a nuclear safety classification, as follows:

- Safety-related SSCs: Those SSCs that are relied upon to remain functional during normal operating conditions and during and following design basis events to assure:
  - The integrity of the portions of the reactor coolant boundary relied upon to maintain coolant level above the active core (see below);
  - The capability to shut down the reactor and maintain it in a safe shutdown condition; or
  - The capability to prevent or mitigate the consequences of accidents which could result in potential exposures exceeding the limits set forth in 10 CFR 100.11.
- Non-safety related: Those SSCs that are not in the above safety classification.

Note that the definition of safety-related described above is different from that specified in 10 CFR 50.2, Definitions. The definition in 10 CFR 50.2 is based on LWR technologies which rely on a reactor coolant pressure boundary as one of the three fission product retention barriers. As discussed above, the FHR technology does not credit a coolant pressure boundary for fission product retention, but rather relies on a functional containment as described in Section 6.2. However, the reactor vessel is credited for retaining the fuel pebbles in the reactor core in a Flibe-wetted environment for heat removal under all postulated events, as described in Section 4.3. Note that no other portion of the reactor coolant boundary is credited in the safety analysis for fission product retention or to ensure decay heat removal. Therefore, departure from the definition of safety-related in 10 CFR 50.2 is necessary for the FHR technology. Note that the term “safety-significant” used in Reference 4 is not used in this definition because the term is not applicable to the Hermes reactor, consistent with the discussion in Section 3.1.

A summary of SSC safety classifications is provided in Section 3.5.

1.2.4 Potential Accidents at the Facility

Potential 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. An evaluation of potential events, including the maximum hypothetical accident, is summarized in Chapter 13.

Note that the requirements in 10 CFR 50.34(g) require applicants for construction permits to address the analyses and descriptions of equipment for combustible gas control as required by 10 CFR 50.44 in the application. The combustible gas requirements for non-water cooled reactors is specified in 10 CFR 50.44(d). Accidents involving combustible gases are not technically relevant to the design of the Hermes reactor. The postulated events for the reactor do not feature phenomena that result in the generation of combustible gas. As a result, combustible gases do not represent a hazard to the integrity of the functional containment barrier and its fission product retention capability.

1.2.5 References

1. Oak Ridge National Laboratory, "MSRE Design and Operations Report, Part 1, Description of the Reactor System," ORNL-TM-728. January 1965.
2. Oak Ridge National Laboratory, "An Overview of Liquid-Fluoride-Salt Heat Transport Systems," ORNL-TM-2010-156. 2010.
3. Kairos Power LLC, "Principal Design Criteria for the Kairos Power Fluoride Salt Cooled High Temperature Reactor," KP-TR-003-NP-A. June 2020.
4. Kairos Power LLC, "Regulatory Analysis for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor," KP-TR-004-NP-A. June 2022.

1.3 GENERAL DESCRIPTION OF THE FACILITY

1.3.1 Geographical Location

The facility is located within the East Tennessee Technology Park (ETTP) in Oak Ridge, Tennessee. The facility latitude and longitude are provided in Section 2.1. The site location is illustrated in Figure 2.1-1.

1.3.2 Principal Characteristics of the Site

The site consists of an area located in the northwestern portion of the Heritage Center within the ETTP (the ETTP consists of the Heritage Center, site of former uranium enrichment operations, and the Horizon Center Industrial Park). The property is at the site of the former Buildings K-31 and K-33 of the Oak Ridge Gaseous Diffusion Plant (ORGDP), where uranium enrichment operations occurred from 1954 until the mid-1980s. The overall site is an approximately 185 acre (74.8 hectare) parcel that had been used as farmland prior to the construction of the ORGDP. The site has since been restored to a brown field site by DOE and the former above-grade portions of the buildings were removed.

The site is entirely contained within the ETTP, Oak Ridge, Tennessee. The dominant land use in the site area is a brown field from the ORGDP site. Other operations in the site area are associated with DOE facilities, ongoing conversion of former DOE sites for commercial use, and various industrial activities. Other activities on the site include the proposed Hermes test reactor, which is a single 35 MWth non-power producing test reactor, a non-nuclear engineering test unit (ETU) designed for large-scale Flibe system tests that are generally proto-typical of the Hermes test reactor, and a potential future Kairos Power fuel fabrication facility. Principal characteristics of the site are further described in Chapter 2.

1.3.3 Principal Design Criteria, Operating Characteristics, and Safety Systems

1.3.3.1 Principal Design Criteria

The principal design criteria for the facility are described in Section 3.1. The principal design criteria for the facility are based on the criteria included in Kairos Power Topical Report KP-TR-003-NP-A (Reference 1).

1.3.3.2 Operating Characteristics

There are two reactor units at the facility. Each reactor is designed to achieve a reactor power of 35 MWth (design rated thermal power) and a licensed lifetime of 11 years. The reactor parameters are provided in Table 4.1-1. Combined electrical power output from the two units is approximately 20 MW electric.

1.3.3.3 Safety Systems

The facility is a fluoride salt-cooled, high temperature reactor. The design of the reactor and fuel are discussed in detail in Chapter 4. The reactor coolant is addressed in Chapter 5. The safety-system classification is provided in Table 3.6-1.

1.3.4 Engineered Safety Features

Engineered safety features (ESF) are SSCs of the facility designed to mitigate the consequences of postulated events. For the non-power reactor facility, the ESFs are related to the containment of fission products, and the passive removal of decay heat. The ESFs are described in Chapter 6.

1.3.5 Instrumentation, Control, and Electrical Systems

The instrumentation and control (I&C) system monitors and controls plant operations during normal operations and planned transients. The system also monitors and actuates protection systems in the event of unplanned transients. The I&C system is comprised of the plant control system and the reactor protection system. The I&C system is discussed in Chapter 7.

The electrical system provides the normal and backup power to the facility and a switchyard for distribution of electric power production to the offsite electrical grid. The electrical system is discussed in Chapter 8.

1.3.6 Cooling and Other Auxiliary Systems

The chemistry control system (CCS) is used during normal plant operations to monitor the coolant chemistry and circulating activity in the reactor system and primary heat transport system, through the interface with the inventory management system, for compliance with Flibe specifications described in Section 5.1. The CCS is addressed in Section 9.1.1.

The inert gas system (IGS) provides argon gas flow to multiple locations in the reactor vessel, pebble handling and storage system, primary salt pump, inventory management system, reactivity control and shutdown system, and the chemistry control system. The IGS provides cover gas cleanup from impurities such as oxygen, water, and particulates. The IGS is addressed in Section 9.1.2.

The tritium management system (TMS) manages tritium generated in the reactor. The TMS provides for recovery and storage of tritium from various systems. Multiple systems provide for the collection, separation, and treatment of tritium. The TMS is a non-safety related system that provides for the collection and disposition pathway. This system is addressed in Section 9.1.3.

The reactor thermal management System (RTMS) consists of two primary subsystems - the equipment and structural cooling system and the reactor auxiliary heating system. Neither subsystem is credited with performing a safety-related function. The RTMS is addressed in Section 9.1.5.

Fire protection systems and programs are designed for varying levels of detection and notification of a fire events, suppression of small fires, and prevention of small fires from becoming large fires. Fire protection systems and programs are addressed in Section 9.4.

The power generation systems receive heat energy from the intermediate heat transport system to create superheated steam energy. The steam is directed to the turbine generator system and converted to electrical power. The power generation systems do not perform any safety-related functions and are not classified as safety-related. Portions of these systems are shared between the two nuclear units. These systems are addressed in Section 9.9.

Other auxiliary systems are also addressed in Chapter 9.

1.3.7 Radioactive Waste Management and Radiation Protection

A radiation protection program is established to protect the radiological health and safety of workers. The program complies with the regulatory requirements of 10 CFR Parts 19, 20, and 70. This program also includes the elements of an as low as reasonably achievable (ALARA) program, radiation monitoring and surveying, exposure control, dosimetry, contamination control, and environmental monitoring. The radiation protection program is addressed in Section 11.1 and will be provided in the application for an Operating License 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 testing and demonstration of the Kairos Power fluoride-salt cooled high temperature reactor technologies. It is expected that the testing and demonstration conducted at the facility will include such activities as:

- Fuel pebble irradiation testing
- Materials corrosion and irradiation testing
- Transient and power maneuvering testing
- Electric power production using nuclear-generated heat

The capability to perform these activities is included as part of the normal systems design described in this report and no additional facilities or capabilities are required.

Additionally, the facility will be used to test design options for non-safety related systems, as a part of the iterative development and testing approach.

1.3.9 Research and Development

The requirements in 10 CFR 50.34(a) require that the PSAR identify those structures, systems or components of the facility that require additional research and development to confirm the adequacy of their design; and identification and description of the research and development program which will be conducted to resolve any safety questions associated with such structures, systems, or components; and a schedule of the research and development program showing that such safety questions will be resolved at or before the latest date stated in the application for completion of construction of the facility. Such additional development activities are described below:

- Perform a laboratory testing program to confirm fuel pebble behavior (Section 4.2.1)
- Develop a high temperature material surveillance sampling program for the reactor vessel and internals (Section 4.3.4)
- Perform testing of high temperature material to qualify Alloy 316H and ER16-8-2 (Section 4.3)
- Perform analysis related to potential oxidation in certain postulated events for the qualification of the graphite used in the reflector structure (Section 4.3)
- Development and validation of computer codes for core design and analysis methodology (Section 4.5)
- Develop and perform qualification testing for a fluidic diode device (Section 4.6)
- Justification of thermodynamic data and associated vapor pressure correlations of representative species. (Section 5.1.3)
- Complete compatibility evaluations of the intermediate coolant and reactor coolant chemical interaction (Section 5.1.3)
- Develop process sensor technology for key reactor process variables (Section 7.5.3)
- Develop the reactor coolant chemical monitoring instrumentation (Section 9.1.1)

1.3.10 References

1. Kairos Power LLC, “Principal Design Criteria for the Kairos Power Fluoride Salt Cooled High Temperature Reactor,” KP-TR-003-NP-A. June 2020.

1.4 SHARED FACILITIES AND EQUIPMENT

The facility contains two reactors that only share portions of systems and equipment that are non-safety related. The shared systems include the plant control system, main control room, normal and backup power, balance of plant power generation systems, plant communication system, service water system, treated water system, auxiliary site services, facility physical security, portion of sensors, portions of the fire protection system, and portions of the radiation monitoring systems.

In all cases, shared systems or equipment are non-safety related. It is anticipated that some onsite infrastructure, not credited to perform a safety function or for safe operation, may be shared with other nearby or onsite facilities. Examples include site utilities such as electrical, gas, and water supply systems; warehousing and storage; environmental monitoring; security; and site access roads. Hazards to safe operation presented by nearby industrial, transportation, and military facilities are evaluated in Section 2.2.

1.5 COMPARISON WITH SIMILAR FACILITIES

1.5.1 Comparison of Physical Plant and Equipment

As stated in Section 1.2, the reactor uses a pebble-based TRISO fuel with molten fluoride salt reactor coolant in the reactor core. While there are no facilities that compare to the specific reactor fuel and coolant technology combination, elements of the FHR technology are present in existing technologies. The use of a molten fluoride salt coolant was demonstrated in the MSRE at Oak Ridge National Laboratory. Pebble-based fuel designs have been demonstrated in international high temperature gas cooled reactors and non-pebble TRISO fuel has been used in other designs. The reactor core and primary heat transport system have been developed specifically for use in this design. The reactor core is discussed in Chapter 4 and the primary heat transport system is discussed in Chapter 5.

1.5.2 Comparison of Reactor Core

1.5.2.1 Pebble Bed

The reactor core is similar to a gas-cooled pebble-bed reactor (PBR). The PBR is designed for a graphite-moderated, gas-cooled nuclear reactor operated at very high temperatures as compared to light water reactor designs. The basic design of the PBR features spherical fuel elements (pebbles). These tennis ball-sized pebbles are made of graphite (which acts as the moderator), and each pebble contains thousands of micro-fuel TRISO particles. TRISO particle fuel was used in Peach Bottom Unit 1 and Fort St. Vrain, albeit in stationary (non-pebble) fuel form.

The fuel pebbles in a PBR are similar to but larger than the fuel pebbles used in the Hermes reactor (approximately 60 mm (2.36 in) versus approximately 40 mm (1.57 in)). Unlike PBRs, the Hermes fuel is buoyant and moves in the same direction as the coolant. The pebbles also are based on an annular fuel layer, unlike PBR pebbles.

The PBR is cooled by an inert gas. The coolant has no phase transitions – it starts as a gas and remains as a gas. The reactor utilizes a molten fluoride salt coolant with a high freezing temperature and a high boiling temperature and no phase transition at operating and accident conditions.

1.5.2.2 Graphite

The reactor uses graphite as a moderator and, in this respect, is similar to several other designs. A graphite-moderated reactor uses carbon as a neutron moderator. The Flibe coolant used in the Hermes reactor also functions as a moderator. The Advanced Gas-cooled Reactor (AGR) is a type of graphite-moderated nuclear reactor designed and operated in the United Kingdom. The AGR is the second generation of gas-cooled reactors designed in the United Kingdom, using graphite as a neutron moderator and carbon dioxide as coolant.

The reactor utilizes a graphite reflector assembly comprised of a cylindrical side reflector that surrounds the active core, a bottom reflector structure below the active core, and an upper reflector structure that sits above the active core. The graphite reflector assembly in the reactor provides thermal inertia and neutron moderation while also reflecting neutrons back into the active core region. The Flibe coolant in gaps between reflector blocks and below the reactor core also moderates neutrons and prevents streaming of neutrons through these openings. The graphite reflector is positively buoyant in the Flibe reactor coolant. The reflector assembly and Flibe in the gaps between the reflector blocks also shields outer metallic structures from fast neutrons.

1.5.3 Comparison of Support Systems

Reactor auxiliary systems such as inventory control and chemistry monitoring are functionally similar to conventional systems but are Flibe-based. Other plant auxiliary supporting systems, including ventilation, cooling water systems, waste processing systems, electrical power systems, balance of plant power generation systems, and instrumentation and control, are generally conventional in nature. These supporting systems are discussed in Chapters 7 through 9.

1.6 SUMMARY OF OPERATIONS

As noted in Section 1.1, the purpose of the non-power reactor facility is to test and demonstrate the key technologies, design features, and safety functions of the KP-FHR technology and its SSCs for a two-unit plant, including electric power production. The facility will also provide data and insights for the safety analysis tools and computational methodologies used for the design and licensing of a KP-FHR commercial power reactor. The combined electrical output from the two units is approximately 20 MW electric. The expected cost to own and operate the plant would exceed revenue from the sale of electricity by at least a factor of two (50% of the annual cost of owning and operating the facility), in accordance with 10 CFR 50.22. The major programs to be performed in the facility will be provided in the application for an Operating License consistent with 10 CFR 50.34(b)(2).

The reactors will be operated for an 11-year lifetime over the full range of power to evaluate these aspects of the technology. The process system designs include the necessary features to monitor and assess plant performance in support of these objectives as described elsewhere in this report. The activation product inventory and fission product inventory from the normal operation of the facility and effluent release pathways to the environment, are discussed in Section 11.1 and a description of the radiation sources for the facility will be provided in the application for an Operating License consistent with 10 CFR 50.34(b)(3).

An analysis of postulated events from operation of the facility, including the radiological consequences of unplanned releases, is addressed in Chapter 13.

1.7 COMPLIANCE WITH THE NUCLEAR WASTE POLICY ACT OF 1982

Kairos Power intends to enter into a contract with the Department of Energy (DOE) for the disposition of high-level waste and spent nuclear fuel. The contract will provide that the DOE accept title to the fuel and the obligation to take the spent fuel and/or high-level waste for storage or reprocessing. This will be discussed further in the application for the Operating License, consistent with Section 302(b)(1) of the Nuclear Waste Policy Act of 1982.

1.8 FACILITY MODIFICATIONS AND HISTORY

This report is an application for the new construction of a non-power reactor facility. There are no prior operating histories of existing Nuclear Regulatory Commission licensed facilities nor modifications to existing licensed facilities to report.

Chapter 2

Site Characteristics

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

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 site selection of the Hermes 2 reactor site (site). The factors stated in 10 CFR 100.10 regarding site selection for test reactors are considered in the collection and assessment of the site data presented in this chapter. The site characteristics considered include:

- Geography and demography
- Nearby industrial, transportation, and military installations
- Meteorology
- Hydrology
- Geology, seismology, and geotechnical engineering

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 Site Location and Description

2.1.1.1 Specification and Location

The site is in the City of Oak Ridge in Roane County, Tennessee (Reference 1). Figure 2.1-1 illustrates the site location within the city, county, and state.

The site is located on a parcel that was previously part of the U.S. Department of Energy (DOE) Oak Ridge Reservation (ORR). The site previously housed Buildings K-31 and K-33, which were part of the ORR gaseous diffusion plant used to enrich uranium. The area of the ORR containing K-31 and K-33, as well as other uranium enrichment facilities, is also known as the East Tennessee Technology Park (ETTP) (Reference 2). Uranium enrichment operations at K-31 and K-33 were active from the 1950s until 1985. Reindustrialization of the ETTP began in 1996 by DOE in cooperation with the Community Reuse Organization of East Tennessee in preparation for conversion of the site to a private sector industrial park. Today, almost 2,000 acres, along with major site infrastructure, have been or will be transferred for economic development. Another 3,000 acres have been placed in a conservation easement for public recreational use, and more than 100 acres have been set aside for historic preservation as part of the Manhattan Project National Historical Park. The Manhattan Project National Historical Park will be designed to honor and share the stories of those that built and operated the site (Reference 3).

The site boundaries encompass approximately 185 acres (74.8 hectares). The center point of the proposed Hermes 2 reactor facility (facility) has the following coordinates:

<u>Latitude and Longitude</u>

- North 35° 56’ 19.1”
- West 84° 24’ 11.6”

<u>Universal Transverse Mercator Coordinates – (meters)</u>

- North 3,980,259
- East 734,248

<u>Roane County State Plane Coordinates – (meters)</u>

- North 744,179
- East 179,172

As shown in Figure 2.1-2, the site is adjacent to Poplar Creek and 0.4 mile (0.6 kilometer [km]) from the Clinch River arm of the Watts Bar Reservoir. Poplar Creek is a tributary of the Clinch River arm of the Watts Bar Reservoir. Figure 2.1-2 shows prominent natural and man-made features within approximately 5 miles (8 km) of the site. The distance and direction from the site's center point of the safety-related area to major nearby features are as follows (Reference 1):

- Oak Ridge National Laboratory (ORNL) (4.8 miles)
- Clinch River Nuclear Site (3.6 miles)
- Interstate 40 (4.9 miles)
- Railroads (Heritage Railroad Corporation [1,132 feet], Norfolk Southern Railway Company [3.5 miles])
- Poplar Creek (0.2 miles)
- Clinch River arm of the Watts Bar Reservoir (0.4 miles)
- Duct Island (0.6 miles)

The region in which the site is located is known as the Great Valley of East Tennessee, which is comprised of valleys at elevations of around 800 feet above mean sea level and ridges around 1,000 feet above mean sea level or higher. The area is situated between the Cumberland Mountains, approximately 23.5 miles (38 km) to the northwest, and the Great Smoky Mountains approximately 31.6 miles (51 km) to the southeast. The area is characterized by forest, streams and reservoirs fragmented by urban development and agriculture. Part of the Ridge and Valley Province of East Tennessee, the site is located to the west of Poplar Creek on a gently rolling valley between Black Oak Ridge and Pine Ridge (Reference 4). Knoxville Tennessee is the nearest major metropolitan area, located approximately 25 miles (40 km) east of the site (Reference 5). Major transportation corridors in the region are Interstate 40 (I-40, the major east-west interstate highway located south of the site) and Interstate 75 (I-75, which travels in a north-south direction). I-40 and I-75 intersect approximately 9.5 miles (15.3 km) east-southeast of the site.

Outside the City of Oak Ridge, which includes the ORNL and the site, the surrounding land uses are generally residential and agricultural in nature, used primarily for single-family residences and small farms. Popular recreational activities in the area include fishing, hunting, boating, water skiing, and swimming (Reference 5).

Figure 2.1-2 shows the highways, railways, and waterways that traverse or are close to the site.

Figure 2.1-3 illustrates the topography within the vicinity of the site. The finished site grade elevation is approximately 765 feet North American Vertical Datum of 1988 (NAVD 88). The site and adjacent ground within a radius of approximately 0.5 miles is flat. Topographic elevations range from approximately 1,525 feet (464.8 meters) NAVD 88, to approximately 737 feet (224.6 meters) NAVD 88 to the east of the site. Therefore, the topography within a 5-mile radius ranges from approximately 28 feet below to approximately 760 feet above the site grade elevation (Reference 7).

The tallest buildings to be constructed at the site are the Reactor Buildings, which at their highest point are approximately 90 feet above the site grade level.

2.1.1.2 Boundary and Zone Area Maps

The reactors are located within the Reactor Buildings shown on Figure 2.1-3. Figure 2.1-3 shows the site and exclusion area boundary (EAB) with respect to the reactor locations. In accordance with 10 CFR 20.1003, the site boundary defines the area owned, leased, or controlled by the licensee. In accordance with 10 CFR 100.3 and ANSI/ANS-15.16-2015 (R2020), the operations boundary (or EAB) is the area within the site boundary where the reactor site management has direct authority over all activities including exclusion or removal of personnel and property from the area.

The EAB is coincident to the site boundary.

The Low Population Zone (LPZ) is 800 meters from the reactor buildings as shown in Figure 2.1-3. The Emergency Planning Zone (EPZ) boundary is set coincident to the site boundary. The EPZ is an area used for emergency activities in the event of an emergency (Reference 6). The doses at the EPZ are below the Environmental Protection Agency (EPA) Protective Action Guide (PAG) Manual guidelines for protective action, as recommended by ANSI/ANS-15.16-2015 (R2020) and pursuant to Regulatory Guide 2.6, "Emergency Planning for Research and Test Reactors." This approach is consistent with the allowance for a smaller EPZ in 10 CFR 50, Appendix E.I.3.

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 9, Reference 10):

- Beginning of the requested construction period (2026)
- Five years after the beginning of the requested construction period (2031)
- Approximate end of the requested license period (2040)

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 9). The population data is used in the environmental monitoring program discussed in Chapter 11.

2.1.2.1 Resident Population

The distribution of the resident population for the area within 5 miles (8 km) of the site is shown in Figures 2.1-4 to Figure 2.1-8. The maps illustrate town, city, and county boundaries.

Figure 2.1-4 shows the population by block group using the most recent and currently available decennial census year (2020) within the site. Figure 2.1-5 also shows the population as of 2020 decennial census but distributes the population 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 9). 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 geographic information system software known as 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, as illustrated in Table 2.1-1.

Figures 2.1-6, 2.1-7, and 2.1-8 show the population estimates for 2026 (the beginning of the construction period), 2031 (5 years later) and 2040 (approximate end of the license period). Projections are based on county estimates for Roane and Morgan Counties derived from the Boyd Center for

Business and Economic Research, Tennessee's state demographer (Reference 9). The basis of the projection method was the application of the historic annual growth rate sourced from the county projections to the 2020 decennial census block data. The growth (or loss) rate was determined by the state demographer. The same annual rates were then applied to the base year of 2020 for each county, which was the most recent decennial census data available, and projected forward for the years 2026, 2031, and 2040. The same rates were used to project population changes in each distance/direction segment in each county.

Tables 2.1-1 and 2.1-2 show the historical population for 2020 and the projected resident population for the years 2026, 2031, and 2040 that fall within the distance bands for Roane and Morgan counties (Reference 9, Reference 10).

As shown in Figure 2.1-2, the nearest permanent residence to the reactor is a residence located 0.7 miles away from the site boundary to the northwest. Figure 2.1-3 demonstrates that the nearest resident is outside the LPZ.

2.1.2.2 Transient Population

Transient populations are temporary or seasonal populations residing in the area, such as in lodging accommodations, dormitories, or classrooms on a college campus. According to the results of the Google Earth desktop research, there are no schools or lodging facilities within 5 miles (8 km) of the site. Thus, there are no transient populations in the area.

2.1.3 References

1. Environmental Systems Research Institute (ESRI), Tennessee Map. 2021.
2. U.S. Department of Energy (DOE) Oak Ridge Environmental Management Program, ETTP fact sheet. 2019. Retrieved from https://www.energy.gov/sites/default/files/2019/01/f58/ETTP%20fact%20sheet_0.pdf.
3. Oak Ridge Office of Environmental Management, East Tennessee Technology Park. Website: https://www.energy.gov/orem/cleanup-sites/east-tennessee-technology-park.
4. Parr, P.D, and Hughes, J.F., Oak Ridge Reservation Physical Characteristics and Natural Resources, Oak Ridge National Laboratory, ORNL/TM-2006/110. September 2006.
5. U.S. Department of Energy (DOE), "Environmental Monitoring Plan for the Oak Ridge Reservation," DOE/ORO--2227/R5. October 2012.
6. ANSI/ANS-15.16-2015(R2020), "Emergency Planning for Research Reactors."
7. U.S. Geological Survey (USGS), "Elevations for Site Buildings," 2021.
8. Not Used.
9. US Census Bureau, 2020 Census—Block Maps. 2020. Retrieved from https://www.census.gov/geographies/reference-maps/2020/geo/2020-census-block-maps.html. Accessed May 17, 2023.
10. Tennessee State Data Center, Boyd Center Population Projections, Population Projections for Tennessee Counties 2019-2070. October 22, 2019. Retrieved from https://tnsdc.utk.edu/estimates-and-projections/boyd-center-population-projections/. Accessed May 5, 2023.

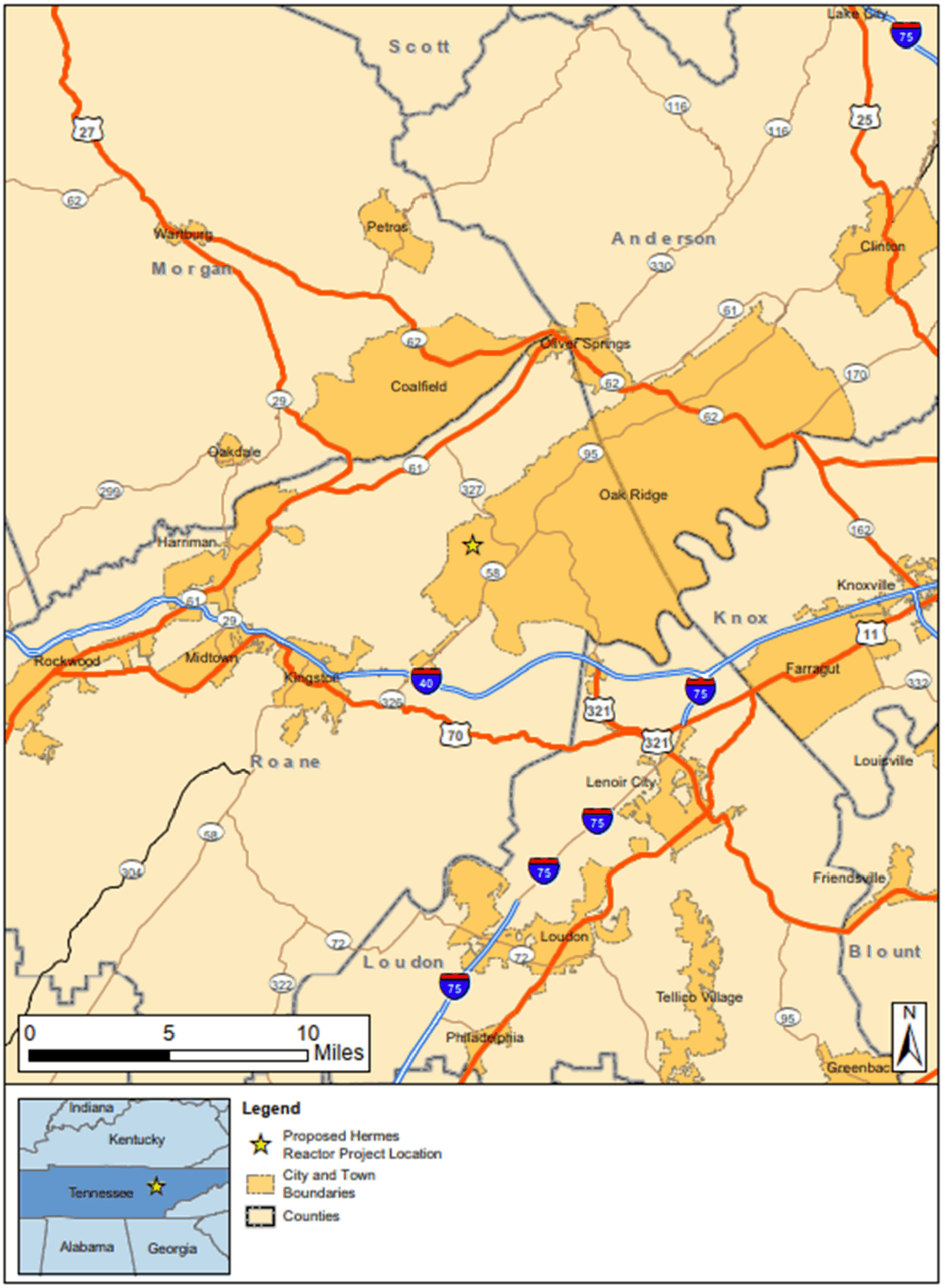
Table 2.1-1: Resident Population Distribution within 5 miles (8 km) of the Site in Roane County

Year	Distance Band (miles)					
	0–0.5	0.5–1	1–2	2–3	3–5	Total
2020	0	15	386	1,779	7,925	10,105
2026	0	16	402	1,854	8,257	10,529
2031	0	16	416	1,918	8,545	10,895
2040	0	17	443	2,040	9,088	11,587
Sources: Reference 9, Reference 10						

Table 2.1-2: Resident Population Distribution within 5 miles (8 km) of the Site in Morgan County

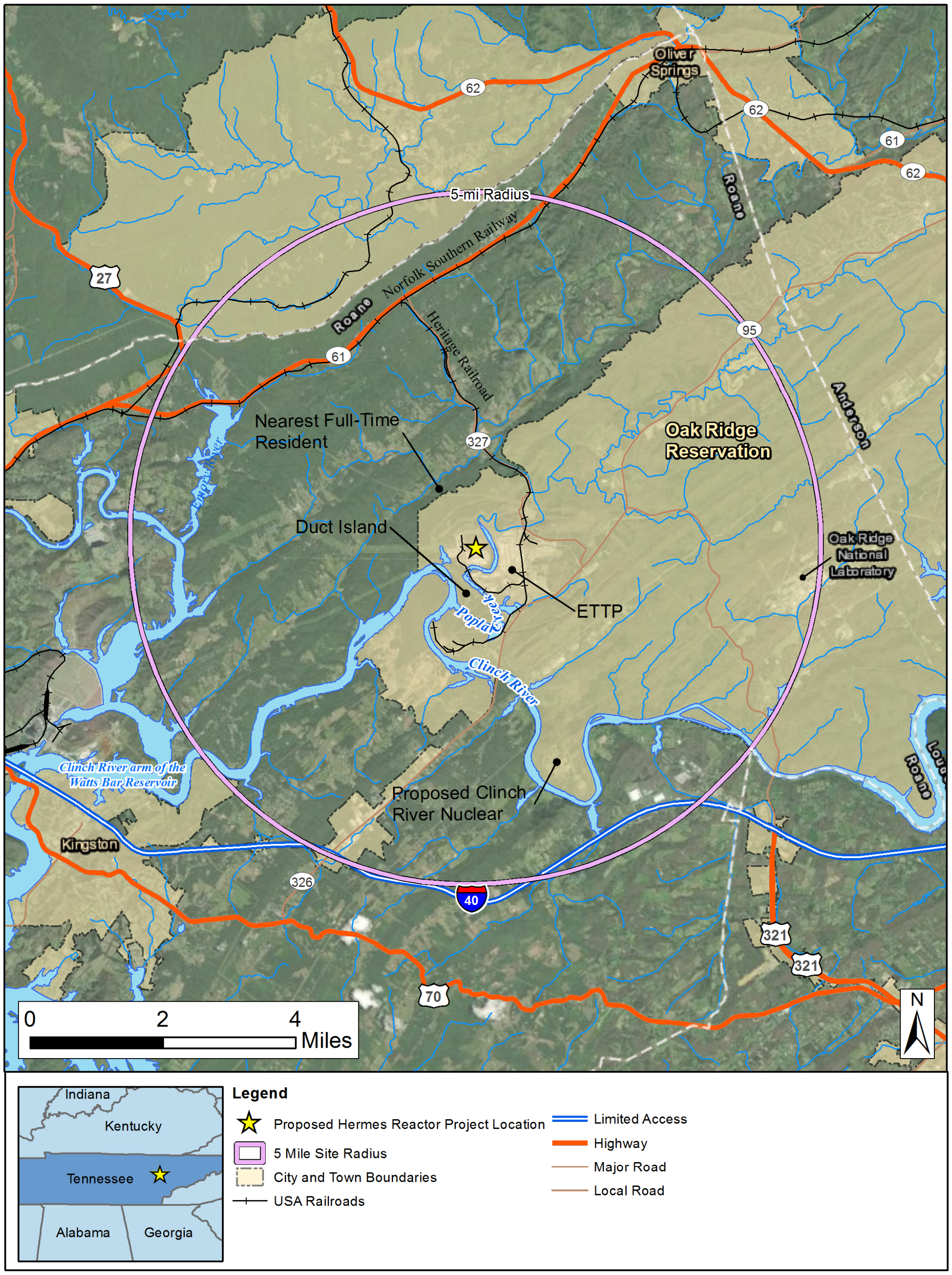
Year	Distance Band (miles)					
	0–0.5	0.5–1	1–2	2–3	3–5	Total
2010	0	0	0	0	381	381
2026	0	0	0	0	386	386
2031	0	0	0	0	390	390
2040	0	0	0	0	398	398
Sources: Reference 9, Reference 10						

Figure 2.1-1: Location of the Site



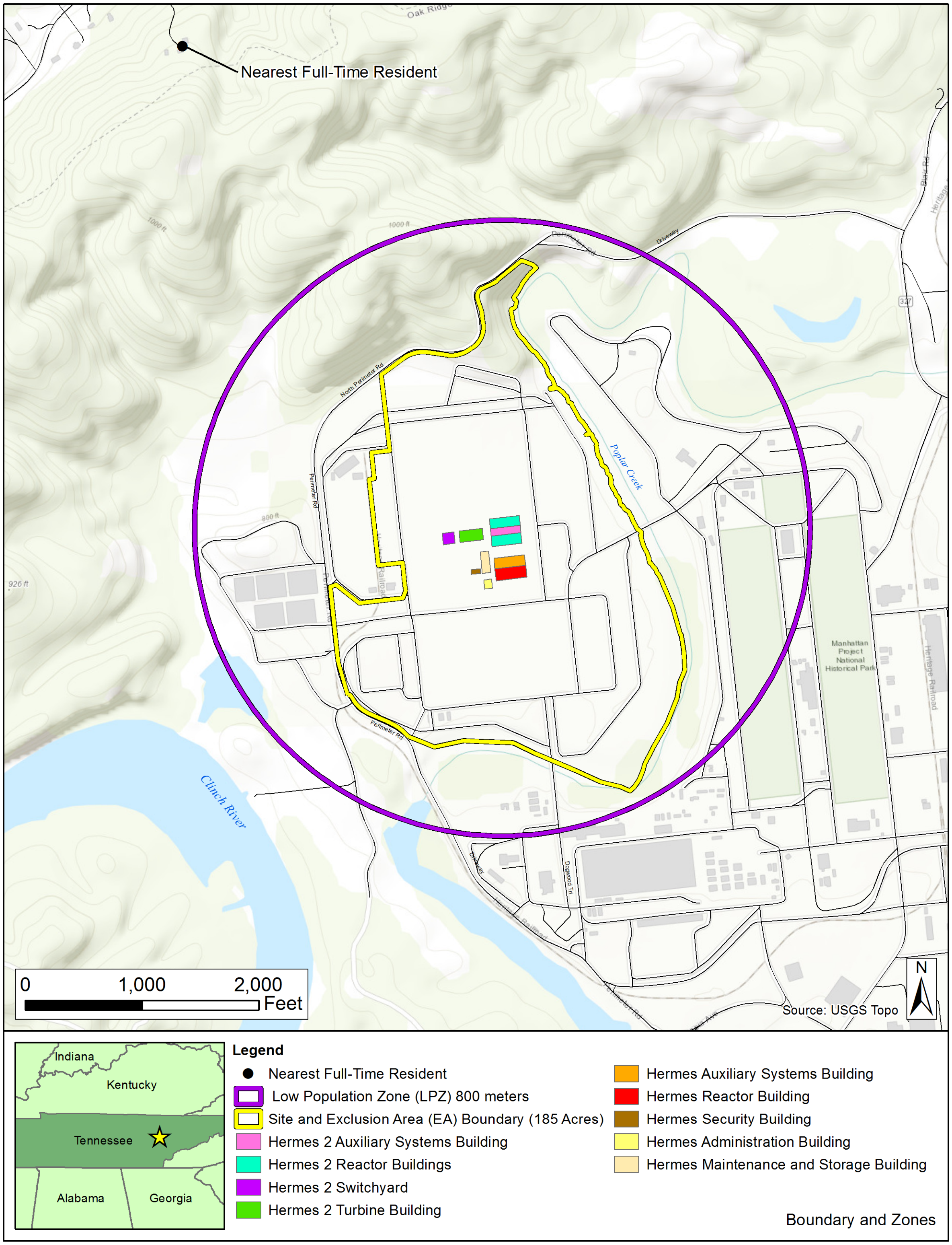
Source: Reference 1

Figure 2.1-2: Prominent Features in Site Area



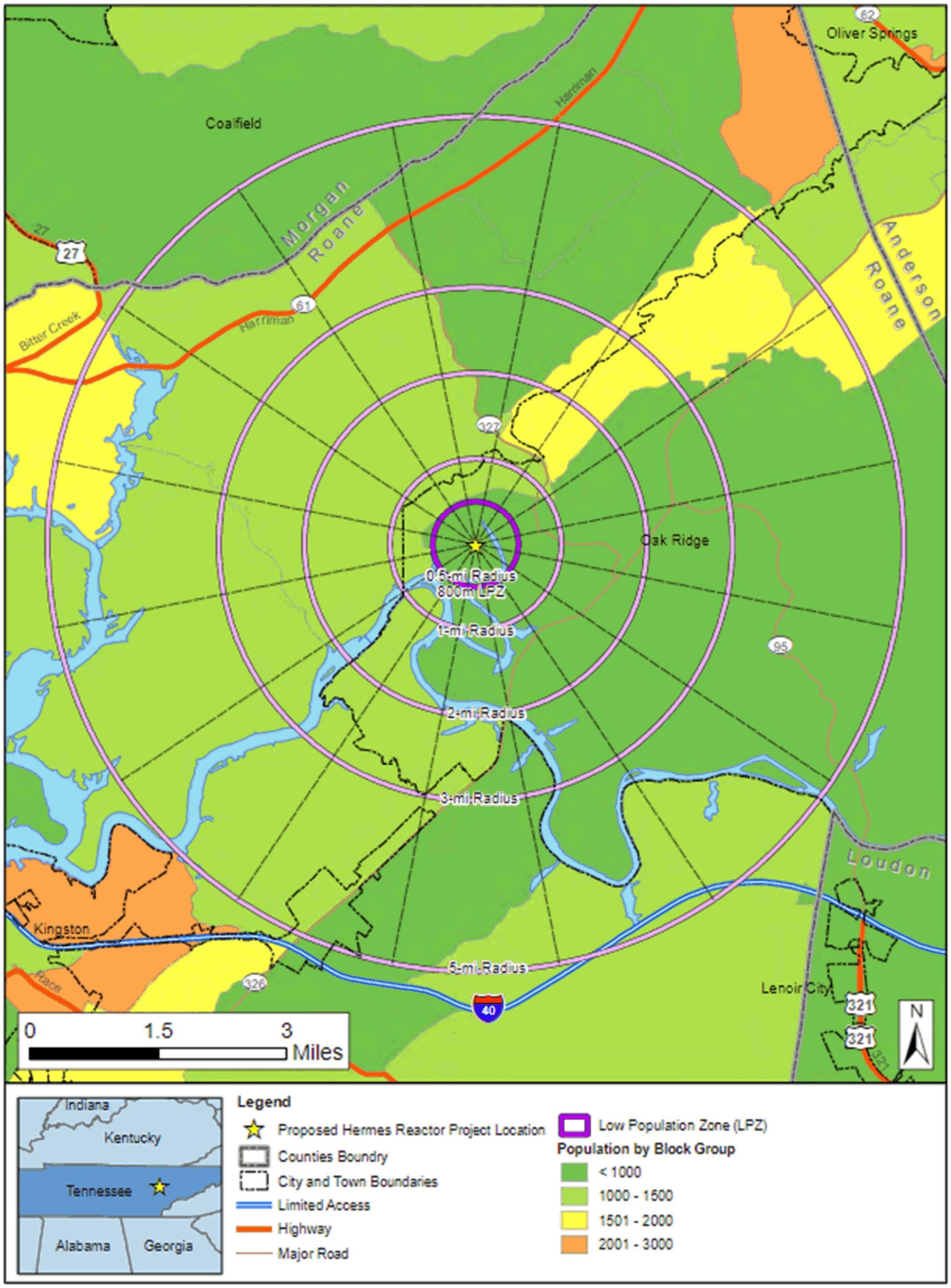
Source: Reference 1

Figure 2.1-3: Project Site Area and Zones Associated with the Facility



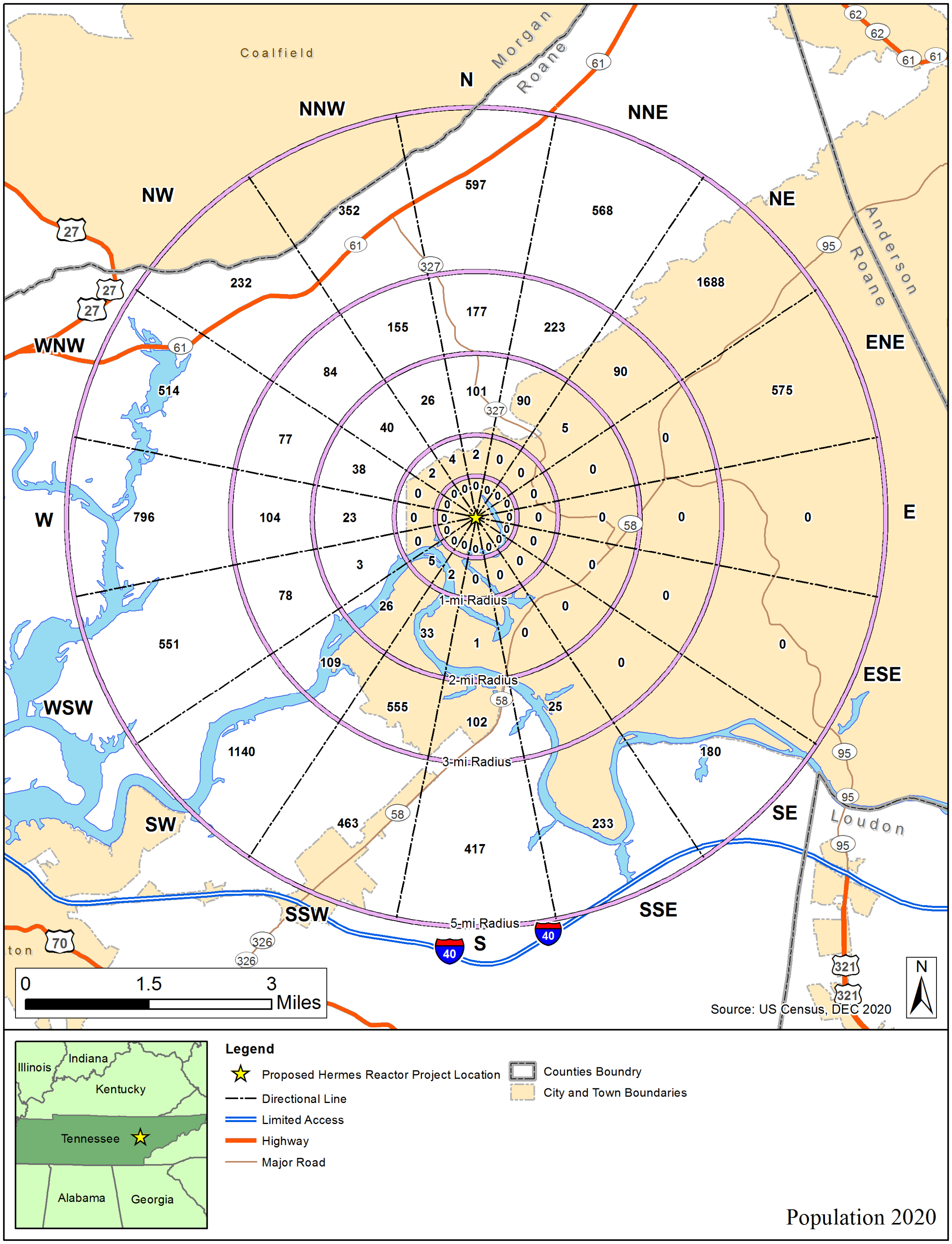
Source: Reference 1

Figure 2.1-4: Population Groupings within 5-miles (8-km) Radius



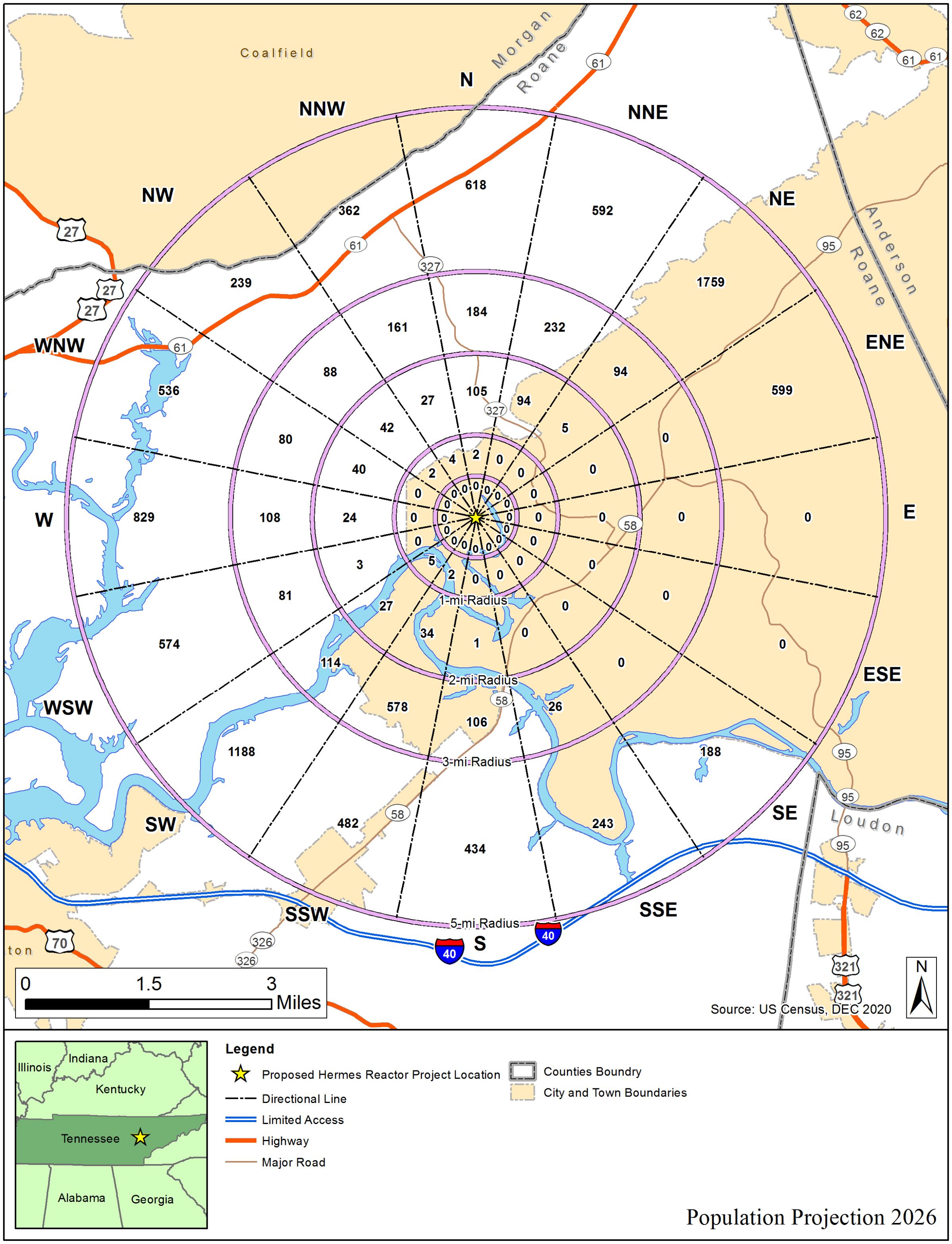
Sources: Reference 1, Reference 9

Figure 2.1-5: Resident Population Distribution - 2020



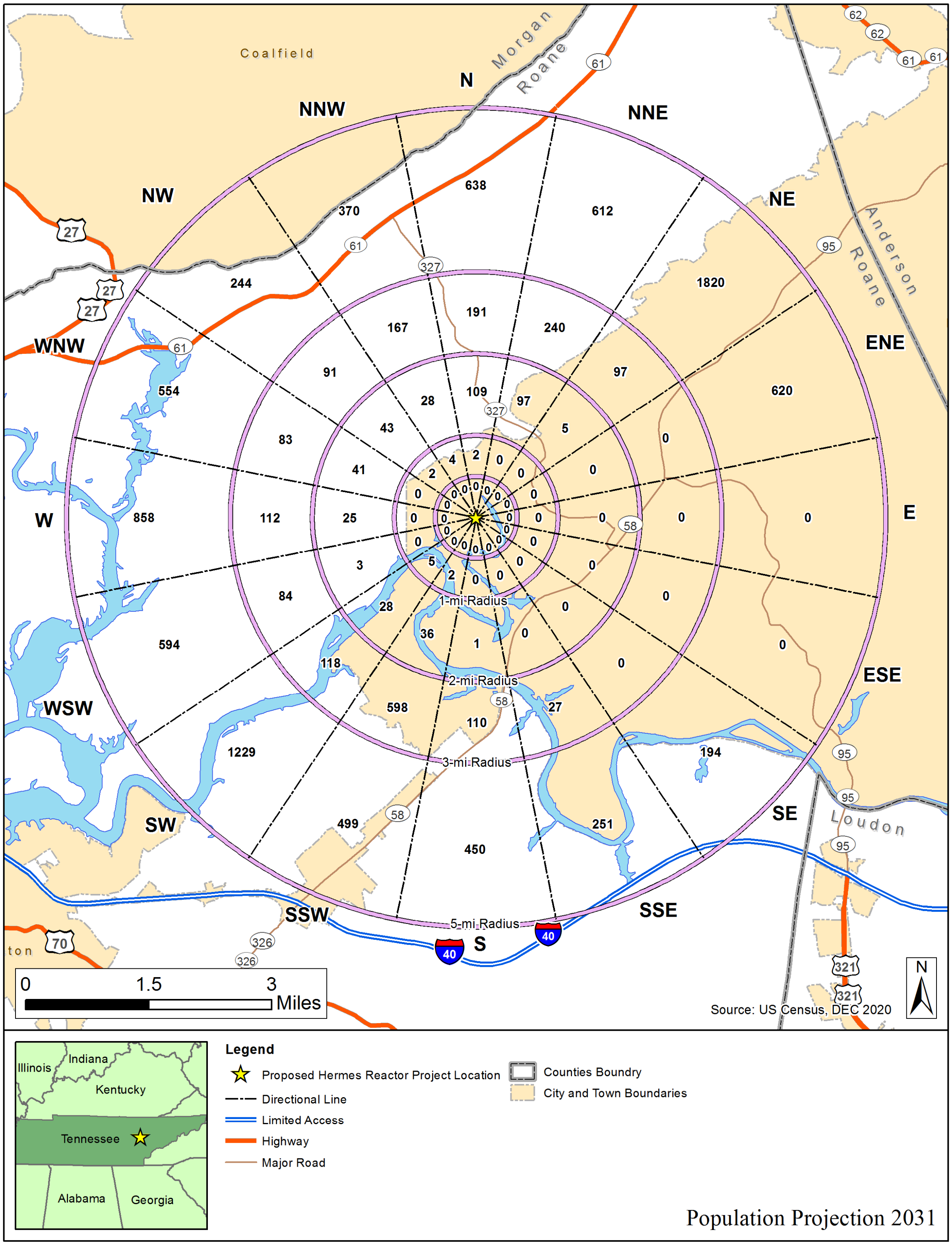
Sources: Reference 1, Reference 9

Figure 2.1-6: Resident Population Distribution – 2026



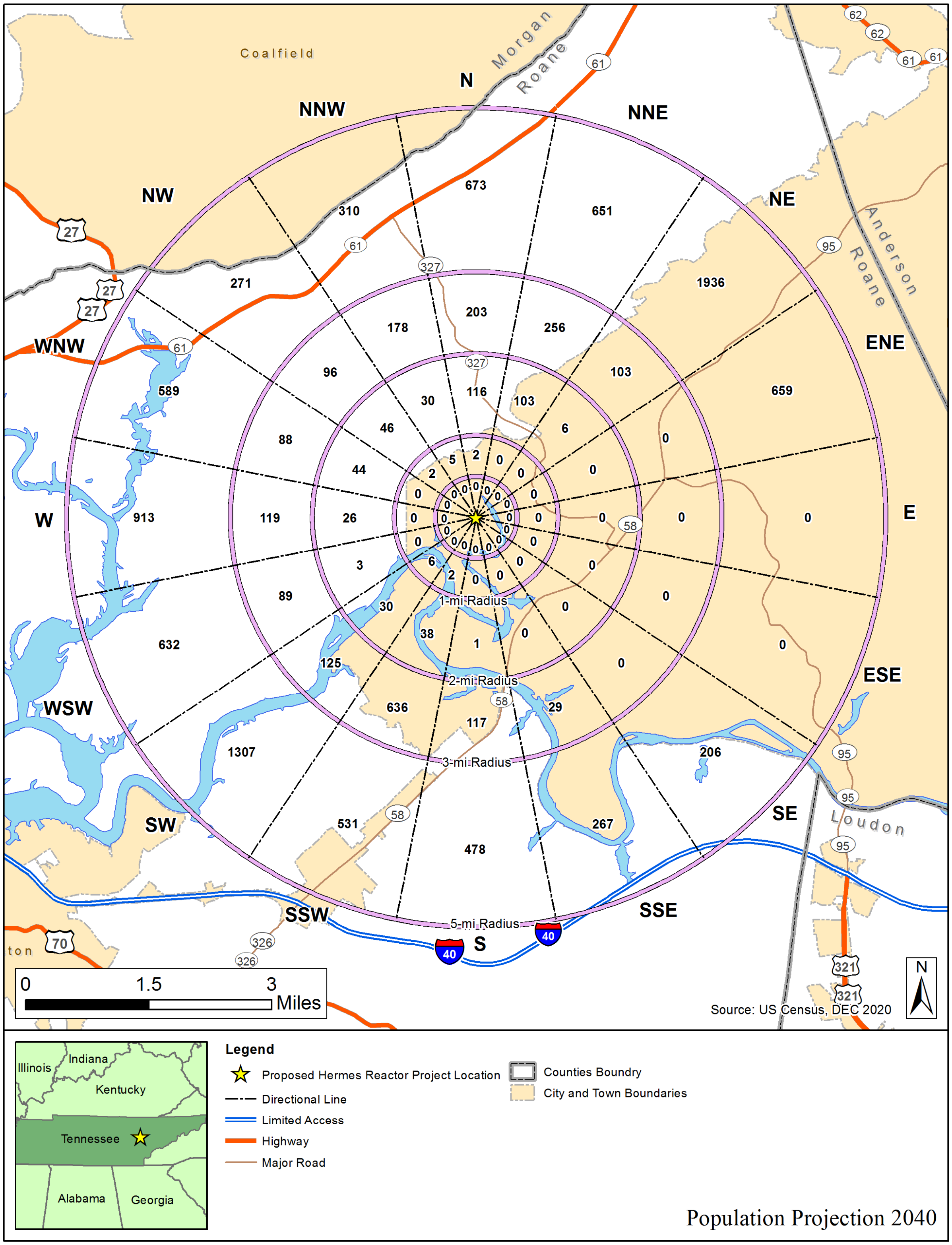
Sources: Reference 1, Reference 9, Reference 10

Figure 2.1-7: Resident Population Distribution – 2031



Sources: Reference 1, Reference 9, Reference 10

Figure 2.1-8: Resident Population Distribution – 2040



Sources: Reference 1, Reference 9, Reference 10

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY INSTALLATIONS

This section identifies and evaluates present and projected future industrial, transportation, and military installations and operations in the area around the site. This section considers the facilities and activities within 5 miles (8 km) of the reactor, consistent with the guidance of NUREG-1537. The reactor is located within the exclusion area of the site. Therefore, this section identifies facilities and activities within 5 mi (8 km) of the boundaries of the exclusion area and considers them in the evaluation of potential hazards. In addition, facilities and activities at greater distances were considered but none was analyzed in detail due to their insignificance with respect to accident impact on the facility.

2.2.1 Locations and Routes

An investigation of industrial, transportation, and military facilities within 5 miles (8 km) of the site was performed. Figure 2.2-1 shows the location of nearby facilities including industrial and transportation facilities within 5 miles (8 km) of the site, with the exception of airways. Figure 2.2-2 illustrates the airports, jet routes, and airway routes identified within 10 miles (16 km) of the site.

An evaluation of the identified transportation routes and pipelines within the 5-mile vicinity of the site identified one navigable waterway, one major highway, four major roads, one major rail line, one minor rail line, two natural gas pipelines, and one proposed airport for assessment. These notable nearby facilities are listed below:

- Clinch River arm of the Watts Bar Reservoir
- Interstate 40 (I-40)
- Tennessee State Highways 58 (TN 58), 61 (TN 61), 95 (TN 95), and 327 (TN 327)
- Norfolk Southern rail line north of the site
- Heritage Railroad Corporation Railway
- Two active natural gas transmission pipelines: East Tennessee Natural Gas Pipeline 1 (East) and Pipeline 2 (North)
- Proposed Kairos Power Hermes Facility
- Proposed Kairos Power Atlas Fuel Fabrication Facility
- Proposed TRISO-X Fuel Facility
- Proposed General Aviation Airport at the East Tennessee Technology Park (ETTP) Heritage Center
- Proposed Coqui Pharma Facility
- Proposed Clinch River Nuclear Site

There are no chemical plants, refineries, mining/quarrying, or military facilities within 5 miles (8 km) of the site. However, including the features listed above, a total of 19 existing or proposed features and facilities that require consideration with respect to possible adverse effects on the reactor are identified in Figures 2.2-1 and 2.2-2. Table 2.2-1 provides a description of these features and facilities, including their primary functions and major products.

In addition, an analysis of the potential hazards to the facility due to chemical storage both on and off the site is presented in Section 2.2.3.

2.2.1.1 Description of Pipelines

Enbridge operates two East Tennessee Natural Gas pipelines within 5 miles of the site. Pipeline 1, located east of the site, has a 6-inch diameter and was constructed in 1957. Pipeline 2, located north of the site, has a 22-inch diameter and was constructed in 1950. Both pipelines operate at a maximum allowable operating pressure of 720 pound-force per square inch gage (psig) and are buried to a minimum depth of 3 feet (36 inches) below grade. The pipelines have various isolation (gate) valves located along the pipeline route that can be reached and operated within one hour of notification. The pipeline operating parameters are obtained from Spectra Energy, which was the parent company of East Tennessee Natural Gas (Reference 1). The closest branch of either pipeline is Pipeline 2, which is approximately 1 mile (1.6 km) north-northeast of the site. Figure 2.2-1 illustrates the natural gas pipelines located within 5 miles (8 km) of the site. These pipelines are evaluated further as hazards in Section 2.2.3 (Reference 2, Reference 3).

2.2.1.2 Description of Waterways

The Clinch River flows southwest from Tazewell, Virginia, through the Great Appalachian Valley to Kingston, Tennessee just west of Knoxville, where it joins the Tennessee River/Watts Bar Reservoir. Significant waterborne transport in the site vicinity is only possible on the Clinch River arm of the Watts Bar Reservoir. Annual waterborne commerce data compiled by the U.S. Army Corps of Engineers (USACE) Waterborne Commerce Statistics Center, for the period of 2001 to 2015, indicates that there were very few shipping cargos on the Clinch River, with no transport of hazardous materials (e.g., chemicals and related products, petroleum, ordnance) that could pose a threat to operations at the site (Reference 4, Reference 5). These shipment cargos were classified as machinery (not electric), fabricated metal products, limestone, and wood in the rough. Therefore, 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. White Oak Dam is located approximately 5 miles north at the terminus of White Oak Creek into the Clinch River. There are no materials stored at the facility, and therefore the dam has been removed from further evaluation.

2.2.1.3 Description of Highways

The most significant highway near the site is I-40, which runs roughly east-west on the opposite side of the Clinch River arm of the Watts Bar Reservoir. At its closest point, I-40 is approximately 4.9 miles (7.9 km) from the site. According to the Tennessee Department of Transportation, the annual average daily vehicle count just east of the I-40 and TN 58 interchange (approximately 4.5 miles south of the site) was 44,470 vehicles in 2018 (Reference 6).

Other larger roads near the site include TN 58, TN 61, TN 95, and TN 327, the closest of which is TN 327, located approximately 0.6 mile (1 km) east of the site. The intersection of TN 327 and TN 58 lies approximately 1.3 miles (2.1 km) east of the site. According to the Tennessee Department of Transportation, the annual average daily vehicle count at TN 327 west of the intersection with TN 58 was 2,485 in 2018 (Reference 7), and the annual average daily vehicle count at TN 58 north of the intersection with TN 327 was 12,641 in 2018 (Reference 8).

I-40 and TN 58 were identified as those roads within 5 miles (8 km) of the site on which chemicals may be transported. These are considered further in Section 2.2.3.

2.2.1.4 Description of Railroads

The nearest major rail line to the site is operated by Norfolk Southern and runs roughly northeast from Harriman, Tennessee, parallel to TN 61 toward Clinton, Tennessee. At closest approach, this line is approximately 3.3 miles (5.3 km) north-northwest of the site (Figure 2.1-2). A second major rail line operated by Norfolk Southern lies south of the site and runs roughly northeast through Loudon, Tennessee, to Knoxville. At closest approach, this line is approximately 12 miles (19.3 km) from the site. Due to the large distances from these lines to the site and the complex intervening terrain (wooded ridges and valley), accident scenarios on these lines are not evaluated further (Reference 9).

The nearest minor rail line is owned and operated by the Energy*Solutions*, LLC, doing business as Heritage Railroad Corporation for industrial uses. The railroad runs from the Heritage Center Industrial Park to the Blair Interchange on the Norfolk Southern main line north of the site, a distance of approximately 11.5 miles (18.5 km). Within the ETTP, the main line serves the intermodal transfer area operated by Energy*Solutions* and a rail car repair area operated by East Tennessee Rail Car Company. During fiscal year 2020, approximately 121 railcars were moved over the line. In May 2016, Energy*Solutions* directed the Southern Appalachian Railroad Museum to stop operations of their excursion trains on the Heritage Railroad Line due to liability concerns in case of an accident. Materials transported on this rail line consist mostly of solid, low-level radioactive wastes, which do not pose a significant threat to the site due to their physical properties. Solids have a vapor pressure sufficiently low such that the formation of a vapor cloud is not likely. That is, the air dispersion hazard of the material is not a likely exposure route nor is the solid material considered explosive. These wastes do not pose a significant threat to the site due to the physical properties of the waste; therefore, accidents from the transport of hazardous materials in the vicinity of the site by rail are not considered further (Reference 9).

2.2.2 Air Traffic

2.2.2.1 Identification of Air Traffic Near the Site

There are no existing commercial airports located within 10 miles (16 km) of the site. A general aviation airport located approximately 1 mile to the southeast has been proposed by the Oak Ridge City Council (Reference 10). The proposed airport is scheduled to begin construction in 2023 and reach operational status by 2025. The runway of the proposed airport would be oriented such that aircraft would not depart or approach on a trajectory over the site. Figure 2.2-2 shows the location of the jetways and airways.

A 2016 Environmental Assessment (EA) prepared by the DOE for the property transfer for the development of the general aviation airport (Reference 11) provides annual operation forecasts for the proposed airport. The EA describes the forecasted operations in terms of local and itinerant operations, with an operation consisting of a single event, either a take-off or landing. Local operations are those arrivals or departures performed by aircraft remaining within the airport traffic pattern or those that occur within sight of the airport (e.g., training activity, flight instruction). Itinerant operations are arrivals and departures that do not remain within the airport traffic pattern (e.g., flights originating or destined for another airport). The average annual local and itinerant operations estimates presented in the EA are 25,472 and 24,241, respectively, for a total of 49,713 annual operations (Reference 11). These estimates were based on reported Federal Aviation Administration (FAA) records from other airports in the region. The EA also estimates that the operations would consist of approximately 2,486 fixed-wing turbine aircraft operations, 45,736 fixed-wing piston aircraft operations, and 1,491 helicopter operations (Reference 11). Given the proximity to the site, the proposed airport is further evaluated in Section 2.2.2.3.

Liles Airport in Harriman, Tennessee is an inactive historic airport within 10 miles (16 km) of the site with no visible facilities or runway, and was eliminated from further evaluation. There are also several private airfields outside the 5-miles (8-km) radius that were eliminated from further evaluation.

No military airports or training routes are located within 10 miles (16 km) of the facility and, therefore, military airports were eliminated from further evaluation consistent with NUREG-1537 Section 2.2.2.

Two federal airways are located within 10 miles (16 km) of the facility. The centerline of Jet route J46 is approximately 0.9 miles (1.5 km) north of the facility, and the centerline of Airway V16 is approximately 6.2 miles (10 km) south of the facility, measured as the distance from the center of the facility to the nearest edge of the airway (Reference 12). Federal Airways and Jetways are 8 nautical miles (approximately 9.2 statute miles) wide (Reference 13), and distance was measured from the centerline for mapping purposes. Table 2.2-5 provides further description of the two airways. 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 mi from the facility, NUREG-0800 Section 3.5.1.6 allows the airway to be screened out with no further evaluation. Each of the two federal airways located within 10 miles (16 km) of the facility were identified as having an edge of the airway within 2 statute miles of the facility. These airways were further evaluated in Section 2.2.2.2. Figure 2.2-2 identifies the centerline of federal airways within 10 miles (16 km) of the facility. NUREG-0800 is applicable to power reactors only and is used here merely as guidance in the absence of specific guidance in NUREG-1537.

2.2.2.2 Evaluation of Airway Hazards

The 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 14) and utilizes crash rates for non-airport operations.

The non-airport crash impact frequency evaluation is determined from using the following "four factor formula" (Reference 14):

$$F_j = N_j \cdot P_j \cdot f_j(x,y) \cdot A_j \qquad \text{(Equation 2.2-1)}$$

Where:

F_j	=	crash impact frequency
j	=	each type of aircraft suggested in the DOE Standard
$N_j \cdot P_j$	=	expected number of in-flight crashes peryear
$f_j(x,y)$	=	probability, given a crash, that the crash occurs in a 1-square-mile area surrounding the facility
A_j	=	effective area of the facility in units of square miles

Tables B-14 and B-15 of DOE-STD-3014-2006 provide $N_jP_jf_j(x,y)$ 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). Oak Ridge National Laboratory $N_jP_jf_j(x,y)$ values are used for the facility and are provided in Table 2.2-6.

The effective facility area (A_j) for the safety-related structures of the site depends on the length, width, and height of the facility, as well as the aircraft’s wingspan, skid distance, and impact angle as explained below (Reference 14):

$$A_j = A_f + A_s \qquad \text{(Equation 2.2-2)}$$

Where:

$$A_f = (WS + R) \cdot H \cdot \cot\Phi + (2 \cdot L \cdot W \cdot WS / R) + (L \cdot W) \qquad \text{(Equation 2.2-3)}$$

And:

$$A_s = (WS + R) \cdot S \quad \text{(Equation 2.2-4)}$$

Where:

A_f	=	effective fly-in area
A_s	=	effective skid area
WS	=	aircraft wingspan (Table 2.2-6)
R	=	length of the diagonal of the facility = $(L^2 + W^2)^{0.5}$
H	=	facility height, facility-specific
cotΦ	=	mean of the cotangent of the aircraft impact angle (Table 2.2-6)
L	=	length of facility, facility-specific
W	=	width of facility, facility-specific
S	=	aircraft skid distance (mean value) (Table 2.2-6)

The total effective area (A_j) for the safety-related structures of the facility is calculated. Dimensions of the facility used in the analysis include two safety-related structures each with a width of 50 feet, a length of 170 feet, and a height of 42 feet.

The calculated effective areas for the six aircraft types are provided in Table 2.2-7.

The airway crash impact probabilities for small non-military aircraft (i.e., general aviation and air taxi), large non-military aircraft (i.e., air carriers), and military aircraft (i.e., small aircraft and large aircraft) from airways are provided in Table 2.2-9.

2.2.2.3 Evaluation of Airport Hazards and Helicopter Operations

Given the approximate location of the proposed airport and the proposed number and type of operations described in Section 2.2.2.1, the crash impact frequency from the proposed nearby airport operations is calculated in accordance with Equation 2.2-1 using methodologies outlined in DOE Standard DOE-STD- 3014-2006 for airport operations (Reference 14).

The estimated annual number of landings and takeoffs (N_j) was obtained from the 2016 DOE EA for the property transfer for the development of the proposed general aviation airport (Reference 11), excluding helicopter operations (N_j = 24,111). All flights were assumed to be general aviation fixed-wing operations based on information provided in Table 2.5 in the 2016 DOE EA (Reference 11). Tables B-4 and B-5 of DOE-STD-3014-2006 provides f(x,y) takeoff and landing values for near-airport conditions for general aviation aircraft (Reference 14). The x and y distances from the Reactor Building to the midpoint of proposed airport are estimated to be 0.7 miles and 1.2 miles, respectively. Because the runway takeoff and landing directions are not yet established, the x and y distances are evaluated as both positive and negative (±), and the most conservative f(x,y) values are obtained from Tables B-4 and B-5. Table B-1 of DOE-STD-3014-2006 provides aircraft crash rate (P_j) values for takeoff and landing (Reference 14). The effective area of the facility (A_j) is calculated as described above in Equations 2.2-2 through 2.2-4. Applicable airport inputs are provided in Table 2.2-8. The crash impact frequency for the fixed-wing general aviation operations are provided in Table 2.2-9.

An estimated 3 percent of flight operations are expected to be attributed to helicopter operations at the proposed airport (Reference 11). 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" (Reference 14). Using available information for the proposed airport from the 2016 DOE EA and guidance from the DOE-STD-3014-2006, the helicopter impact frequency evaluation is determined by the following formula:

$$F_H = N_H \cdot P_H \cdot (2/L_H) \cdot A_H \qquad \text{(Equation 2.2-5)}$$

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

Table 2.5 in the 2016 DOE EA provides estimates of the expected number of helicopters at the proposed airport (Reference 11). Table B-1 in DOE-STD-3014-2006 provides aircraft crash rate values, and the helicopter effective area is calculated in the same manner as Equations 2.2-2 through 2.2-4 (Reference 14). The average flight distance of 37 miles is selected based on the generic flight length provided in Table B-43 of DOE-STD-3014-2006 (Reference 14). Applicable inputs are provided in Table 2.2-8. The crash impact frequency for the helicopter operations are provided in Table 2.2-9.

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. NUREG-1537 does, however, state 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 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 14). The total crash frequency for all airway, helicopter operations, airport takeoff operations, and airport landing operations of 8.00E-05 exceeds this criterion. Excluding near-airport, the impact risk from jetway/airways of 1.39E-05 is greater than the screening criterion of 1.00E-06. Additionally, the proposed nearby airport and helicopter operations crash frequency is 6.61E-05 and exceeds the 1.00E-06 screening criterion. In all cases, the crash frequency criterion is exceeded due to small, non-military aircraft from general aviation or helicopter operations. The risk from large commercial aviation aircraft is well below the screening criterion. As a result, the safety-related portion of the Reactor Building structure will be designed to withstand the impact of a small non-military general aviation aircraft as described in Section 3.5. The maximum crash frequency for all aircraft type and aircraft operations are provided in Table 2.2-9.

2.2.3 Analysis of Potential Accidents at Facilities

Each of the nineteen nearby facilities listed in Table 2.2-1 is considered with respect to possible effects on the reactor facility that could precipitate an event. It was determined that ten of the nearby facilities do not have a significant potential to affect the facility. Table 2.2-2 lists the nearby facilities that were concluded not to affect the reactor facility with a brief description of the basis for that finding. The remaining facilities are evaluated in this section. Six of these facilities (i.e., the Kairos Power Hermes

Facility, Kairos Power Atlas Fuel Fabrication Facility, TRISO-X Fuel Facility, Clinch River Nuclear Site, Coqui Pharmaceutical, and the regional airport) are currently proposed and not yet under construction.

The PSAR for the proposed Kairos Power Hermes Facility has demonstrated that the off-site radiological impacts at the site during routine operations and severe accidents would be within regulatory limits (Reference 33). The Hermes facility would include storage of approximately 21,555 gallons of diesel fuel in an onsite fuel tank for the standby diesel generator. The Hermes facility will have an inventory of 40,000 pounds of low-pressure, molten salt coolant i.e., Li_2BeF_4 (Flibe). The location and quantities of all chemicals that would be stored at the Kairos Power Hermes Facility have not yet been determined or finalized. The explosion analysis of the Hermes onsite fuel tank is further discussed in Section 2.2.3.1. The hazard due to the release of radioactive material from the Kairos Power Hermes Facility, as a result of normal operations or an unanticipated event, is not expected to affect the safety of the facility due to the low reported doses. Smoke detectors, radiation detectors, and associated control equipment will be installed at various plant locations as necessary to provide the appropriate operation of the systems. Radiation monitoring of the control room environments will be provided by the radiation monitoring system.

The Early Site Permit Application (ESPA), Part 2, Site Safety Analysis Report (SSAR) for the Clinch River Nuclear Site has demonstrated that the off-site radiological impacts from one or more nuclear reactors at the site during routine operations and severe accidents would be within regulatory limits (Reference 9). Furthermore, while the site would be within the Low Population Zone (LPZ) for Clinch River Nuclear Site reactor(s), the site would be outside the Clinch River Nuclear Site Emergency Planning Zone (EPZ) (Reference 9).

The Coqui Pharmaceutical site is excluded from the discussion in the following sections as there is currently not enough information available for an analysis. However, the radiological effects from the radiopharmaceutical production facility at the Coqui Pharmaceutical site would be within regulatory offsite dose limits for routine operations and accidents. Additionally, the operations are expected to be similar to the SHINE Medical Technologies which received a construction permit from the NRC in 2016 for a radioisotope production facility located in Janesville, Wisconsin. SHINE Medical Technologies demonstrated in its PSAR that releases of onsite chemicals were not a hazard to personnel in the facility control room (Reference 18). As such, chemicals stored onsite at the Coqui Pharmaceutical site would similarly not be expected to have an impact on nearby facilities, including the Hermes 2 facility.

The location and quantities of chemicals that would be stored nearby at the proposed Kairos Power Hermes Facility, the proposed Kairos Power Atlas Fuel Fabrication Facility, the proposed TRISO-X Fuel Facility, the proposed Clinch River Nuclear Site, the proposed Coqui Pharmaceutical Facility, and the proposed regional airport have not yet been determined. The effects of explosions, flammable vapor clouds, and toxic chemicals for onsite chemical storage at these nearby facilities will be reviewed in the operating license application and accounted for in the final design to the extent information regarding hazards from those nearby facilities are available.

The general aviation airport is discussed in Section 2.2.2.3 and is also evaluated on the basis of estimated information below.

The remaining nearby facilities with potential to affect the reactor facility are evaluated below. The potential effects of those nearby facilities in terms of design parameters or physical phenomena were identified considering guidance in Regulatory Guide 1.78, Revision 1 "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Regulatory Guide 1.91, Revision 2, "Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants," Regulatory Guide 4.7, Revision 3 "General Site

Suitability Criteria for Nuclear Power Stations," and NUREG-1537. Although the Regulatory Guides listed do not apply, they were consulted for applicable guidance in the absence of specific guidance in NUREG-1537.

The following event categories are considered: explosions, flammable vapor clouds (delayed ignition), toxic chemicals, and fires. The postulated events with the potential to result in a chemical release are analyzed at the following locations:

- Nearby transportation routes and nearby natural gas pipelines
- Nearby chemical and fuel storage facilities
- Chemicals stored or used on site

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, therefore, the guidance in Regulatory Guide 1.91, Revision 2, "Evaluations of Explosions Postulated to Occur at Nearby Facilities and on Transportation Routes Near Nuclear Power Plants," was considered in determining allowable (i.e., standoff) and actual distances of hazardous chemicals transported or stored.

Regulatory Guide 1.91 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. Regulatory Guide 1.91 defines this standoff distance by the relationship:

$$R \geq kW^{1/3} \qquad \text{(Equation 2.2-6)}$$

Where:

R = distance in feet

W = equivalent pounds of trinitrotoluene (TNT)

k = scaled distance constant at a given overpressure

The TNT mass equivalent, W, is determined following guidance in NUREG-1805 "Fire Dynamics Tools," where the heat of combustion of the chemical is compared to the heat of combustion of TNT (Reference 15). NUREG-1805 is for power reactors and is used as guidance in the absence of guidance in NUREG--1537.

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 Reactor Building, a probabilistic analysis is used to show that the rate of exposure to a peak positive incident overpressure in excess of 1 pound per square inch differential pressure (psid) (6.9 kPa) is less than 1E-06 per year, when based on conservative assumptions, or 1E-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. 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 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 fuel 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.

For unconfined explosions of propane, methane, or hydrogen, the yield factor is 3 percent from the Handbook of Chemical Hazard Analysis Procedures (Reference 16).

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 causes a pressure wave. The methodology for a BLEVE overpressure analysis is from the SFPE Handbook of Fire Protection Engineering (Reference 17).

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 released 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.

The chemicals of interest (Table 2.2-3 and Table 2.2-4) are evaluated to ascertain which hazardous materials have the potential to explode, thereby requiring further analysis.

The calculated rate of occurrence of severe consequences from a postulated external explosion accident at the site is less than 1E-06 occurrences per year, and qualitative arguments demonstrate that the realistic probability is lower. Regulatory Guide 1.91 cites 1 psi as a conservative value of peak positive incident overpressure, below which no significant damage would be expected. Safety-related areas at the site are designed to withstand a peak positive overpressure of at least 1 psi without loss of function or significant damage.

The analyses presented in this section demonstrate that a 1 psi peak positive overpressure would not be exceeded at the safety-related portion of the Reactor Building structure for any of the postulated explosion event scenarios. As a result, postulated explosion event scenarios would not result in severe consequences. Most of these assessments are based on evaluation of the results from the Clinch River Nuclear Site Early Site Permit ESPA, Part 2, SSAR (Reference 9), concluding that those results indicate sufficient site stand-off distances from applicable hazards.

Pipelines

A natural gas pipeline explosion occurring in the vicinity of the release point along either Pipeline 1 or Pipeline 2 (described in Section 2.2.1.1) would be unconfined. A damaging detonation from an unconfined natural gas release is not credible according to the NRC Safety Evaluation Report for

Robinson (former Hartsfield) Nuclear Power Plant (NUREG-0014). 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 an unconfined detonation. Thus, the dominant hazards from exterior natural gas pipelines are from the heat effect of thermal radiation from a sustained jet fire. Damaging explosions where the natural gas vapor cloud becomes confined either outside or by migration inside a building are not credible along Pipeline 1 or Pipeline 2.

However, the effects of an explosion along Pipeline 1 and Pipeline 2 were evaluated in the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9). The analysis concluded that the safe distances (the distance to where the peak incident pressure does not exceed 1 psi) for Pipeline 1 and Pipeline 2 were reported as 1,250 feet and 2,970 feet, respectively. These distances are less than the minimum separation distance from the site to the respective pipelines. Therefore, overpressures from an explosion from a rupture in either pipeline will not adversely affect the safe operation or shutdown of the reactor.

Waterway Traffic

Further analysis for potential impacts due to water transportation of hazardous materials is not necessary because the USACE reported an inconsequential amount of shipping on the Clinch River, and no transportation of hazardous materials. Additionally, there is no shipping on Poplar Creek (Reference 9).

Highways and Railways

Within 5 miles (8 km) of the site, there is one interstate highway (I-40), and four state highways (TN-58, TN-61, TN-95, and TN-327). The most significant highway near the site is I-40, which runs roughly east-west on the opposite side of the Clinch River arm of the Watts Bar Reservoir relative to the facility. At its closest point, I-40 is approximately 4.9 miles (7.9 km) from the facility. I-40 was identified as a road within 5 miles (8 km) of the site on which chemicals may be transported. It is also the road with the highest traffic volume per year. TN-58 is closer to the site (approximately 1.2 miles), and is used as a feeder highway to I-40. Typical hazardous materials transported on I-40 and TN-58 are provided in Table 2.2-4 (Reference 9).

The effects of an explosion along I-40 or TN-58 were evaluated in the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9). Table 2.2-10 provides the results of the explosion hazard evaluation for materials transported near the Clinch River Nuclear Site as well as explosive chemicals stored as ORNL as provided in the Clinch River Nuclear Site ESPA, Part 2, SSAR, (Reference 9) with distances measured to the site. The table demonstrates that the most limiting explosion, a 11,500-gallon butane tanker, has a minimum safe distance of 3,708 feet from I-40 or TN-58. The shortest distance from the site to either I-40 or TN-58 is 6,336 feet to TN-58.

There are two railways in proximity to the site. The first railway is run by Norfolk Southern, and transports significant traffic along two main lines located in the vicinity of the site. The first line is located approximately 3.3 miles (5.3 km) to the north-northwest, running from Harriman, Tennessee to Clinton, Tennessee. The second line runs from Loudon, Tennessee, to Knoxville, and is 12 miles (19.3 km) south of the site. Due to the large distances from these lines to the site and the complex intervening terrain (wooded ridges and valley), accident scenarios on these lines are not evaluated further (Reference 9). The other railway is operated by Energy*Solutions*, and is located on the adjacent property. This railway only transports, solid low-level radioactive waste, which by its nature is not explosive.

Nearby Facilities

Three facilities near the site were evaluated in the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9). They were the ORNL-Battelle, located approximately 5 miles (8 km) east of the site; Tennessee Valley Authority (TVA) Kingston Fossil Plant, located approximately 7 miles (11.2 km) southwest of the site; and the TVA Bull Run Fossil Plant, located approximately 15 miles (24.1 km) east of the site. The Clinch River Nuclear Site ESPA, Part 2, SSAR used a conservative TNT equivalency method to determine safe distances for the identified potentially explosive materials. All damaging overpressure safe distances were less than the minimum distance from the storage areas to the Clinch River Nuclear Site. With the exception of the TVA Kingston Fossil Plant, the site is farther away from the nearby facilities evaluated than the Clinch River Nuclear Site, and all distances between the nearby facilities and the site are greater than the minimum safe distances reported in the Clinch River Nuclear Site ESPA, Part 2, SSAR.

During operations, the Kairos Power Hermes facility would include storage of approximately 21,555 gallons of diesel fuel in an onsite fuel tank for the standby diesel generator. The 21,555-gallon diesel fuel storage vessel was assumed to contain fuel vapors at the upper UFL. This is conservative because the UFL 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. Using other conservative assumptions, an explosion could result in an overpressure safe distance of 500 feet (0.09 miles). The expected distance from the Kairos Power Hermes facility diesel tank to the Hermes 2 facility is assumed to be greater than or equal to 0.1 miles. Therefore, until the exact location of diesel fuel transportation and storage locations are determined, a potential explosion involving the Hermes facility fuel storage tank could result in overpressure that exceeds more than 1 psi, and the Hermes 2 facility's structural designs should account for this. The effects of explosions, flammable vapor clouds, and toxic chemicals from onsite chemical storage at the Kairos Power Hermes facility will be evaluated in the operating license application and accounted for in the final design.

Based on the proposed Oak Ridge General Aviation Airport EA, a fuel farm is proposed to be constructed operating two 10,000-gallon aboveground tanks for aviation fuels (Reference 11). These tanks would be of double-walled construction (or would employ some other means of secondary containment) and would be equipped with appropriate overfill and spill protection devices. Additionally, spill response equipment, such as absorbent booms and pads, would be made readily available. These tanks may also be required to contain vapor control devices depending on the actual monthly throughput of aviation fuels.

An evaluation of the explosive hazard from a jet fuel tank is provided in the SHINE Medical Technologies PSAR (Reference 18). The SHINE Medical Technologies PSAR evaluated tank of jet fuel containing 500,000 pounds or approximately 75,000 gallons. The tank was modeled using TNT equivalency methodologies to determine minimum separation distance. The model determined that the minimum separation distance from a tank containing 500,000 pounds of jet fuel was 0.22 miles. Combined, the two 10,000-gallon fuel tanks suggested in the Oak Ridge Airport EA (Reference 11) at the proposed Oak Ridge Airport would contain less fuel than the single tank modeled in the SHINE Medical Technologies PSAR, indicating that the minimum separation distance of 0.22 miles would be acceptable for the Oak Ridge Airport as well. Because the distance from the site to the Oak Ridge Airport would be greater than 0.22 miles, fuel stored at the airport would not present an explosive hazard with a potential of impacting the site.

Although unlikely, a boiling liquid expanding vapor explosion (BLEVE) could occur to one or both of the 10,000-gallon fuel tanks at the proposed Oak Ridge airport as a result of a high-temperature fire. Therefore, BLEVE analysis was conducted for a single 10,000-gallon tank of jet fuel (Jet A) and a single

10,000-gallon tank of aviation gasoline (AvGas) using the methods provided in Regulatory Guide 1.91. These analyses, which were conducted individually for each fuel type, indicated that the impact of the BLEVE would extend 0.40 miles for the jet fuel and 0.38 miles for the aviation fuel. Therefore, a BLEVE accident would not have an impact on the site approximately 1.1 miles from the proposed airport runway (the location of fuel tanks is not shown on proposed airport figures reviewed). It should be noted that the explosion impact distance is not linear and incorporating BLEVEs from both tanks at the same time would result in a safe distance range increase by approximately 1.23 times (or 0.49 miles for 20,000 gallons of jet fuel).

The locations and quantities of chemical that would be stored onsite at the Clinch River Nuclear Site were not evaluated in the ESPA, Part 2, SSAR (Reference 9). The ESPA, Part 2, SSAR noted that the effects on explosion events from onsite chemical storage would be evaluated in the combined license application for a future reactor project (Reference 9). Chemicals stored at the future reactor site would be maintained and stored in a manner that would be protective of on-site personnel and the on-site reactor(s). Furthermore, due to the distance from the Clinch River Nuclear Site to the site, a chemical explosion at the Clinch River Nuclear Site would not adversely affect the safe operation of the facility.

Onsite Chemicals

The location and quantities of chemicals stored at the site have not yet been determined. The effects of explosions from onsite chemical storage will be evaluated in the application for an Operating License.

2.2.3.2 Flammable Vapor Clouds

Flammable materials in the liquid or gaseous state is hypothetically postulated to form an unconfined vapor cloud that could drift toward the plant before ignition occurs. When a flammable chemical is released into the atmosphere and forms a vapor cloud, it disperses as it travels downwind. The parts of the cloud where the concentration is within the flammable range, between the lower and upper flammability limits, could burn if the cloud encounters an ignition source. The speed at which the flame front moves through the cloud determines whether it is a deflagration or a detonation. If the cloud burns fast enough to create a detonation, an explosive force may be generated.

Offsite chemicals evaluated in the Clinch River Nuclear Site ESPA, Part 2, SSAR are shown in Tables 2.2-3 and 2.2-4. No additional significant offsite sources of chemicals were identified. Therefore, due to the proximity of the Clinch River Nuclear Site to the site, the analysis presented in the Clinch River Nuclear Site ESPA, Part 2, SSAR is considered to be directly applicable to the analysis of the site. The chemicals listed in Table 2.2-3 were evaluated in the Clinch River Nuclear Site ESPA, Part 2, SSAR to ascertain which hazardous materials have the potential to form a flammable vapor cloud or vapor cloud explosion. For those chemicals with an identified flammability range, the Areal Locations of Hazardous Atmospheres (ALOHA), an air dispersion model, was used to determine the distances where the vapor cloud may exist between the UEL and the LEL, presenting the possibility of ignition and potential thermal radiation effects.

The Clinch River Nuclear Site ESPA, Part 2, SSAR concluded that none of the offsite chemicals presented an unacceptable hazard to the Clinch River Nuclear Site. As the distances between the Clinch River Nuclear Site and the offsite chemical storage locations are similar to the distances between the site and the same offsite chemical storage locations, the conclusion that the accidents involving the chemicals will not have an adverse effect on the Clinch River Nuclear Site is also applicable to the site.

Pipelines

As indicated previously, the East Tennessee Natural Gas has two natural gas pipelines within 5 miles (8 km) of the 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 towards 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 Clinch River Nuclear Site Early Site Permit ESPA, Part 2, SSAR (Reference 9) evaluated the distance a vapor cloud could travel to reach the lower flammability limit (LFL) boundary once a vapor cloud has formed from an accidental release of natural gas (as methane) from the pipeline using the ALOHA dispersion model. The LFL boundary distances for Pipeline 1 and Pipeline 2 were determined to be 477 feet and 1,401 feet, respectively. Furthermore, safe distances for vapor cloud exposure resulting in less than 1 psi of peak incident pressure were 1,575 feet and 4,572 feet for the respective pipelines. Lastly, a safe distance evaluating the heat flux of 5 kW/m^2 from a jet fire scenario concluded distances were 312 feet and 1,203 feet were required for Pipeline 1 and Pipeline 2, respectively. These distances are well short of the distances between these pipelines and the site. Therefore, a jet fire or a flammable vapor cloud ignition or explosion from either a rupture in the East Tennessee Natural Gas Pipeline 1 or East Tennessee Natural Gas Pipeline 2 is not expected to adversely affect the safe operation or shutdown of the reactor.

Waterway Traffic

Further analysis of potential impacts due to waterway transportation of hazardous materials is not necessary because the USACE reported an inconsequential amount of shipping on the Clinch River, and no transportation of hazardous materials. Additionally, there is no shipping on Poplar Creek (Reference 9).

Highways and Railways

As indicated previously, within 5 miles (8 km) of the site, there is one interstate highway (I-40) and four state highways (TN-58, TN-61, TN-95, and TN-327).

The effects of various materials transported on I-40 were evaluated as part of the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9) 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. The model results indicated that any plausible vapor cloud that could form and mix sufficiently following an incident on I-40 would be below the LFL boundary before reaching the Clinch River Nuclear Site. Butane results in the longest flammable plume of 1,827 feet. Furthermore, a vapor cloud explosion analysis was completed 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 and 618 feet for gasoline. These distances are less than the distance to the closest point of I-40 and TN-58 from the site. Therefore, a flammable vapor cloud formed from the release of chemicals transported along I-40 and TN-58 with the possibility of ignition or explosion would not adversely affect the safe operation or shutdown of the reactor.

The Norfolk Southern rail line north of the site is far greater than the distance to TN 58, and the above analysis is considered to bound incidents on the rail line. Energy*Solutions* does not transport hazardous materials other than low-level radioactive waste on its Heritage Railroad. Therefore, transportation of materials on the Heritage Railroad, which is closer to the site than TN 58, does not pose a risk from hazardous vapor clouds.

Onsite Chemicals

The location and quantities of chemicals stored at the site have not yet been determined. The effects of flammable vapor clouds from onsite chemical storage will be evaluated in the application for an Operating License.

Nearby Facilities

Three facilities were evaluated for flammable vapor clouds in the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9) using the ALOHA dispersion model to determine the safe distance for each postulated flammable vapor cloud scenario. The safe distance was measured as the distance to the outer edge of the LFL section of the vapor cloud. The facilities evaluated were ORNL-Battelle, located approximately 5 miles (8 km) east of the site; TVA Kingston Fossil Plant, located approximately 7 miles (11.2 km) southwest of the site; and the TVA Bull Run Fossil Plant, located approximately 15 miles (24.1 km) east of the site. Each material stored at the identified offsite facilities was evaluated with respect to its potential for formation of flammable/explosive vapor clouds. Each material was then dispositioned based on the identified physical properties of the material and whether a bounding analysis exists. The materials identified for further analysis with regard to the potential formation of flammable/explosive vapor clouds were: anhydrous ammonia, ethanol and gasoline (gasoline blend A and gasoline B).

Using the ALOHA dispersion model with conservative assumptions, the Clinch Nuclear Site ESPA, Part 2, SSAR (Reference 9) determined that the model results indicated that any plausible vapor cloud that could form and mix sufficiently under stable atmospheric conditions would be below the LFL boundary before reaching the Clinch River Nuclear Site. Furthermore, a vapor cloud explosion analysis was also completed to obtain safe distances (the minimum distance required for an explosion to have less than a 1 psi peak incident pressure). The results indicated the LFL distances and explosive safe distance were less than the shortest distance to between the Clinch River Nuclear Site and the storage location of these chemicals. As the site is also not within these aforementioned safety distances, a flammable vapor cloud from the storage of chemicals at these offsite facilities would not adversely affect the safe operation of the facility.

The locations and quantities of chemical that would be stored onsite at the Clinch River Nuclear Site were not evaluated in the ESPA, Part 2, SSAR (Reference 9). The ESPA, Part 2, SSAR noted that the effects of flammable vapor clouds and vapor cloud explosions from onsite chemical storage would be evaluated in the combined license application for a future reactor project (Reference 9). Chemicals stored at the future reactor site would be maintained and stored in a manner that would be protective of on-site personnel and the on-site reactor(s). Furthermore, due to the distance from the Clinch River Nuclear Site to the site, a flammable vapor cloud from the Clinch River Nuclear Site would not adversely affect the safe operation of the facility.

2.2.3.3 Toxic Chemicals

Events involving the release of chemicals in the vicinity of the site are considered for their potential toxicity and ability to affect personnel in the Main Control Room. The potential for an offsite toxic gas release is evaluated within 5 miles (8 km) of the site.

The evaluation considers stationary sources and mobile sources expected to be transported on nearby roads, on nearby waterways, or on local railroads. The effects of a chemical release from a pipeline are considered bounded by the delayed ignition explosion of a pipeline.

Chemicals are screened in several ways. Only chemicals with vapor pressures greater than 10 Torr at 100°F considered for further evaluation. Mobile sources are not considered if their shipment is not frequent (i.e., less than 10 shipments per year for truck traffic or 30 shipments per year for rail traffic).

In some cases, chemicals are screened as being bounded by other chemicals. A chemical determined to not present a toxic hazard to the site can be considered bounding to other chemicals that meet these four criteria: (1) have the same or lower vapor pressure; (2) have similar or lower toxicity; (3) are located the same or a farther distance away; and (4) are present in a similar or lower quantity. Additionally, in order to bound some chemicals, it is assumed, given identical meteorological conditions, initial chemical inventories, and travel distances, that:

- A chemical that exists as a gas or vapor will result in higher downwind concentrations than one that exists as a liquid.
- Volatile liquids, liquids with higher vapor pressures, or liquids with low boiling points near ambient temperatures will result in higher downwind concentrations than non-volatile liquids, liquids with lower vapor pressures, and liquids with high boiling points.
- A spill or leak of a solid chemical will not result in significant atmospheric concentrations capable of incapacitating an operator at the site, regardless of the chemical. This is because solids typically have very low vapor pressures, and solid particulates are heavier than vapor or gas molecules, and are therefore much less widely dispersed in air.

For these chemicals, airborne dispersion was evaluated deterministically for the nearby Clinch River Nuclear Site in the Clinch River Nuclear Site ESPA, Part 2, SSAR, using worst-case wind directions, and a temperature and wind speed with an annual exceedance probability of 5 percent. Only maximum concentration accidents were evaluated based on releases of the maximum expected amounts of chemicals. Maximum concentration-duration accidents were not evaluated because after shutting down the facility the operators do not need to take other actions to assure facility safety. These deterministic evaluations for the Clinch River Nuclear Site were performed using ALOHA.

The reactor control room considered in the Clinch River Nuclear Site evaluation was assumed to have an air-exchange rate of 1.2 exchanges per hour. The Main Control Room would have a similar air-exchange rate.

Pipelines

There are two bounding natural gas pipelines 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 Immediately Dangerous to Life or Health (IDLH) or Emergency Response Planning Guideline levels for methane. The distance to the asphyxiating limit for the East Tennessee Natural Gas Pipeline 1 and Pipeline 2 were evaluated in the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9). The Clinch River Nuclear Site ESPA, Part 2, SSAR determined an asphyxiating distance limit under the worst-case meteorological conditions of 282 feet and 846 feet, respectively (Reference 9). These distances are less than the separation distance from either pipeline. The closest branch of either pipeline is Pipeline 2, which is approximately 1 mile (1.6 km) north northeast of the site. Therefore, a break in either the East Tennessee Gas Pipeline 1 or East Tennessee Gas Pipeline 2 will not 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 not be a threat to the facility.

Waterway Traffic

As discussed in Section 2.2.3.2, there is an inconsequential amount of shipping on the Clinch River, and no transportation of hazardous materials. Additionally, there is no shipping on Poplar Creek. Therefore, chemicals transported by boat are not evaluated.

Highways and Railways

The site safety-related area is located approximately 4.8 miles (7.7 km) from I-40 and approximately 1.2 miles from TN-58. For this analysis, these distances were also used as the distance from I-40 and TN-58, respectively, to the Main Control Room.

The hazardous chemicals evaluated are primarily based on those chemicals identified in Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9). The selection of mobile sources for an analysis of potential impact to the Main Control Room is based on:

- Mobile sources of hazardous chemicals described in Table 2.2-4
- Stationary sources within 5 miles where deliveries or shipments could be transported on local roads
- Large quantities of stationary sources elsewhere in the county where deliveries or shipments could be transported on major roads or rail lines
- Direct communication with facilities regarding their types, quantities, and frequencies of shipments

An evaluation of hazardous materials potentially transported on I-40 was performed using the ALOHA dispersion model for the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9). The results indicated that, except for 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 the transportation route (I-40) are less than the minimum separation distances from the Clinch River Nuclear Site power block area to I-40 (approximately 1.1 miles). A release of anhydrous ammonia would result in a distance of 2.6 miles to the toxicity endpoint, and a release of chlorine results in a distance of 4.5 miles to the toxicity endpoint, which are both less than the 4.8 mile distance separating the site and I-40. The exceptions are the potential impacts from transportation of chemicals on TN-58. The shortest distance from the site to TN-58 is approximately 1.2 miles (6,336 feet), less than the minimum safe distance for a toxic vapor cloud of chlorine (23,760 feet) or anhydrous ammonia (13,728 feet). Therefore, an incident involving chlorine or anhydrous ammonia on TN 58 could have an adverse impact on the Main Control Room. Therefore, the Main Control Room is designed with chlorine and ammonia detectors in the ventilation system as discussed in Section 7.4.

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 an Operating License.

Nearby Facilities

Five facilities were evaluated in the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9) as facilities of concern with regard to storage of chemicals with the potential for formation of toxic vapor clouds within the vicinity of the site. They were ORNL (including ORNL-URS and ORNL-Battelle), located approximately 5 miles (8 km) east of the site; TVA Kingston Fossil Plant, located approximately 7 miles (11.2 km) southwest of the site; the TVA Bull Run Fossil Plant, located approximately 15 miles (24.1 km) east of the site; the Oak Ridge Water Treatment Plant (WTP) located approximately 9.5 miles (15 km) northeast of the site; and Hallsdale Powell Utility District Melton Hill Water Treatment Plant located approximately 18 miles (29 km) east of the site. Each material was then dispositioned based on the identified physical properties of the material and whether a bounding analysis existed. The material stored at ORNL-URS identified for further analysis was nitric acid. The materials stored at ORNL–Battelle identified for further analysis with regard to toxicity potential are: anhydrous ammonia, argon, carbon dioxide, chloroform, chromic chloride, ethanol, gasoline (gasoline blend A and gasoline B), hydrogen fluoride, nitrogen, and sulfur hexafluoride. The material stored at TVA Kingston Fossil Plant and TVA Bull

Run Fossil Plant identified for further analysis was anhydrous ammonia. The material stored at the Oak Ridge WTP and the Hallsdale Powell Utility District Melton Hill WTP identified for further analysis was chlorine.

In the Clinch River Nuclear Site ESPA, Part 2, SSAR, the above-identified chemicals were analyzed using the ALOHA dispersion model to determine whether the formed vapor cloud would reach the Clinch River Nuclear Site power block area with concentrations greater than the determined toxicity limit (Reference 9). In the case of each of the atmospheric gases analyzed, the distances to the IDLH/asphyxiating or other determined toxicity limit was calculated. The results indicated that any plausible toxic vapor cloud that could form would be below the IDLH or other identified toxicity limit before reaching the Clinch River Nuclear Site power block area. This conclusion would be the same for the site.

The modeling of the aforementioned facilities indicated the accidental release of the analyzed hazardous materials stored on site would not adversely affect the safe operation or shutdown of units within the Clinch River Nuclear Site power block area as indicated in the Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9). Of the chemicals identified for analysis at ORNL–Battelle, a release of sulfur hexafluoride from ORNL–Battelle resulted in the longest distance to the toxicity endpoint, 2 miles, which is less than both the distance to the Clinch River Nuclear Site power block area and the site (located approximately 5 miles away). Therefore, the formation of a toxic vapor cloud following an accidental release of the analyzed hazardous materials stored at ORNL-Battelle would not adversely affect the safe operation or shutdown of the facility.

The locations and quantities of chemical that would be stored onsite at the Clinch River Nuclear Site were not evaluated in the ESPA, Part 2, SSAR (Reference 9). The ESPA, Part 2, SSAR noted that the effects of toxic chemical releases from onsite chemical storage would be evaluated in the combined license application for a future reactor project (Reference 9). Chemicals stored at the future reactor site would be maintained and stored in a manner that would be protective of on-site personnel and the on-site reactor(s). Furthermore, due to the distance from the Clinch River Nuclear Site to the site, a toxic vapor cloud from the Clinch River Nuclear Site would not adversely affect the safe operation of the facility.

2.2.3.4 Fires

As demonstrated in the previous sections, analysis conducted in the Clinch River Nuclear Site ESPA, Part 2, SSAR considered potential external accidents that could lead to high heat fluxes. The analyses showed that chemicals stored at nearby facilities and transported on I-40 and TN-58 would not result in a vapor cloud with a potential to affect the Clinch River Nuclear Site. The previous sections also demonstrate that the explosive and flammable vapor cloud analyses provided in the Clinch River Nuclear Site ESPA, Part 2, SSAR are acceptable for application to the site.

The effects of fires from brush or forest fires will be evaluated in the application for an Operating License.

2.2.4 References

1. Mapbox, East Tennessee Natural Gas – Enbridge. 2021.
2. Enbridge, Enbridge Asset Map. 2021. Retrieved from https://www.enbridge.com/map#map:infrastructure,gaspipelinetrans,gatheringInfrastructure,NGLPipelines,fractionation.

3. Global Energy Monitor, East Tennessee Pipeline. 2021. Retrieved from https://www.gem.wiki/East_Tennessee_Gas_Pipeline.
4. U.S. Army Corps of Engineers, Manuscript cargo and trips data files, statistics on foreign and domestic waterborne commerce move on the United States waters [2000-2019 cargo]. 2019. Retrieved from https://usace.contentdm.oclc.org/digital/collection/p16021coll2/id/1795.
5. U.S. Army Corps of Engineers, Manuscript cargo and trips data files, statistics on foreign and domestic waterborne commerce move on the United States waters [2000-2016 cargo]. 2016. Retrieved from https://usace.contentdm.oclc.org/digital/collection/p16021coll2/id/6819.
6. Tennessee Department of Transportation, Annual Average Daily Traffic, Roane County, East of Kingston, Route 1-40. 2018. Retrieved from https://www.arcgis.com/apps/webappviewer/index.html?id=075987cdae37474b88fa400d65681354.
7. Tennessee Department of Transportation, Annual Average Daily Traffic, Roane County, North of SR 95, Route TN-327. 2018. Retrieved from https://www.arcgis.com/apps/webappviewer/index.html?id=075987cdae37474b88fa400d65681354.
8. Tennessee Department of Transportation, Annual Average Daily Traffic, Roane County, Oak Ridge, Route TN-58, 2018. Retrieved from https://www.arcgis.com/apps/webappviewer/index.html?id=075987cdae37474b88fa400d65681354.
9. Tennessee Valley Authority (TVA), Clinch River Nuclear Site Early Site Permit Application, Part 2, Site Safety Analysis Report, Revision 2, Section 2.2. March 8, 2019.
10. Pounds, B., Oak Ridge Airport, Oak Ridger. March 16, 2021. Retrieved from https://www.aviationpros.com/airports/news/21214654/more-steps-taken-toward-creation-of-new-oak-ridge-airport.
11. U.S. Department of Energy, Office of Energy, Oak Ridge Office of Environmental Management, Environmental Assessment, Property Transfer to Develop a General Aviation Airport at the East Tennessee Technology Park Heritage Center, Oak Ridge, Tennessee. DOE/EA-2000, FINAL. Oak Ridge, Tennessee. February 2016.
12. UCOR, Heritage Center Revitalization Plan. May 2017. Retrieved from http://heritagectr.com/wp-content/uploads/2018/02/Heritage-Center-Revitalization-Plan-Final-5.26.17-1.pdf.
13. Federal Aviation Administration (FAA), Instrument Procedures Handbook, Chapter 2 En Route Operations. https://www.faa.gov/regulations_policies/handbooks_manuals/aviation/instrument_procedures_handbook/media/FAA-H-8083-16B_Chapter_2.pdf.
14. U.S. Department of Energy, Accident Analysis for Aircraft Crash Into Hazardous Facilities. DOE Standard DOE-STD-3014-2006, Reaffirmation, Washington DC. May 2006.
15. Fire dynamics Tools (FDT): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program, Final Report. December 2004.
16. Federal Emergency Management Agency. Handbook of Chemical Hazard Analysis Procedures. OSWERHCHAP. 1989. Accessed 7/18/2021. Located at:

https://www.ci.benicia.ca.us/vertical/sites/%7BF991A639-AAED-4E1A-9735-86EA195E2C8D%7D/uploads/FEMA_1989.pdf.

17. U.S. Nuclear Regulatory Commission, Fire Risk Analysis For Nuclear Power Plants. Draft Chapter 14. 2016. Accessed 7/18/2021. Located at: https://www.nrc.gov/docs/ML1408/ML14084A314.pdf.
18. SHINE Medical Technologies, LLC, SHINE Medical Isotope Production Facility Preliminary Safety Analysis Report. August 2015.
19. Energy*Solutions*, Bear Creek Processing Facility. 2016. Retrieved from https://www.energysolutions.com/waste-processing/bear-creek-processing-facility/.
20. Oak Ridge National Laboratory, User Facilities, Website: https://www.ornl.gov/content/user-facilities. Accessed June 27, 2021.
21. Munger, F., "What to do With White Oak Lake," Cleanup, EPA, Oak Ridge EM, ORNL, TDEC. July 09, 2015. Retrieved from http://knoxblogs.com/atomiccity/2015/07/09/what-to-do-with-white-oak-lake/.
22. Van Winkle, J.E., "Baseline Environmental Analysis Report for the K-1251 Barge Facility at the East Tennessee Technology Park," Oak Ridge, Tennessee. 2007. Retrieved from https://www.osti.gov/biblio/964674-EpJWXe/.
23. Krause, C., "Medical Isotope Firm CEO tells of plans for OR facility," The Oak Ridger. March 19, 2020. Retrieved from https://www.oakridger.com/story/news/technology/2020/03/19/medical-isotope-firm-ceo--tells-of-plans-for-or-facility/112376002/.
24. Roane Regional Business & Technology Park, Roane ECD, Website: https://www.roaneecd.com/sites/roane-regional-business-tech-park/. Accessed June 27, 2021.
25. City of Oak Ridge, Tennessee, Horizon Center Industrial Park, Website: http://oridb.net/horizon-center-park/. Accessed 06/27/2021.
26. Not Used.
27. Pounds, B., "UT-Battelle Gives City $500K for Airport," Oak Ridger, Aviation Pros. March 24, 2021. Retrieved from https://www.aviationpros.com/airports/news/21215666/utbattelle-gives-city-500k-for-airport.
28. Oak Ridge Municipal Planning Commission, Virtual Regular Business Meeting. March 18, 2021. Retrieved from http://oakridgetn.gov/images/uploads/Documents/Boards&Commissions/Planning%20Commission/Agenda/PC_Packet_03_18_2021.pdf.
29. National Institute for Occupational Safety and Health (NIOSH), "NIOSH Pocket Guide to Chemical Hazards." September 2007.
30. U.S. Environmental Protection Agency and National Oceanic and Atmospheric Administration's Office of Response and Restoration, Computer-Aided Management of Emergency Operations (CAMEO), Website: https://www.epa.gov/cameo.
31. The Chlorine Institute, Pamphlet 5, Bulk Storage of Liquid Chlorine, 7th ed. October 2005.
32. Weldship Corporation, Super Jumbo Tube Trailers, Product Specifications, Website: http://www.weldship.com/superjumbo.php. Accessed July 18, 2021.

33. Kairos Power LLC, "Preliminary Safety Analysis Report for the Kairos Power Fluoride Salt-Cooled, High Temperature Non-Power Reactor (Hermes)," Revision 3, May 2023.

Table 2.2-1: Nearby Facilities (Page 1 of 2)

Project Name	Summary of Project	Location (from Reactor Building)	Status	Notes
Federal Facilities				
Proposed Clinch River Nuclear Site (Reference 9)	Two or more small modular reactors to be built by TVA	3.5 miles South-southeast	Proposed NRC issued ESP-006 on December 19, 2019	Potential for overlapping construction timeline
Energy*Solutions*, LLC, Bear Creek Facility (Reference 19)	Processing and packaging of radioactive material for permanent disposal	Approximately 2.1 miles southeast	Operational	
ORNL (Reference 20)	DOE Nuclear and High-Tech Research Facility	Approximately 5 miles east	Operational since 1943	
White Oak Dam (Reference 21)	Manhattan Project impoundment on White Oak Creek with 25 ac settling pond. Formed to reduce radioactive waste runoff into Clinch River	Approximately 5 miles southeast	Operational since 1943	
Y-12 Shipping and Receiving (Onsite verification)	Non-hazardous shipping and receiving facility	West and adjacent <1000 ft	Operational	
K-1251 Barge Facility (Reference 22)	Barge docking facility approximately 1-ac in size	2 miles Southeast	Operational	
TVA Kingston Fossil Plant	Coal-fired electrical generating facility	7 miles southwest	Operational	
TVA Bull Run Fossil Plant	Coal-fired electrical generating facility	15 miles east	Operational	
Industrial Facilities				
Coquí Pharma (Reference 23)	Planned Medical Isotope Production Facility	Duct Island; Approximately 0.75 miles south	Proposed	Close proximity and potential for overlapping construction timeline
Roane Regional Business and Technology Park (Reference 24)	Business and industrial park with sites for development	Approximately 5 miles southeast	Operational since 2001	Operational, contains the following businesses: • Advanced Plasma Products • C.R. Barger & Sons • Dynamic Tooling Systems • H.T. Hackney • Proton Power
Horizon Center Industrial Park (Reference 25)	Industrial park with sites for development; current residents include the Carbon Fiber Technology Facility and the ORNL pilot demonstration facility for reducing the cost of carbon fiber	Approximately 2.3 miles northeast	Operational/ under development	Accidents due to the current materials and operations at this facility are not expected to affect the site.

Table 2.2-1: Nearby Facilities (Page 2 of 2)

Project Name	Summary of Project	Location (from Reactor Building)	Status	Notes
Industrial Facilities Continued				
Hallsdale Water Treatment Plant	Municipal waste water treatment facility	Approximately 18 mi east	Operational	Due to its distance to the site, accidents due to the current materials and operations at this facility is not expected to affect the site.
City of Oak Ridge Water Treatment Plant	Municipal waste water treatment facility	Approximately 9.5 mi northeast	Operational	Due to its distance to the site, accidents due to the current materials and operations at this facility is not expected to affect the site.
Kairos Power Hermes Facility	Low-power nuclear demonstration reactor facility	Within site boundary	Proposed	The presence of a diesel fuel storage tank has been analyzed; other chemical hazards will be analyzed with the operating license application
Kairos Power Atlas Fuel Fabrication Facility	TRISO based nuclear fuel fabrication facility	Within site boundary	Proposed	
Ultra Safe Nuclear Corporation Pilot Fuel Manufacturing Facility	TRISO based pilot fuel fabrication facility	Approximately 0.8 miles southeast	Operational	Accidents due to the current materials and operations at this facility are not expected to affect the site
TRISO-X Fuel Facility	TRISO based nuclear fuel fabrication facility	Approximately 2.4 miles northeast	Proposed	
Transportation				
Proposed General Aviation Airport at the East Tennessee Technology Park Heritage Center (Reference 10, Reference 27)	Development of a general aviation airport	Approximately 1.1 miles east	Proposed	Close proximity and potential for overlapping construction timeline
Residential Facilities				
The Preserve at Clinch River water treatment facility (Reference 28)	Residential water and wastewater treatment facility	Approximately 2 miles south	Operational since 2002	

Table 2.2-2: Facilities Unable to Affect the Facility

Project Name	Summary of Project	Location (from Reactor Building)	Status	Notes
Federal Facilities				
Energy*Solutions*, LLC, Bear Creek Facility (Reference 19)	Processing and packaging of radioactive material for permanent disposal	Approximately 2.1 miles southeast	Operational	Accidents due to the materials and operations at this facility would not affect the site.
White Oak Dam (Reference 21)	Manhattan Project impoundment on White Oak Creek with 25 ac settling pond. Formed to reduce radioactive waste runoff into Clinch River.	Approximately 5 miles southeast	Operational	No materials at this facility would affect the Facility; failure of the dam would not affect the site.
Y-12 Shipping and Receiving (Onsite verification)	Non-hazardous shipping and receiving facility	West and adjacent <1,000 feet	Operational	No materials at this facility would affect the site.
K-1251 Barge Facility (Reference 22)	Barge docking facility approximately 1-ac in size	2 miles Southeast	Operational	No materials at this facility. The only potential for explosion or fire would be when it is in use supporting construction at the Clinch River Nuclear Site.
Industrial Facilities				
Ultra Safe Nuclear Corporation Pilot Fuel Manufacturing Facility	TRISO based pilot fuel fabrication facility	Approximately 0.8 miles southeast	Operational	Accidents due to the current materials and operations at this facility are not expected to affect the site
Roane Regional Business and Technology Park (Reference 24)	Business and industrial park with sites for development	Approximately 5 miles southeast	Operational	Accidents due to the materials and operations at this facility should not affect the site due to its distance from the facility.
Horizon Center Industrial Park (Reference 25)	Industrial park with sites for development; current residents include the Carbon Fiber Technology Facility and the ORNL pilot demonstration facility for reducing the cost of carbon fiber	Approximately 2.3 miles northeast	Operational/ under development	Accidents due to the current materials and operations at this facility are not expected to affect the site.
Hallsdale Water Treatment Plant	Municipal waste water treatment facility	Approximately 18 mi east	Operational	Due to its distance to the site, accidents due to the current materials and operations at this facility is not expected to affect the site.
City of Oak Ridge Water Treatment Plant	Municipal waste water treatment facility	Approximately 10 mi northeast	Operational	Due to its distance to the site, accidents due to the current materials and operations at this facility is not expected to affect the site.
Residential Facilities				
The Preserve at Clinch River water treatment facility (Reference 28)	Residential water and wastewater treatment facility	Approximately 2 miles south	Operational	Accidents due to the materials and operations at this facility should not affect the site.

Table 2.2-3: Nearby Facility Chemical Storage

Chemical	Facility/Location	Capacity[a] (pounds)	Toxicity limit IDLH[b]
Anhydrous Ammonia	ORNL-Battelle	999	300 ppm
Argon	ORNL-Battelle	9,999	Not Available
Carbon Dioxide	ORNL-Battelle	4,999	40,000 ppm
Chloroform	ORNL-Battelle	99	500 ppm
Chromic Chloride	ORNL-Battelle	99	25 mg/m^3
Diesel Fuel Oil #2	ORNL-Battelle	24,999	Not Available
Diesel Fuel Oil #2	Kairos Power Hermes Facility	152,958	Not Available
Ethanol/Gasoline Blend (85-15)	ORNL-Battelle	4,999	3,300 (as ethanol)
Ferri-Floc (Feric Sulfate)	ORNL-URS	24,999	Not Available
Fertilizer (18-24-12)	ORNL-URS	24,999	Not Available
Flibe (Li_2BeF_4) Molten salt coolant	Kairos Power Hermes Facility	40,000	Not Available
Gasoline (unleaded)	ORNL-Battelle	999	300 ppm TWA(b) 750 ppm (as n-Heptane)[c]
Hydrogen Fluoride	ORNL-Battelle	499	30 ppm
Lead	ORNL-URS	499,999	100 mg/m^3
Lead	ORNL-Battelle	9,999	100 mg/m^3
Limestone (AgriPel Pelletized Calcite)	ORNL-URS	49,999	Not Available
Lithium Hydride	ORNL-URS	24,999	0.5 mg/m^3
Lithium Hydride	ORNL-Battelle	4,999	0.5 mg/m^3
Mercury	ORNL-Battelle	99	10 mg/m^3
Nitric Acid	ORNL-URS	499,999	25 ppm
Nitric Acid	ORNL-Battelle	999	25 ppm
Nitrogen	ORNL-Battelle	9,999	Asphyxiant
Oils	ORNL-Battelle	4,999	2,500 mg/m^3
Salt (Sodium Chloride)	ORNL-Battelle	4,999	Not Available
Sodium Bisulfate Solution	ORNL-Battelle	9,999	Not Available
Sulfuric Acid	ORNL-Battelle	9,999	15 mg/m^3
Sulfur Hexafluoride	ORNL-Battelle	499,999	1000 ppm as TWA
Sodium Hydroxide Solution	ORNL-URS	499,999	10 mg/m^3
Sodium Metal	ORNL-URS	49,999	Not Available
Sulfuric Acid	ORNL-URS	24,999	15 mg/m^3
Sulfuric Acid	ORNL-Battelle	9,999	15 mg/m^3
Chlorine	Rarity Ridge WWTP	10,000	10 ppm

[a] Where a capacity number was obtained from the Superfund Amendments and Reauthorization Act (SARA) Title III, Tier II report, the upper range number is shown and was used in the analysis.
[b] Immediately Dangerous to Life or Health. "Not Available" indicates that there has not been a toxicity limit established for this chemical.
[c] Gasoline does not have an identified IDLH. The Threshold Limit Value–Short Term Exposure Limit TLV–STEL is 500 ppm; the Threshold Limit Value–Time-weighted Average (TLV–TWA) is 300 ppm; and the Protective Action Criteria (PAC) PAC-2 guideline is 1,000 ppm for gasoline. For the analyses, n-Heptane is used as a surrogate and has an IDLH of 750 ppm. This selection is conservative given the PAC-2 guideline most closely correlates with the definition of IDLH.

Notes:
ppm = parts per million; mg/m^3 = milligram per cubic meter; IDLH = Immediately Dangerous to Life or Health;
WWTP = Wastewater Treatment Plant; TWA = Time-weighted Average
Sources: Reference 29, Reference 30, Reference 31

Table 2.2-4: Hazardous Materials Potentially Transported Along I-40 and TN-58 in the Vicinity of the Facility

Chemical	Quantity	Toxicity Limit IDLH[a]
Anhydrous Ammonia	11,500 gal[b]	300 ppm
Argon	50,000 lb[c]	Asphyxiant
Butane	11,500 gal[b]	Asphyxiant
Carbon Dioxide	50,000 lb[c]	40,000 ppm
Chlorine	44,000 lb[d]	10 ppm
Chloroform	50,000 lb[c]	500 ppm
Chromic Chloride	50,000 lb[c]	25 mg/m^3
Ethanol	50,000 lb[c]	3,300 ppm
Gasoline	8,500 gal[e]	300 ppm TWA[h] 750 ppm (as n-heptane)[h]
Hydrogen Gas	15,032 ft^3/tube[f]	Not Available[i]
Hydrogen Fluoride	50,000 lb[c]	30 ppm
Nitric Acid	6,000 gal[g]	25 ppm
Nitrogen	50,000 lb[c]	Asphyxiant
Sodium Hypochlorite	50,000 lb[c]	10 ppm as Chlorine
Sulfur Hexafluoride	50,000 lb[c]	1,000 ppm[j]

[a] IDLH. “Not Available” indicates that there has not been a toxicity limit established for this chemical.
[b] The maximum capacity of MC-331 high pressure tank truck is 11,500 gal per 49 CFR 173.315.
[c] Per Regulatory Guide 1.91, the maximum probable cargo for a single highway truck is 50,000 lb and used for the quantity transported unless a more appropriate value could be determined.
[d] Chlorine gas quantity determined from The Chlorine Institute Bulk Storage of Liquid Chlorine (Reference 31).
[e] The maximum highway cargo capacity, 50,000 lb provided in Regulatory Guide 1.91 was converted to gal for gasoline.
[f] Hydrogen gas quantity determined from Weldship Corporation super jumbo tube product specifications (the largest size tube available) (Reference 32).
[g] The maximum capacity of MC-312/DOT412 corrosive tanker is 6,000 gal.
[h] Gasoline does not have an identified IDLH. The Threshold Limit Value–Short Term Exposure Limit (TLV–STEL) is 500 ppm; the Threshold Limit Value–Time-weighted Average (TLV–TWA) is 300 ppm; and the Protective Action Criteria (PAC) PAC-2 guideline is 1,000 ppm for gasoline. For the analyses, n-Heptane is used as a surrogate and has an IDLH of 750 ppm. This selection is conservative given the PAC-2 guideline most closely correlates with the definition of IDLH.
[i] This analysis is bounding for ALOHA vapor cloud dispersion modeling of gaseous hydrogen due to the extreme buoyancy of hydrogen. That is, hydrogen gas would rise extremely rapidly and not cause a travelling vapor cloud.
[j] No IDLH is established for sulfur hexafluoride; therefore, the TWA is used as a toxic limit.
Notes: parts per million (ppm); milligram per cubic meter (mg/m3); Immediately Dangerous to Life or Health (IDLH); Time-weighted Average (TWA)
Sources: Reference 29, Reference 30, Reference 31, Reference 32.

Table 2.2-5: Federal Airways within Ten miles (16 km) of the Site

Airway	Distance from Airway Centerline to Site (miles)[a]	Airway Width (miles)[a][b]	Distance from Airway Edge to Site center (miles)[a][c]
J46	0.88	9.2	(b)
V16	6.24	9.2	1.64

[a] Statute miles
[b] Site is within the airway width
[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.

Table 2.2-6: DOE Input Values

NjPjfj(x,y) Values	
	NjPjfj(x,y) Value[a] (1/mi²)
Air Carrier	6.00E-07
Air Taxi	2.00E-06
General Aviation	2.00E-03
Small Military	6.00E-07
Large Military	1.00E-07

Effective Area Input Values			
	WS(ft)[b]	**cot(ϕ)[c]**	**S (ft)[d]**
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

Reactor Building Safety Related Area Dimensions[(e)]		
	feet	**miles**
Length (L)	170	3.22E-02
Width (W)	50	9.47-03
Height (H)	42	7.95E-03

[(a)] Source: Tables B-14, B-15 for Oak Ridge National Laboratory from Reference 14.
[(b)] Source: Table B-16 from Reference 14.
[(c)] Source: Table B-17 from Reference 14.
[(d)] Source: Table B-18, assume takeoff skid length for in-flight crashes from Reference 14.
[(e)] Final area used in calculations for Table 2.2-7 through Table 2.2-10 is increased to account for both Reactor Buildings.

Table 2.2-7: Calculated Effective Areas of Safety-Related Structures (square miles) by Aircraft Type Used for the Evaluation of Airways and Airport

Aircraft Type	Effective Area (A_j) (sq mi)
Air Carrier	3.06E-02
Air Taxi	2.78E-02
General Aviation	6.91E-03
Small Military	1.07E-02
Large Military	2.36E-02
Helicopter	1.13E-03

Table 2.2-8: Near-Airport and Helicopter Crash Frequency Inputs and Calculations

	N, Number of Operations Per Year[a]	x distance mi[b]	y distance mi[b]	f(x,y) value[c]	P, Crash Rate[d]	A, mi^2	Impact Frequency[e]
General Aviation Takeoff	2.41E+04	+0.7	-1.2	1.30E-02	1.10E-05	6.91E-03	2.38E-05
General Aviation Landing	2.41E+04	-0.7	+1.2	1.20E-02	2.00E-05	6.91E-03	4.00E-05

	N, Number of Operations Per Year[a]	P, Crash Rate[d]	A, mi^2	L_H[f]	Helicopter Impact Frequency[g]
Helicopter	1,491	2.50E-5	1.13E-03	37	2.28E-06

[a] Obtained from Table 2.5 in Oak Ridge EA, (Reference 11). Annual helicopter operations (1,491) were subtracted from total annual aircraft operations (49,713) total operations and the remainder was assumed 50% takeoff and 50% landing operations.
[b] Orthonormal distance from the site to the center of each runway at the flight source. Distance values were estimated based on the best current available information. Takeoff is assumed to be to the southwest and landing is assumed to be to the northeast.
[c] Reference 14, Tables B2-B5. Flight direction is currently unknown; therefore, the largest value of f(±x,±y) was selected for conservatism.
[d] Reference 14, Tables B-1. Assumed representative fixed wing for General Aviation operations.
[e] Calculated from Equation 5-1 (Reference 14).
[f] Reference 14, Table B-43.
[g] Calculated from Equation 5-3 (Reference 14).

Table 2.2-9: Total Crash Probability

Aircraft Type	Airway Operations	Near-Airport Operations			Total
		Airplane Takeoff	Airplane Landing	Helicopter Operations	
Air Carrier	1.83E-08	-	-	-	1.83E-08
Air Taxi	5.56E-08	-	-	-	5.56E-08
General Aviation	1.38E-05	2.38E-05	4.00E-05	2.28E-06	7.99E-05
Small Military	6.45E-09	-	-	-	6.45E-09
Large Military	2.36E-09	-	-	-	2.36E-09
Total	1.39E-05	2.38E-05	4.00E-05	2.28E-06	**8.00E-05**

Table 2.2-10: Evaluation of Chemical Explosion Hazards Near the Site

Source	Chemical Evaluated[a]	Quantity Analyzed[a]	Heat of Combustion (Btu/lb)[a]	Approximate Distance to the Site (ft)	Safe Distance for Explosion to have less than 1 psi of Peak Incident Pressure (ft)[a]
Nearby Offsite Facilities					
ORNL-Battelle	Anhydrous Ammonia	999 lb	7,992	26,400	47.8
	Ethanol (85%)	4,249 lb	11,570		103.3
	Gasoline Blend A (as n-Heptane)	750lb	18,720		63.4
	Gasoline B (as n-Heptane)	999 lb	18,720		75.4
Kairos Power Hermes Facility (Proposed)	Diesel-fuel	21,555 gal	17,000	528	500
Nearby Transport Routes/Roadways					
I-40	Butane	11,500 gal	19,152	25,872	3,708
	Gasoline	8,500 gal	18,720		273
	Hydrogen	15,032 ft^3/tube[c]	50,080		520[d]
TN-58[b]	Butane	11,500 gal	19,152	6,336	3,708
	Gasoline	8,500 gal	18,720		273
	Hydrogen	15,032 ft^3/tube[c]	50,080		520[d]

[a] From the Clinch River Nuclear Site ESPA SSAR (Reference 9)

[b] Assumes that any chemicals and quantities transported on I-40 would be the same chemicals and quantities that could be transported on TN-58 because TN-58 feeds into I-40.

[c] Transport quantity for a super jumbo tube (Reference 32).

[d] Minimum safe distance per super jumbo tube determined from Clinch River Nuclear Site ESPA, Part 2, SSAR (Reference 9). An independent evaluation was performed per Regulatory Guide 1.91 using conservative assumptions from a single explosion involving nine super jumbo tubes (i.e., typical trailer capacity). For nine tanks, the minimum standoff distance would correspond to 1,200 ft, well below the distance to the Site.

Figure 2.2-1: Nearby Industrial and Military Facilities

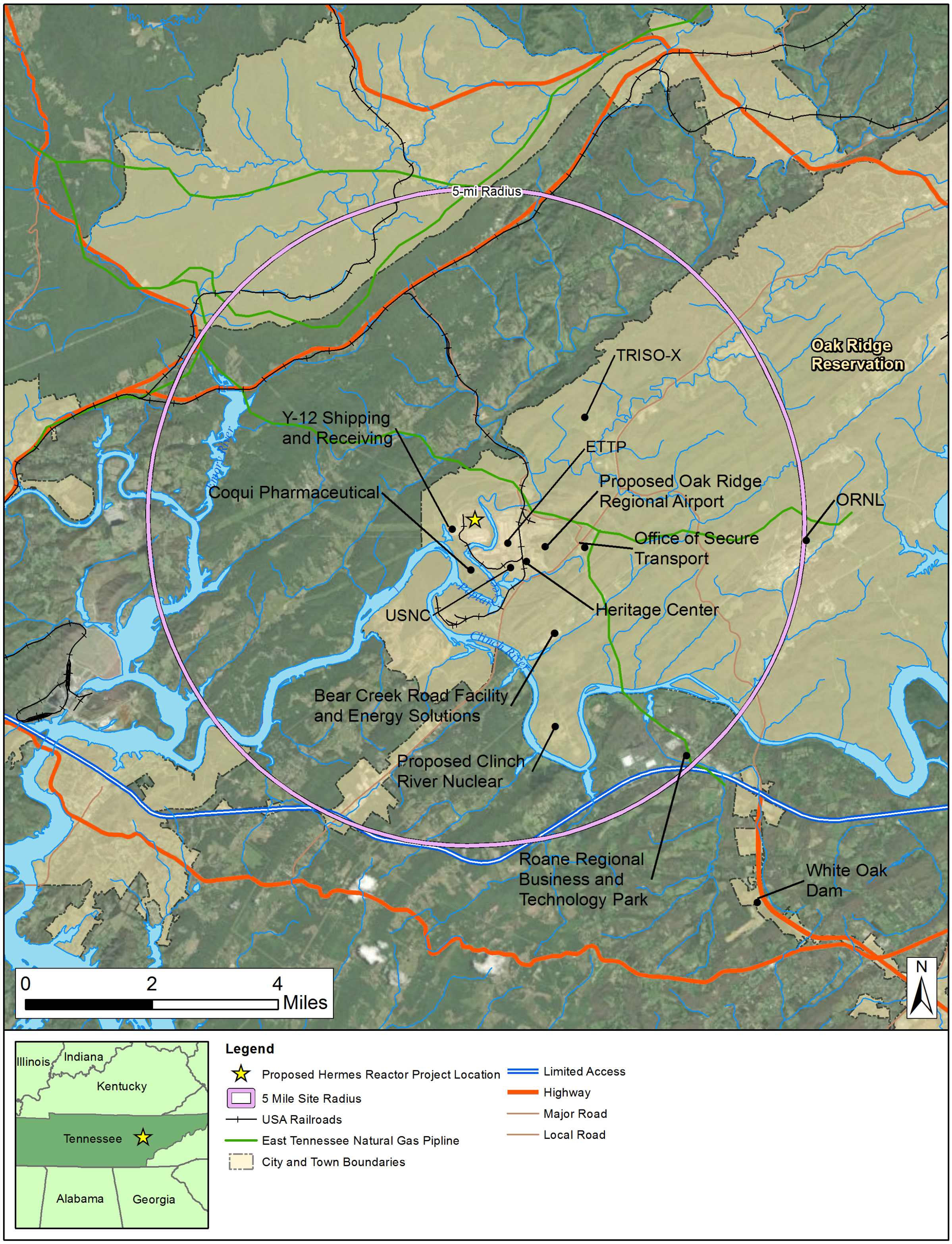
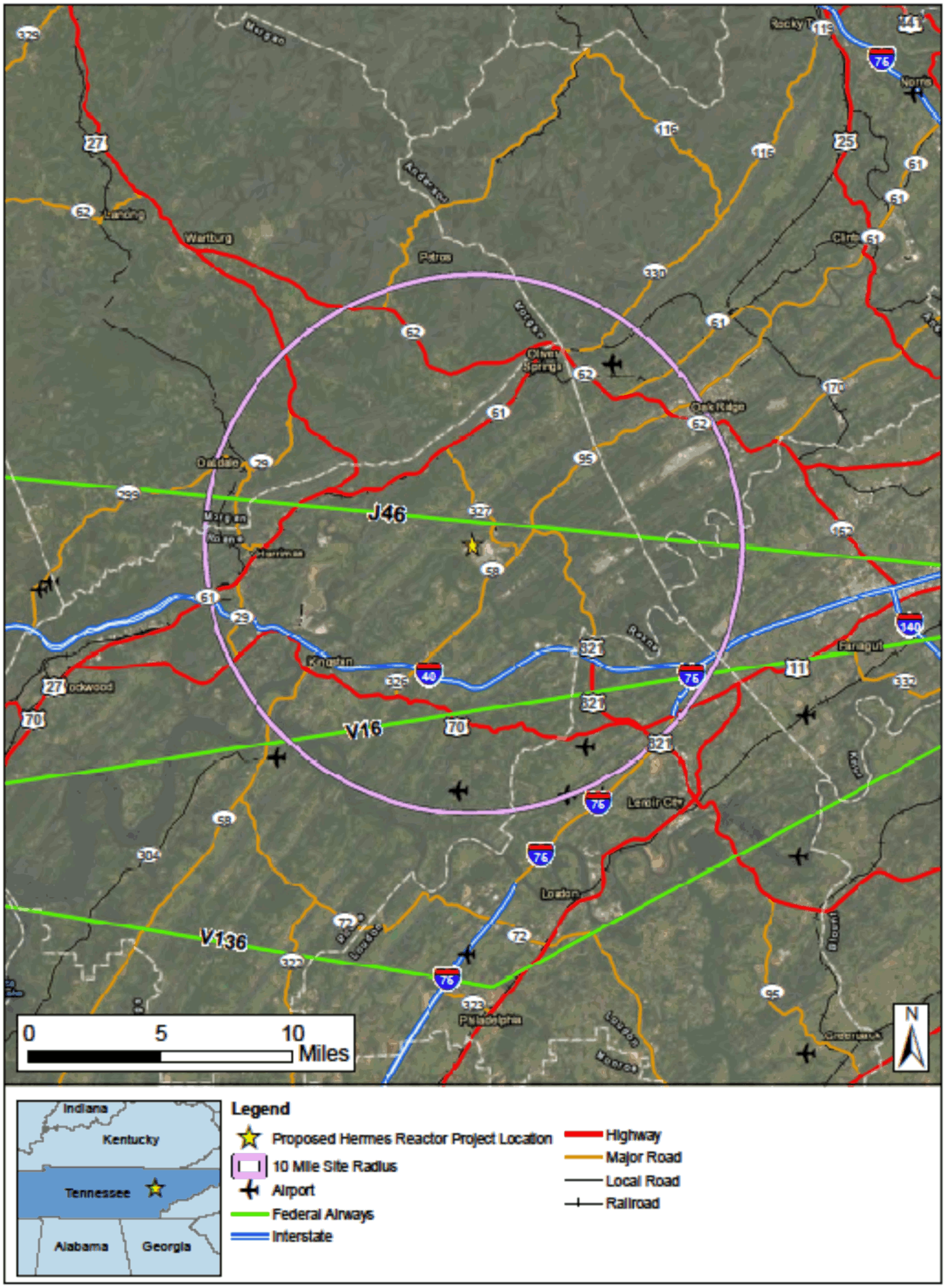


Figure 2.2-2: Airports, Jet Routes, and Airway Routes Within 10 miles (16 km) of the Site



2.3 METEOROLOGY

This section of the PSAR addresses the local and regional climatology and meteorology, as well as air quality in the vicinity of the site. The information is consistent with independent evaluations and assessments of atmospheric diffusion characteristics by Oak Ridge Reservation (ORR) scientists. A similar analysis was recently completed for the proposed Clinch River Nuclear (CRN) Site. The information provided in Section 2.3 of the CRN Early Site Permit Application, Part 2, Site Safety Analysis Report (Reference 1), has been used in part for this analysis because that project location is less than 3.5 miles south-southeast of the site in a similar terrain setting.

2.3.1 Regional Climatology

The site is located in Roane County in the eastern portion of Tennessee. The location is part of a region commonly referred to as “The Great Valley,” and is shown in Figure 2.3-1. The site is located approximately 8.7 miles southwest of the City of Oak Ridge, Tennessee, business district.

This region of Tennessee is dominated much of the year by the Azores-Bermuda anti-cyclonic circulation. This dominance is most pronounced in late summer and early fall and is accompanied by extended periods of fair weather and widespread atmospheric stagnation. In winter and early spring, eastward moving migratory high- or low-pressure systems bring alternately cold and warm air masses into the area. The resultant changes in wind, atmospheric stability, precipitation, and other meteorological elements cause the normal circulation to become more diffuse over the region. In the summer and early fall, the migratory systems are less frequent and less intense. Frequent incursions of warm, moist air from the Gulf of Mexico and occasionally from the Atlantic Ocean are experienced in the summer. The site is primarily affected by cyclones from the southwest and Gulf Coast that move toward the northeast United States by passing along either the west side or the east side of the Appalachian chain and by cyclones from the plains or Midwest that move up the Ohio River Valley.

Topography influences the weather and climate of the region around the site. The site is situated between two major mountain regions. To the northwest lie the Cumberland Mountains and to the southeast are the Great Smoky Mountains. These mountainous regions orient “The Great Valley” in a southwest-to-northeast alignment as shown in Figure 2.3-1. Prevailing winds in the region reflect the channeling of air flow caused by the orientation of the valleys and ridges. Average wind speeds are low, with a mean annual wind speed of 2.7 miles per hour (mph) at Oak Ridge (Reference 2). During winter when the jet stream moves southward, the Cumberland Mountains also serve to retard or moderate cold outbreaks by blocking dense, cold polar continental air masses. The Cumberland Mountains also reduce the intensity of thunderstorms in the summer that are produced by synoptic-scale systems crossing the region due to the downward momentum of the air mass as it comes off the higher terrain and moves into the Great Valley. Thunderstorms are more frequently caused by the heating of the land during the day. The orographic lift produced by the local topography may enhance these “air mass” thunderstorms (Reference 3).

Area temperatures measured in Oak Ridge indicate warm summers and mild winters. In January, the normal daily maximum temperature is about 48°F with a normal daily minimum temperature of about 30°F based on 30 years of data. In July, the normal daily maximum temperature is about 89°F, while the normal daily minimum temperature is about 69°F based on 30 years of data (1991-2020) from the National Climatic Data Center (NCDC) (Reference 4). Relative humidity in the region averaged 71 percent based on a 30-year period of record from the Knoxville Local Climatological Data (1991-2020) from the NCDC (Reference 5). The site is located in Tennessee Climate Division 1, also known as the East Tennessee Climate Division.

Precipitation averages about 56 inches annually (Reference 4). Late winter (January-March) is usually the wettest season, with more than about 16 inches, while late summer-early autumn (August-October) is the driest season, with more than 10 inches. Droughts are uncommon in this region of the United States.

Snowfall in the Oak Ridge area, though normally light, usually occurs from November through March. Severe storms are relatively infrequent as the region is east of maximum tornado activity, south of the most significant snowstorms, and inland from hurricane and tropical storm tracks (Reference 6).

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

2.3.1.1 Severe Weather

Severe weather phenomena may require consideration in the design of safety-related structures, systems, and components. Statistics on severe weather phenomena are obtained from historical data. Most data are taken from the NCDC Storm Events Database (Reference 7) that covers the 71-year period of 1950–2020, but even longer data periods are used for some phenomena to better capture the occurrence of rare events, such as maximum historical snowpack (see Section 2.3.1.11).

2.3.1.2 Thunderstorms

Thunderstorms are common in the Oak Ridge region with a normal range of 34–65 days with thunderstorms based on data collected from 2001–2021 at the Oak Ridge National Laboratory (ORNL) (Reference 8). The greatest frequency of thunderstorms is during the summer with a range of 18-40 days during May–August. This is characteristic of a diurnal afternoon thunderstorm pattern due to solar heating.

2.3.1.3 Hail

In Roane County, severe hail (3/4 inch in diameter or larger) has been reported only 36 times during 1950–2022 (Reference 7). This corresponds to less than one severe hail event per year. During the same period, surrounding counties reported severe hail between 50 (Loudon) and 94 (Knox) times.

2.3.1.4 Lightning

The site averages four to eight cloud-to-ground lightning flashes per square kilometer annually based on a 26-year period from 1993-2018 (Reference 9).

A review of cloud-to-ground lightning strike data from a 10-year period from 2011–2019 at the site indicates that 7 of the 10 years had a lightning strike occurring within 500 hundred feet of the site or beside the site (Reference 46). One of these years, 2012, was a year with an exceptionally high number of cloud-to-ground lightning strikes. Eleven lightning strikes occurred within the site boundary with several more strikes occurring within 500 hundred feet of the site (Reference 47).

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 due to 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 Oak Ridge and Knoxville, Tennessee, (References 5, Reference 2) and hourly observations from ORR's meteorological Tower J and Tower L near the site (Reference 10). Tower J is approximately 1.1 km southeast of the site and has the wind measurements at 20 meters. Tower L is approximately 1.6 km southeast of the site and has multiple measurement levels at 15 and 30 meters.

Hourly average (scalar) wind speeds at the 20-m level are available for this climate review from Tower J during 2018-2020, from the 10-meter or 15-meter level plus the 30-meter level during 2016-2020 from the Tower L, from the 10-meter level at the Oak Ridge airport station during 1999-2020, and from the 10-m level at the Knoxville Airport during 1981-2020. The anemometer on Tower L was located at the 10-m level from January 2016-October 2017 and was moved in November 2017 to the 15-meter level where it remained through May 2021. The wind data from all 5 years were analyzed together.

The maximum hourly average wind speed for the three years of data analyzed (2018-2020) at Tower J is 24.8 mph. The maximum hourly average wind speed for the five years of data analyzed (2016-2020) at Tower L is 21.4 mph at the 10-m or 15-meter level and 24.4 mph at the 30-meter level. In comparison, Oak Ridge has a maximum hourly average wind speed of 29.0 mph, and Knoxville has a maximum hourly average wind speed of 60 mph. Tower L recorded a peak wind speed of 78.3 mph at the 15-meter level and 84.5 mph at the 30-meter level. Oak Ridge recorded a peak wind speed of 53 mph, and Knoxville recorded a peak wind speed of 68 mph.

For a 100-year return period, the fastest mile of wind in the site area is approximately 90 mph (Reference 11).

2.3.1.6 Precipitation Extremes

Historical precipitation data for the site were obtained from several surrounding National Weather Service (NWS) and Tennessee Valley Authority (TVA) sites (Reference 5, Reference 12, Reference 13, Reference 14, Reference 15), and are summarized in Table 2.3-1. 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 suggest that these statistics are reasonably representative of precipitation extremes that might be expected at the site. Droughts are uncommon in the vicinity of the site. Records indicate that 16 episodes of severe drought have occurred in the past 200 years. The worst was the decade of the 1980s, the driest overall period in the state’s history. Several severe heat waves hit the continental United States throughout the 1980s, including Tennessee, causing severe to extreme drought conditions in eastern Tennessee as classified by the Palmer z-Index (Reference 16, Reference 17, Reference 18, Reference 19).

The estimated annual precipitation is in the range of 47–53 inches. The maximum 24-hour rainfall is less than 10 inches, and the maximum monthly rainfall is less than 20 inches (see Table 2.3-1 for details). The probable maximum precipitation (PMP) is discussed in Section 2.3.2.6.

The average annual snowfall in the vicinity of the site is less than 12 inches. Normal and extreme snowfall events are discussed in Subsection 2.3.1.11.

2.3.1.7 Tornadoes

The probability of a tornado occurring at the site is low based on records from the NWS Morristown Tornado Database (Reference 20) and the NCDC Storm Events Database (Reference 7). During the 73-year period of 1950-2022, five tornadoes were reported within 10 miles of the site (Table 2.3-2). The intensities ranged from F0/EF0 to F3/EF3.

Based on the tornado strike probability presented in NUREG/CR-4461 (Reference 22), the number of tornado events from 1950 through August 2003 within a 2-degree box surrounding the site is 226. This gives an annual average of four tornado events striking somewhere within the 2-degree box.

2.3.1.8 Hurricanes

Hurricane winds are mainly a concern for coastal locations as shown by the wind speed contours presented in Regulatory Guide 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants" (Reference 21), and NUREG/CR-7005, "Technical Basis for Regulatory Guidance on Design-Basis Hurricane Wind Speeds for Nuclear Power Plants" (Reference 23). Regulatory Guide 1.221 is for power reactors and not applicable, but is used as guidance specifically on hurricane wind speed. Due to 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 site. The wind speed contours in Regulatory Guide 1.221 and NUREG/CR-7005 stop well short of the site location with a wind speed contour of 130 mph.

Due to the significant inland distance from both the Atlantic Ocean and the Gulf of Mexico (more than 300 miles), tropical storm impacts are rare at the site and are mostly from storm remnants. Impacts are generally restricted to flood effects from heavy rains (addressed in Subsection 2.3.1.6). From 1905 to the present, there have been 12 tropical storms within a 50-mile radius of the site. Although some of these were originally classified as hurricanes, all were classified as tropical storms when they reached the site area.

In 2021 two tropical systems passed within a 50-mile radius of the site. Both were tropical depressions when they passed through the area. In 2022 a system which had been classified as a hurricane came within a 50-mile radius of the site, but by the time it reached the area, it was downgraded to a tropical depression, and dissipated over eastern Tennessee.

A review of the NCDC Storm Events Database for the period of January 1, 1950, through December 31, 2022, shows that there was only one tropical storm on September 16, 2004, near Roane County, and it caused minimal damage. This storm was associated with Hurricane Ivan.

2.3.1.9 Winter Storm Events

The maximum reported 24-hour snowfall depth at Knoxville (Reference 6) reported during the 61-year period of record was 23.3 inches in February 1960. Snowfall records for stations around the site (Table 2.3-1) show a maximum 24-hour snowfall of 20 inches (March 1993) at Chattanooga (Reference 12).

Frost penetration depth is important for protection of water lines and other buried structural features that are subject to freeze damage. The extreme depth is slightly less than 19.6 inches based on Figure 13 in Reference 24.

2.3.1.10 Ice Storms

Estimations of regional glaze probabilities have been made by Tattelman and Gringorten (Reference 25). For Region V, which contains Tennessee, storms with ice greater than or equal to 1 inch of ice occurred five times in 50 years and storms with ice greater than or equal to 2 inches of ice occurred two 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 for 25- and 50-year return periods, and 1.4 inches for a 100-year return period.

Based on the data provided in American Society of Civil Engineers (ASCE) Standard No. 7-10 (Reference 26), Figure 10-2, the specification for calculating the ice load on a structural element is: the 50-year mean recurrence interval of uniform ice thickness due to freezing rain for Roane County is 0.75 inches with a concurrent 3-second wind gust of 30 mph.

For glaze ice, the point probabilities for ice thicknesses are about 0.20 for greater than or equal to 0.5 inches and 0.36 for greater than or equal to 0.25 inches. These probabilities correspond to recurrence intervals of once in 5 years and once in 3 years, respectively. Glaze ice thicknesses less than or equal to 0.5 inches generally results in little structural damage. However, storms that produce these lesser ice thicknesses 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 7-1 of ASCE 7-10, the 50-year mean recurrence interval snowpack for the Oak Ridge area is determined to be 10 pounds per square foot (psf). Converting this to a 100-year return period snowpack, using the 1.22 adjustment factor presented in Table C7-3 of ASCE 7-10 results in the 100-year return period snowpack determined to be 12.2 psf.

From the maximum reported snow depth at Chattanooga, Tennessee (Reference 27), the highest snow depth at a nearby NWS station, was used to estimate the weight of the maximum historical snowpack at the site. The greatest snow depth reported during the 77-year period of record (1938–2014) for Chattanooga was 19 inches in March 1993. Interim Staff Guidance (ISG) on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures (ISG-7) (Reference 28), provides an algorithm (below) for converting historical maximum snowpack depth to a ground snow load.

$$L = 0.279D^{1.36} \qquad \text{(Equation 2.3.1-3)}$$

Where:

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

Using the 19-inch snow depth for Chattanooga 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 Oak Ridge is 15.7 inches during a March 1960 snowstorm (Reference 29, Reference 20) and 18.8 inches for Knoxville during a February 13–14, 1960 snowstorm (Reference 30, Reference 31). The historical maximum snowfall event for a 48-hour period was determined to be 28 inches recorded in Westbourne, Tennessee, from February 19, 1960 to February 21, 1960 (Reference 31). 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.

$$L = 0.15 \times S \times 5.2 \qquad \text{(Equation 2.3.1-4)}$$

Where:

L is the snow load in psf and S is the snowfall depth in inches.

Using the maximum 100-year return snowfall event of 18.8 inches results in a snow load of 14.7 psf. Using a 28-inch historical maximum snowfall event for a 48-hour period results in a snow load of 21.9 psf.

The Normal Winter Precipitation Event, defined as the maximum ground-level weight (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 21.9 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 21.9 psf.

From the National Oceanic and Atmospheric Administration (NOAA), "Hydrometeorological Report No. 53," (Reference 32), the 48-hour Probable Maximum Winter Precipitation (PMWP) (January through March) for a 10-square-mile area is estimated to be 23.5 inches by logarithmic interpolation. The March PMWP was utilized since the historically highest 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 the plant. The following parameters have been calculated:

- Maximum dry-bulb temperatures at 0.4 percent, 2 percent, and 5 percent annual exceedance levels
- Mean coincident wet-bulb temperatures at 0.4 percent, 2 percent, and 5 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
- 100-year return maximum dry-bulb, mean coincident wet-bulb, maximum non-coincident wet-bulb, and minimum dry-bulb temperatures

Meteorological data from the Chattanooga Lovell Airport was obtained from the NOAA NCDC for use in determining extreme values. This data is the best available long-term data record because the data record for Oak Ridge is incomplete (data gap between 1985 and 1999).

Annual exceedance and 100-year maximum values for dry-bulb and wet-bulb temperatures of 0.4 percent, 2 percent, and 5 percent will be used in the design basis for safety-related ventilation and heat removal system design for the site.

Sixty-six years of raw climatological data were obtained from NOAA/NCDC for the Chattanooga Lovell Airport. This data set contains hourly 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 2.3-3 to Table 2.3-5. Similar evaluations were performed using the NOAA/NCDC data for Knoxville. Because the Chattanooga data produced more conservative (higher temperature) results, these results are used as the design basis.

Monthly climate data for 2017 were found in the American Society of Heating, Refrigerating, and Air Conditioning Engineers (ASHRAE) Handbook - Fundamentals (Reference 33) for Chattanooga Airport and for the Oak Ridge Automated Surface Observing System station. The monthly design dry bulb temperatures with mean coincident wet bulb temperatures and the monthly design wet bulb temperatures are presented in Table 2.3-6 to Table 2.3-9, for annual exceedances listed above. The Chattanooga data produces slightly more conservative results than the Oak Ridge data, but both data sets are very similar, so Chattanooga data are used as the design basis.

2.3.1.13 Meteorological Data for Evaluating Ultimate Heat Sink

The reactor does not rely on an external water source as its ultimate heat sink (UHS), 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 UHS 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 time and spatial scales. The timing, magnitude, relative contributions to, and implications of these changes are generally more speculative, even for specific areas or locations. Further, the most extreme projected changes are for time scales much longer than the approximate 11-year license period for the facility.

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 site are highly uncertain at best, based on current understanding and modeling of global climate change. Predictions provided by the U.S. Geological Survey (USGS) (Reference 34) vary considerably. For example, one model (the BNU-ESM model) gives a summer maximum temperature increase from approximately 89°F to 93°F with a standard deviation of approximately 3°F over the period of 2025 through 2049.

The Southern Climate Impacts Planning Program is a climate hazards research program whose mission is to help Tennessee residents increase their resiliency and level of preparedness for weather extremes now and in the future. Their research (Reference 35) provides roughly consistent predictions relative to the USGS of average temperature increases between 2010 and 2100 of 4-8°F. This climate prediction also indicates more extreme precipitation events that could have an effect on the threat of flooding potential in general.

The ORR, located in Roane and Anderson Counties in east Tennessee about 25 mi (40 km) west of Knoxville, is managed by the DOE. ORR issues Annual Site Environmental Reports (ASERs), available at https://doeic.science.energy.gov/ASER/. Appendix B of the most recent ASER (for 2019) contains a substantial review of the regional climate for the ORR, including a discussion of climate change trends in Section B.1.

Although the long-term climate trend from multiple sources indicates a moderate increase in the average temperature and possibility of extreme precipitation events, as stated above, through the end of the 21st century, the time scale of the licensing period is a minor fraction of this projection period.

2.3.2 Local Meteorology

2.3.2.1 Local Meteorological Data Overview

The facility is located at the southeast portion of the site of the former K-33 building of the East Tennessee Technology Park (ETTP) complex. Since the 1940s, this site has been under the jurisdiction of the Atomic Energy Commission (AEC), which became the Department of Energy (DOE) for this function. In the late 1940s, at the request of the AEC, the United States Weather Bureau conducted, for the first time, a meteorological survey of the Oak Ridge, Tennessee, area to provide detailed information regarding wind flow patterns and other factors to determine dispersion of radioactive contaminants (Reference 36). This study led to the establishment of an extensive network of meteorological towers and forecast capability that is still in existence today. A more recent study of the meteorological patterns in the ORNL area was completed in 2011 (Reference 37). The network of meteorological observations provides a strong basis for the onsite meteorological data needed for the site as well as the facility.

For the period of meteorological analysis using local meteorological towers (2018-2019), the ORNL operated several meteorological towers that provided data on meteorological conditions and on the transport and dispersion aspects of the atmosphere. Data collected at the towers (available at https://metweb.ornl.gov/page1.htm) are used by the DOE in routine dispersion modeling to predict impacts from facility operations and as input to emergency response atmospheric models, which are used for simulated and actual accidental releases from a facility. Data from the towers are also used to support various research and engineering projects. The relevant meteorological towers and their instrumentation and operation are discussed in Subsection 2.3.3.

Environmental monitoring is performed within the ORR, including the ETTP, to measure radiological and non-radiological parameters directly in environmental media adjacent to the facilities. Data from the environmental monitoring program are analyzed to assess the environmental impact of DOE operations on the entire reservation and the surrounding area.

Meteorological data are collected at different levels above the ground, to 60 meters at some towers, to assess the vertical structure of the atmosphere, particularly with respect to wind shear and stability. Stable boundary layers and significant wind shear zones (associated with the local ridge and-valley terrain and the Great Valley of Eastern Tennessee) can significantly affect the movement of a plume after a facility release. Data are collected at the 10- or 15-meter level at most towers, but the wind measurement height is 20 meters for Tower J. Data are collected at some towers at 30-, 33-, 35-, and 60-meter levels. Temperature, relative humidity, and precipitation are measured at some sites at 2 meters, but wind speed and wind direction typically are not. Barometric pressure and solar radiation are measured at one or more of the towers. Instrument calibrations are managed by the University of Tennessee-Battelle, LLC (UT-Battelle), and are performed every 6 months by an independent auditor.

Topography around the site strongly influences the local climate, as shown by Figure 2.3-2 for areas near the site, and in Figure 2.3-3 out to 100 km from the site. Mountain ranges located both northwest and southeast of the site are oriented generally northeast-southwest. The Appalachian Mountains to the east and southeast provide an orographic barrier that reduces the low-level atmospheric moisture from the Atlantic Ocean brought into the area by winds from the east. However, considerable low-level atmospheric moisture from the Gulf of Mexico is often brought into the area by prevailing winds from the south, southwest, or west.

The site is located at an elevation of approximately 765 feet above mean sea level. The site is situated between the Clinch River to the east and McKinney Ridge to the east-northeast. On the southeastern edge of the DOE Oak Ridge area, approximately 1.2 miles from the site is a small area of mountains just over 900 feet in elevation above sea level. Terrain to the south and north of the site is characterized as alternating ridges and valleys oriented along a southwest-to-northeast axis, as shown in Figures 2.3-2 and Figure 2.3-3. McKinney Ridge, Black Oak Ridge, and the ridges to the south/southeast reach an elevation over 1,100 feet above sea level (approximately 300 feet above plant grade). The closest ridge is the Black Oak Ridge acting as the northern boundary of the site. There is a significant gap in the southern ridges to the south of the site (Clinch River Gap). Figures 2.3-4 through 2.3-11 show the elevation profiles within 50 miles of the site in each of eight compass directions (at 45-degree intervals).

The geographic orientation of the ridges and valleys generally aligns with the prevailing regional winds from the southwest, but the gaps in the ridges permit wind flow from other directions as well.

Meteorological measurements from three different towers near the site were reviewed: ORR Tower J, Tower L, and Tower D. Tower J is located approximately 1.08 km southeast of the site (Figure 2.3-2), and has had meteorological instruments at 20 meters. Tower L is located approximately 1.53 km south-southeast of the site, and had meteorological instruments located at various heights during the course of its operation, including 2 meters, 10 meters, 15 meters, and 30 meters. Tower D is located approximately 7.17 km southeast of the site, and has meteorological instruments at 15 meters, 35 meters, and 60 meters. Tower L was shut down on May 6, 2021, but Towers J and D continue to operate. See Table 2.3-10 and Subsection 2.3.3.1 for more details on these towers.

The combination of high pressure associated with the Azores-Bermuda anti-cyclonic circulation and the nearby ridges result in generally light wind speeds. Average surface wind speeds for the site are 4.4 mph at a height of 20 meters above ground level based on ORR Tower J. Tower L observed an average wind speed of 3.4 mph at 15 meters, and Tower D observed an average wind speed of 3.1 mph at 15 meters. These average wind speeds are based on years 2018-2019 wind data at the lowest wind levels for each tower.

Data from all three towers were used to evaluate the impact of topography on the site. Tower L is slightly closer to the ridge that is south of the site than Tower J, and, therefore shows slightly more influence from terrain than Tower J. Tower D is within 1 km of Chestnut Ridge (immediately to the north and west of the tower) and Haw Ridge (to the south and east). The terrain channeling is from southwest to northeast for all of these towers. The principal impact of the terrain channeling is on wind directions as shown in the wind roses (Figures 2.3-12 through 2.3-17). The wind roses have a similar pattern of winds with the prevailing winds coming from the south-southwest to west-southwest directions and northeast to east-northeast directions.

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.

The historical studies from 1953 and 2011 noted above indicate basic flow patterns that have been in place during the recorded weather history of the ORR area. Therefore, it is concluded that meteorological characteristics for the site have not changed significantly over time and are not expected to change over the life span of the project.

Comparing data from site-specific meteorological towers helps to determine if the site is consistent with regional conditions. Data were examined for Towers J, L, and D for the 2-year period of calendar years 2018 and 2019. There is generally good agreement between these local towers and regional offsite locations for the average values. These comparisons indicate that, for these variables, data from the site is consistent with overall meteorological conditions in the Oak Ridge to Knoxville area.

2.3.2.3 Winds

During the January 1, 2018, to December 31, 2019, period, 20-meter wind data was collected by the meteorological Tower J, and 15-meter and 30-meter wind data were collected by the meteorological Tower L, both at the site. During the same time period, a nearby regional station, Tower D, collected wind data at 15 meters, 35 meters, and 60 meters. In November 2017, the lower anemometer height for Tower L was moved from 10 meters to 15 meters to increase siting fetch due to the close proximity of surrounding buildings. Tower D is the closest 60-meter tower in the vicinity of the site, and its data has been used for a consistency check data from Tower L due to the multiple levels of data.

<u>Average Wind Direction and Wind Speed Conditions</u>

The tower data for the project area are presented as wind roses in Figures 2.3-12 through 2.3-17. A wind rose for Chattanooga, based on 10 years of data (2000-2009), is presented in Figure 2.3-18 and a wind rose for Oak Ridge, based on 10 years of data (2000-2009), is presented in Figure 2.3-19.

Wind speeds at the ORR meteorological towers near the site during 2018-2019 (Table 2.3-11) were generally light with an average 20-meter speed of 4.4 mph at Tower J, an average 15-meter wind speed of 3.5 mph at Tower L, and an average 15-meter wind speed of 3.1 mph at Tower D. The maximum hourly average (scalar) wind speed was 24.8 mph at Tower J, 21.4 mph at Tower L, and 16.8 mph at Tower D. At higher levels, Towers L and D also show similar wind behavior. The 30-meter level on Tower L has an average wind speed of 4.1 mph and a maximum hourly average wind speed of 24.4 mph. The 35-meter level on Tower D has an average wind speed of 3.9 mph and a maximum hourly average wind speed of 21.1 mph, while the 60-meter level on Tower D has an average wind speed of 5.0 mph and a maximum hourly average wind speed of 26.2 mph. The geographic orientation of the ridges and valleys generally aligns with the prevailing regional winds from the southwest and northeast, but the gaps in the ridges permit wind flow from other directions as well as noted in the wind roses. The combination of high pressure associated with the Azores-Bermuda anticyclonic circulation and the nearby ridges result in generally light wind speeds with average surface wind speeds for the site being less than or equal to 5 mph. The site is surrounded by complex terrain, with alternating ridges and valleys oriented along a southwest (SW) to northeast (NE) axis. The local wind patterns are influenced by the complex terrain, with up-valley (SSW-WSW)/down-valley (NE-ENE) flow patterns common, and stable conditions with light winds frequently observed as seen at all levels of all three meteorological towers. These flow patterns influence the dispersion around the site.

<u>Wind Direction Persistence</u>

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 2.3-12), the maximum persistence at 15-meters for Tower L for the 2018–2019 time period is 19 hours from NE for the same sector, 39 hours from W-NW for ± 1 sector, and 69 hours from WSW-NNW for ± 2 sectors.

The wind data show a consistent pattern of wind directions with predominant winds from the SSW-SW, with a second maximum of persistent winds from the opposite direction (basically, from the NE-ENE). There is seasonal variation in this pattern (Figure 2.3-20). There is also a diurnal pattern with the winds. During the day (Figure 2.3-21) the winds show the SSW-SW and NE-ENE patterns of predominant winds. During the night (Figure 2.3-22) the winds show the flow coming off of the terrain to the south and east.

2.3.2.4 Air Temperature

Temperature data for Knoxville (Reference 12) and Oak Ridge (Reference 15) are presented in Table 2.3-13 and Table 2.3-14, respectively. Normal temperatures have ranged from the upper 30s (°F) in the winter to the upper 70s in the summer at both locations. Normal daily maximum temperatures ranged from about 47°F in mid-winter to about 88°F in mid-summer. The normal daily minimum temperatures ranged from about 29°F in mid-winter to about 69°F in mid-summer. The extreme daily maxima recorded were 105°F (June and July 2012) at Knoxville and 105°F (July 1952 and June 2012) at Oak Ridge, while the extreme daily minima (during January 1985) were -24°F and -17°F, respectively.

Temperatures measured by Tower L for 2018–2019 (Reference 8) are presented in Table 2.3-15. Tower L shows a similar pattern of daily average temperatures ranging from the mid-20s (°F) in winter to upper 70s (°F) in summer. Normal daily maximum temperatures ranged from about 59.0°F in mid-winter to about 79.1°F in mid-summer. The normal daily minimum temperatures ranged from about 25.3°F in mid-winter to about 71.5°F in mid-summer. A maximum temperature of 96.2°F and a minimum temperature of 0.5°F were recorded over the 2-year period.

2.3.2.5 Atmospheric Moisture

Long-term relative humidity and absolute humidity data for Knoxville and Oak Ridge are presented in Table 2.3-16. Short-term humidity data based on measurements at the ORR meteorological Tower L are summarized in Table 2.3-17. The humidity data among the three sites (Knoxville, Oak Ridge, and the site) are compared in Table 2.3-16 and Table 2.3-17. site data are comparable to the long-term data. The dew points and humidity data are a little higher for the 2018-2019 Tower L period than for the longer-term Knoxville and Oak Ridge data periods.

2.3.2.6 Precipitation

The summary provided in this subsection is taken in large part from the Clinch River Nuclear Site planning documents due the proximity of that project to the site.

Rain

Hourly precipitation observations are available from the Oak Ridge NWS station (approximately 12 miles northeast of the site). The long-term observations from the precipitation data from Oak Ridge (Reference 15) are presented in Table 2.3-18. Precipitation falls an average of about 125 days per year, and the normal annual precipitation is nearly 51 inches. The maximum monthly rainfall has ranged from about 7 inches to just over 19 inches. The minimum monthly amount was a trace in October 1963. The maximum in 24 hours was 7.48 inches in August 1960. With the exception of late-summer/early-autumn (which are slightly drier), precipitation is fairly uniformly distributed through the year. July and March are normally the wettest months of the year.

Precipitation data from the nearby Towers J and L (Reference 8, Table 2.3-19) indicate more than normal precipitation during 2018 and 2019. Maximum rainfall, estimated by statistical analysis of regional precipitation data, is given in Table 2.3-20 for return periods of 1 to 100 years and for rainfall durations from 5 minutes to 10 days. These data were taken from NOAA Atlas 14, Volume 2, Version 3 (Reference 38).

The 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 site area, using a 100-year return period, the PMP for 6, 12, 24, and 48 hours is 4.7, 5.7, 6.8, and 8.3 inches, respectively (see Table 2.3-20). Approximately 49 thunderstorms occur in a typical year (Reference 14). Thunderstorm activity is most predominant in the spring and summer seasons, and the maximum frequency of thunderstorm days is normally in July (Table 2.3-18).

Snow

Appreciable snowfall is relatively infrequent in the area. Snowfall data are summarized in Table 2.3-21 for Knoxville and Oak Ridge. Normal annual snowfall has ranged from about 6.5 inches at Knoxville to about 11 inches at Oak Ridge. Generally, significant snowfalls are limited to December through March. Respective 24-hour maximum snowfalls have been 18 and 12 inches at Knoxville and Oak Ridge.

Precipitation Wind Roses

Figure 2.3-23 shows composite 2018-2019 precipitation and wind directions (vector) data from Tower L. Precipitation is most often associated with wind directions from SSW-SW, corresponding to the predominant wind flow direction sectors. There is a secondary maximum with wind directions from NE-ENE.

2.3.2.7 Fog

Fog data for Knoxville and Oak Ridge are presented in Table 2.3-22. These data indicate that heavy fog (visibility ≤ 1/4 mile) occurs about 30 days per year at Knoxville and 52 days per year at Oak Ridge, with the autumn normally the foggiest season. The site has conditions more similar to Oak Ridge due to proximity.

2.3.2.8 Atmospheric Stability

The frequency of occurrence of Pasquill (classes A-G) atmospheric stability classes based on vertical temperature difference for local ORR meteorological Tower L over a 2-year period (2018-2019) is presented in Table 2.3-23. While the atmosphere at the site for the 2 years analyzed appears to be almost equally stable, neutral, and unstable, the stable lapse conditions (classes E, F, and G - i.e., inversions) occur the majority of the time (42 percent). However, the majority of the stable lapse conditions are only slightly stable (class E), occurring 27 percent of the time. The most stable class (class G) occurs approximately 5.5 percent of the time. Neutral lapse conditions (class D) occur approximately 27 percent of the time. Unstable classes (A, B, and C) occur approximately 31 percent of the time.

2.3.2.9 Inversion Persistence

Table 2.3-24 presents a summary of onsite inversion persistence data, with a breakdown by stability class, at Tower L for 2018-2019. Inversion persistence is defined as two or more consecutive hours of a single stable class (or combination of stable classes). The longest contiguous period of inversion conditions lasted 215 hours.

2.3.2.10 Mixing Heights

Holzworth (Reference 41) provides estimated monthly mean maximum heights for Nashville, Tennessee (the NWS upper air site closest to the site). Seasonal and annual estimates of rural mixing heights for the site are as follows:

- Winter (December, January, February) – 563 meters (morning), 1,123 meters (afternoon)
- Spring (March, April, May) – 606 meters (morning), 1,783 meters (afternoon)
- Summer (June, July, August) – 441 meters (morning), 1,874 meters (afternoon)
- Autumn (September, October, November) – 357 meters (morning), 1,473 meters (afternoon)
- Annual – 492 meters (morning), 1,563 meters (afternoon)

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

Plant systems have a limited potential to noticeably affect local meteorology. The 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 (see Section 6.3). While there would be some steam plumes due to heat rejection exhaust, these would be hot exhaust streams that would rapidly evaporate when mixed with ambient air. There would be some minor air quality and visibility impacts on local air quality during construction, although the impacts would 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

The facility uses existing meteorological monitoring and measurements taken within the ORR.

Tower L is the closest multiple-level tower in the vicinity of the site. It is about 1.55 km from the site with no intervening terrain. Photos of the tower are provided in Figure 2.3-25 and Figure 2.3-26. A 2-year period of full calendar years from January 1, 2018–December 31, 2019 is selected for characterization of the wind patterns and for short-term modeling as the representative meteorological input data. Validated data from all of the ORR meteorological towers is available at https://metweb.ornl.gov/page5.htm. Wind roses for the 15-meter and 30-meter levels of Tower L for 2018–2019 are provided in Figure 2.3-13 and Figure 2.3-14, respectively.

Instrumentation on Tower L consists of the following:

- RM Young 81000 3-D sonic wind monitor 15 meter and 30 meter
- RM Young Temperature / RH 41382F 2 meter, 15 meter, and 30 meter
- Epply 8-48 Solar Radiation 15 meter
- ESC BPM 24/32 2 meter
- Texas Instruments Precipitation Ground

The closest current backup / alternative meteorological tower with multiple levels is Tower D, located about 3.5 miles from the site, and it has wind measurements at the 15-meter, 35-meter, and 60-meter levels. Wind roses for the 15-meter, 35-meter, and 60-meter levels of Tower D for 2018–2019 are provided in Figure 2.3-15, Figure 2.3-16, and Figure 2.3-17, respectively. In general, the wind roses from Towers L and D indicate similar wind patterns at both towers, with the predominant flow along the axis of the valley.

Per the Oak Ridge Reservation Annual Site Environmental Reports available at Home of the Oak Ridge Reservation Annual Site Environmental Report (ASER) (https://doeic.science.energy.gov/ASER/), ORR meteorological monitoring satisfies onsite monitoring requirements for the DOE (Reference 42) and the U.S. Environmental Protection Agency (EPA) (Reference 43). Instrument calibrations are managed by UT-Battelle and are performed quarterly or semi-annually, and are traceable to National Institute of Standards and Technology standards.

On May 6, 2021, Tower L was permanently shut down. Other available sources of meteorological data to determine wind direction, wind speed, temperature, and stability class for modeling purposes are listed below.

- Tower J measurements of wind and temperature
- Tower D measurements of temperature and stability class
- Computer-assisted Protective Action Recommendation System (CAPARS) wind field prediction system (Reference 44) predictions of meteorological variables needed for input to modeling

2.3.4 Short-Term Atmospheric Dispersion Modeling for Accidental Releases

This subsection addresses short-term dispersion modeling approaches for assessing the atmospheric dispersion factors (χ/Q) to evaluate dose consequence of postulated releases resulting from accidents.

The short-term dispersion modeling uses ARCON96 with the atmospheric dispersion methodology as outlined in the KP-FHR Mechanistic Source Term Methodology Topical Report (Reference 45), which is applicable for dispersion distances up to 1,200 meters.

The hourly meteorological data input to ARCON96 consists of the wind direction and speed from two measurement levels, and the stability class.

The Tower L hourly meteorological data from 2018 and 2019, which constitutes two complete annual cycles, are used to calculate the X/Q values at the EAB and LPZ. As discussed in Section 2.3.2.8, the stability class is determined using the vertical temperature difference method. The Tower L temperature measurements taken at the 15-meter and 30-meter levels are converted to ˚C/100m before the Pasquill stability class is determined using Table 2.3-25. Tower L is the closest meteorological tower to the site with two measurement levels, located only about 1 mile south-southeast of the site. Tower L data is used to perform the analysis due to the proximity of Tower L with no terrain obstacles between the tower and the site.

The meteorological data from Tower L is available at the ORR meteorology website at https://metweb.ornl.gov/page5.htm. The provided data are 100 percent complete, although some substitution of data from collocated instrumentation has taken place to handle missing values for individual sensors.

2.3.5 Long-Term Atmospheric Dispersion Estimates for Routine Releases

Details regarding the long-term dispersion modeling, the modeling inputs, and the interpretation of the modeling results will be provided in the application for an Operating License.

2.3.6 References

1. Tennessee Valley Authority, Clinch River Nuclear Site Early Site Permit Application, Part 2, Site Safety Analysis Report, Revision 2, March 2019.
2. National Climatic Data Center, 1999-2020 Annual Local Climatological Data for Oak Ridge, TN. Retrieved from https://www.ncdc.noaa.gov/cdo-web/datatools/lcd.
3. Oak Ridge Reservation Annual Site Environmental Report (ASER), Appendix B. Climate Overview of the Oak Ridge Area. 2019. Retrieved from https://doeic.science.energy.gov/ASER/.
4. National Climatic Data Center, Data Tools: 1991-2020 Normals for Oak Ridge, TN, Retrieved from https://www.ncdc.noaa.gov/cdo-web/datatools/normals. Accessed March 2023.
5. National Climatic Data Center, 1991-2020 Annual Local Climatological Data for Knoxville, TN. Retrieved from https://www.ncdc.noaa.gov/cdo-web/datatools/lcd. Accessed March 27, 2023.
6. Tennessee Department of Environment and Conservation, Air Pollution Control Division, "2020 Tennessee Annual Monitoring Network Plan." July 1, 2020.
7. National Climatic Data Center, Storm Events Database, Website: http://www.ncdc.noaa.gov/stormevents. Accessed March 27, 2023.
8. ORNL Thunderstorm Days 2001-2021. Oak Ridge National Laboratory, Oak Ridge Reservation Meteorology – Climate Data, Normals, and Extremes – Severe Weather Statistics – Oak Ridge Area. Retrieved from https://metweb.ornl.gov/~krbirdwell/web/TStorms_ORNL_2021.pdf. Accessed May 16, 2023.
9. Koehler, Thomas L., Cloud-to-Ground Lightning Flash Density and Thunderstorm Day Distributions Over the Contiguous United States Derived from NLDN Measurements: 1993-2018, Monthly Weather Review, Figure 7. Vol. 148, pp 313-332. 2020.
10. Oak Ridge National Laboratory, Oak Ridge Reservation Meteorology – Climate Data, Normals, and Extremes – Annual, Hourly Climate Data – Oak Ridge Reservation (2001-2020). Retrieved from https://metweb.ornl.gov/page5.htm. Accessed May 3, 2023.
11. Thom, H. C. S., New Distributions of Extreme Winds in the United States, Journal of the Structural Division, Proceedings of the American Society of Civil Engineers, Figure 5, Vol. 94, No. ST 7, pp 1787-1801. 1968.
12. National Climatic Data Center, Chattanooga, TN daily average precipitation data for years 1938 through 2014.
13. National Climatic Data Center, 2013 Annual Local Climatological Data (LCD) for Nashville, TN.
14. National Climatic Data Center, 1998 Annual Local Climatological Data (LCD) for Oak Ridge, TN.
15. National Climatic Data Center, 2013 Annual Local Climatological Data (LCD) for Oak Ridge, TN.

16. The Tennessee Encyclopedia of History and Culture, Tennessee Historical Society. Retrieved from https://tennesseeencyclopedia.net/.
17. Homeland Security Digital Library – information on the 1980 and 1988 heat waves and droughts. Retrieved from https://www.hsdl.org/c/tl/1980-us-heat-wave/ and https://www.hsdl.org/c/tl/1988-u-s-droughtheatwave/.
18. Ross, Tom and Lott, Neal. A Climatology of 1980-2003 Extreme Weather and Climate Events. National Climatic Data Center, Technical Report No. 2003-01. National Climatic Data Center, Asheville, NC. 2003. Retrieved from https://www.ncdc.noaa.gov/monitoring-content/billions/docs/lott-and-ross-2003.pdf.
19. Karl, Thomas R., Young, Pamela J, "The 1986 Southeast Drought in Historical Perspective," Bulletin of American Meteorological Society, Volume 68, Issue 7, pp 773-778. July 1987.
20. National Weather Service, Morristown Tornado Database, Website: http://www.midsouthtornadoes.msstate.edu/index.php?cw=mrx. Accessed November 13,2014.
21. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.221, Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants. October 2011.
22. NUREG/CR-4461, Revision 2, "Tornado Climatology of the Contiguous United States." February 2007.
23. U.S. Nuclear Regulatory Commission, Technical Basis for Regulatory Guidance on Design-Basis Hurricane Wind Speeds for Nuclear Power Plants, NUREG/CR-7005. November 2011.
24. National Oceanic and Atmospheric Administration Manual NOS NGS 1, "Geodetic Bench Marks", NOAA. September 1978.
25. Tattelman, Paul and Gringorten, Irving I., "Estimated Glaze Ice and Wind Loads at the Earth's Surface for the Contiguous United States," Air Force Cambridge Research Laboratories. October 1973.
26. American Society of Civil Engineers, AMSI/ASCE 7-10, "Minimum Design Loads for Buildings and Other Structures." 2011.
27. National Weather Service, Chattanooga, TN Climate Page. Records from 1879-Present. Retrieved from https://www.weather.gov/mrx/chaclimate.
28. Nuclear Regulatory Commission, Interim Staff Guidance (ISG) on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures (ISG-7), Issued Final. 2009.
29. Snow Plow News, Winter Weather News/Snow Research. Retrieved from https://blog.snowplownews.com/snowfall_records/oak-ridge-tennessee/.
30. Huotari, John, "Snowfall Again!", Oakridger. 16 February 2020. Retrieved from https://www.oakridger.com/article/20100216/NEWS/302169991.
31. Tennessee Valley Authority Division of Water Control Planning, "Snow and Ice Storms of 1959-1960 in Tennessee River Basin," Tennessee Valley Authority Division of Water Control Planning Hydraulic Data Branch. 1960. Found on Google Books Website at https://books.google.com/books?id=RPm7SgRW77kC&source=gbs_navlinks_s.
32. National Oceanic and Atmospheric Administration, 1980, "Hydrometeorological Report HMR-53, Seasonal Variation of 10-Square-Mile Probable Maximum Precipitation Estimates, United States East of the 105th Meridian." U.S. Government Printing Office: 311-046/126. 1980.
33. American Society of Heating, Refrigerating, and Air Conditioning Engineers (ASHRAE) Climate Design Conditions 2009/2013/2017 website. Data from the ASHRAE Handbook – Fundamentals. 2017. Retrieved from http://ashrae-meteo.info/v2.0/.
34. U.S. Geological Survey, National Climate Change Viewer, Summary for Roane County, Tennessee. May 13, 2014. Retrieved from https://www2.usgs.gov/landresources/lcs/nccv/maca2/maca2_counties.html.
35. Climate Change in Tennessee, Retrieved from https://geojones2.files.wordpress.com/2011/08/climatechange_tennessee.pdf.

36. U.S. Weather Bureau of Oak Ridge, TN, A Meteorological Survey of the Oak Ridge Area. Report #ORO-99. 578 pp. 1953.
37. Birdwell, K.R., Wind Regimes in Complex Terrain of the Great Valley of Eastern Tennessee. 693 pp. 2011.
38. National Oceanic and Atmospheric Administration's National Weather Service Hydrometeorological Design Studies Center Precipitation Frequency Data Server (PFDS) – NOAA Atlas 14 Point Precipitation Frequency Estimates for Tennessee, September 23, 2021. Retrieved from https://hdsc.nws.noaa.gov/hdsc/pfds/pfds_map_cont.html?bkmrk=tn
39. Not Used.
40. Not Used.
41. Holzworth, George. C., "Mixing Heights, Wind Speeds, And Potential for Urban Air Pollution Throughout the Contiguous United States," Environmental Protection Agency, January 1972.
42. U.S. Department of Energy, DOE HANDBOOK: Environmental Radiological Effluent Monitoring and Environmental Surveillance. DOE-HDBK-1216-2015. 2015. Retrieved from https://www.standards.doe.gov/standards-documents/1200/1216-bhdbk-2015/@@images/file.
43. U.S. Environmental Protection Agency, Meteorological Monitoring Guidance for Regulatory Modeling Applications. Office of Air Quality Planning and Standards, Research Triangle Park, North Carolina. EPA-454/R-99-005. 2005. Retrieved from https://www.epa.gov/sites/production/files/2020-10/documents/mmgrma_0.pdf.
44. AlphaTRAC, The CAPARS system, Website: . (Examples of the near-real-time (updated every 15 minutes) wind maps at ORNL are provided at https://metweb.ornl.gov/page1.htm.)
45. KP-TR-012-P-A, "KP-FHR Mechanistic Source Term Topical Report" , May 2022.
46. Heritage Center Annual Lightning Strikes Archive. Oak Ridge National Laboratory, Oak Ridge Reservation Meteorology – Climate Data, Normals, and Extremes – Severe Weather Statistics – Oak Ridge Area. Retrieved from https://metweb.ornl.gov/~krbirdwell/web/Lightning/. Accessed May 16, 2023.
47. 2012 East Tennessee Technology Park Lightning Strikes. Oak Ridge National Laboratory, Oak Ridge Reservation Meteorology – Climate Data, Normals, and Extremes – Severe Weather Statistics – Oak Ridge Area. Retrieved from https://metweb.ornl.gov/~krbirdwell/web/Lightning/2012%20All%20strikes%20ETTP.jpg. Accessed May 16, 2023.

Table 2.3-1: 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)	Maximum 24-hour Snowfall (inches)	Maximum Monthly Snowfall (inches)
Oak Ridge NWS Station	30[a]	50.91			11.1		
	66[b]		7.48 (Aug 1960)	19.27 (Jul 1967)			
	52[b]					12.0 (Mar 1960)	21.0 (Mar 1960)
Knoxville NWS Station[b]	30	47.86			6.5		
	72		5.98 (Sep 2011)	12.67 (Jan 2013)			
	69					18.2 (Nov 1952)	23.3 (Feb 1960)
Chattanooga NWS Station[b]	30	52.48			3.9		
	74		9.50 (Sep 2011)	16.32 (Mar 1980)			
	76					20.0 (Mar 1993)	20.0 (Mar 1993)
Nashville NWS Station[b]	30	47.25			6.3		
	74		9.09 (May 2010)	16.43 (May 2010)			
	66					10.2 (Dec 1963)	18.9 (Feb 1979)

[a] Reference 14
[b] Reference 5, Reference 12, Reference 13, Reference 15

Table 2.3-2: Tornados within 10 Miles of the Site

Date	Counties Affected	Magnitude (WS range)	Length (miles)	Width (yards)	Closest Distance to the Site (miles)
2/21/1993	Anderson and Knox	F-3 (158-206 mph)	16	150	10
2/21/1993	Roane, Loudon, and Blount	F-3 (158-206 mph)	30	100	7.24
5/18/1995	Morgan	F-0 (40-72 mph)	0.5	23	5.95
11/10/2002	Morgan	F-3 (158-206 mph)	8.3	300	7.74
6/10/2014	Roane	EF-0 (65-68 mph)	0.5	100	7.05
Source: Reference 7, Reference 20					

Table 2.3-3: Chattanooga Maximum Dry Bulb and Mean Coincident Wet Bulb Temperatures

Annual Exceedance	Description	Temperature (°F)
100-Year Run Period	Dry Bulb Temperature	107.0
	Coincident Wet Bulb Temperature	73.1
0.4%	Dry Bulb Temperature	95.0
	Coincident Wet Bulb Temperature	74.9
2%	Dry Bulb Temperature	90.0
	Coincident Wet Bulb Temperature	73.7
5%	Dry Bulb Temperature	85.0
	Coincident Wet Bulb Temperature	71.8
The maximum dry-bulb temperature that has existed at the site for 2 hours or more combined with the maximum wet-bulb temperature that exists in that population of dry-bulb temperatures. Based on hourly data from NOAA/NCDC for Chattanooga.		

Table 2.3-4: Chattanooga Maximum Wet Bulb Temperatures

Annual Exceedance	Temperature (°F)
100-Year Run Period	83.6
0.4%	77.6
The maximum historical wet-bulb temperature recorded for 2 or more hours. Based on hourly data from NOAA/NCDC for Chattanooga.	

Table 2.3-5: Chattanooga Minimum Dry Bulb Temperatures

Annual Exceedance	Temperature (°F)
100-Year Run Period	-9.9
0.4%	16.0
1.0%	21.0
2.0%	25.0
Based on hourly data from NOAA/NCDC for Chattanooga.	

Table 2.3-6: Chattanooga Monthly Design Dry Bulb and Mean Coincident Wet Bulb Temperatures

Monthly Exceedance	Description	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
0.4%	Dry Bulb Temperature	69.4	72.3	81.5	86.5	90.0	95.4	98.2	98.2	95.5	84.9	77.2	71.2
	Mean Coincident Wet Bulb Temperature	61.2	59.2	63.1	66.6	71.4	73.6	75.9	74.8	70.9	67.1	63.7	62.8
2%	Dry Bulb Temperature	64.9	68.0	77.0	82.9	87.4	92.7	94.3	95.2	91.4	81.3	72.7	65.7
	Mean Coincident Wet Bulb Temperature	58.5	57.0	61.7	64.8	69.6	73.8	75.4	75.0	71.4	66.6	60.1	58.8
5%	Dry Bulb Temperature	61.2	64.0	72.9	79.9	84.9	90.1	92.1	92.3	87.6	78.6	69.1	61.5
	Mean Coincident Wet Bulb Temperature	54.7	54.0	58.3	63.3	69.3	72.9	75.0	74.3	70.9	64.9	59.4	56.3

Source: Reference 33

Table 2.3-7: Chattanooga Monthly Design Wet Bulb Temperatures

Monthly Exceedance	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
0.4%	63.5	63.5	66.4	70.7	75.0	77.5	79.5	78.8	76.5	72.0	68.0	65.7
Source: Reference 33												

Table 2.3-8: Oak Ridge Monthly Design Dry Bulb and Mean Coincident Wet Bulb Temperatures

Monthly Exceedance	Description	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
0.4%	Dry Bulb Temperature	68.0	70.3	80.1	85.8	88.9	93.9	95.4	96.1	92.5	83.7	76.1	68.7
	Mean Coincident Wet Bulb Temperature	60.8	57.4	62.2	65.8	70.3	72.5	75.7	74.1	69.1	67.5	63.1	59.5
2%	Dry Bulb Temperature	63.0	65.7	75.4	82.2	86.2	90.9	92.1	92.8	88.7	80.1	71.2	62.4
	Mean Coincident Wet Bulb Temperature	57.6	55.0	60.1	64.0	68.5	72.7	75.0	74.3	70.5	65.7	59.5	55.9
5%	Dry Bulb Temperature	57.9	60.8	70.9	79.3	84.0	88.3	89.8	90.5	85.8	76.6	67.3	58.3
	Mean Coincident Wet Bulb Temperature	51.8	50.0	57.2	62.4	68.0	71.6	74.3	73.9	70.0	63.5	57.2	53.4

Source: Reference 33

Table 2.3-9: Oak Ridge Monthly Design Wet Bulb Temperatures

Monthly Exceedance	Jan	Feb	Mar	Apr	May	Jun	Jul	Aug	Sep	Oct	Nov	Dec
0.4%	62.2	61.0	64.8	69.1	73.4	77.2	79.2	79.2	75.0	71.1	66.7	61.5
Source: Reference 33												

Table 2.3-10: Meteorological Towers Near the Site

Meteorological Tower	Location	Data Collected	Data Collection Period
ORR Tower J	Latitude: 35.930142° N Longitude: 84.394355° W Elevation: 792 ft-msl UTM: Zone 16 Northing: 3979.338 km Easting: 735.073 km	20-m Wind 20-m Temperature Precipitation	June 1, 2017-Present
ORR Tower L	Latitude: 35.925199 ° N Longitude: 84.394196 ° W Elevation: 750 ft-msl UTM: Zone 16 Northing: 3978.790 km Easting: 735.102 km	60-, 10-m Wind 60-, 10-m Temperature Precipitation Atmospheric Pressure Solar Radiation Relative Humidity 60-, 10-m Sigma-Theta	January 1,2000-December 31, 2000
		30-, 10-m Wind 30-, 10-m Temperature Precipitation Atmospheric Pressure Solar Radiation	January 1, 2001-November 1, 2017 (Precipitation and Atmospheric Pressure missing for 2004)
		30-, 15-m Wind 30-, 15-, 2-m Temperature Dewpoint Precipitation Atmospheric Pressure Solar Radiation 30-, 15-m Sigma-Theta 15-m Relative Humidity	November 1, 2017-May 6, 2021
ORR Tower D	Latitude: 35.924992° N Longitude: 84.324946° W Elevation: 858 ft-msl UTM: Zone 16 Northing: 3978.936 km Easting: 741.352 km	60-, 35-, 15-m Wind 60-, 35-, 15-, 2-m Temperature 15-, 2-m Dewpoint Precipitation Atmospheric Pressure Solar Radiation 60-, 35-m Sigma-Theta 15-, 2-m Relative Humidity	April 1, 2014-Present

Table 2.3-11: Average (Scalar) Wind Speed for the Site (2018-2019)

Tower J	
Quarter	**Average (scalar) 20-m Wind Speed (mph)**
2018	
1st quarter	5.19
2nd quarter	4.59
3rd quarter	3.45
4th quarter	4.17
2019	
1st quarter	5.26
2nd quarter	4.64
3rd quarter	3.11
4th quarter	4.83
Overall	4.40

Tower L	
Quarter	**Average (scalar) 20-m Wind Speed (mph)**
2018	
1st quarter	4.20
2nd quarter	3.74
3rd quarter	2.75
4th quarter	3.25
2019	
1st quarter	4.13
2nd quarter	3.73
3rd quarter	2.64
4th quarter	3.26
Overall	3.46

Tower D	
Quarter	**Average (scalar) 20-m Wind Speed (mph)**
2018	
1st quarter	3.61
2nd quarter	3.38
3rd quarter	2.59
4th quarter	2.92
2019	
1st quarter	3.61
2nd quarter	3.41
3rd quarter	2.38
4th quarter	2.93
Overall	3.10

Table 2.3-12: Wind Direction Persistence for Tower L (2018-2019)

Wind Sector	Maximum Hours of Wind Direction Persistence at 15 m for Tower L		
	Same Sector	+/-1 Sector	+/-2 Sector
N	4	12	32
NNE	8	26	39
NE	**19**	38	49
ENE	8	33	44
E	9	14	33
ESE	6	11	20
SE	5	9	16
SSE	5	13	32
S	8	18	34
SSW	12	34	63
SW	12	36	45
WSW	6	27	57
W	12	36	54
WNW	16	**39**	**69**
NW	7	29	50
NNW	7	12	29

Notes:
Bold indicates the maximum values.
Grey fill indicates the sector range.
Data Period: January 1, 2018 – December 31, 2019

Table 2.3-13: Air Temperatures for Knoxville, Tennessee

	Normal Daily Maximum	Normal Dry Bulb	Normal Daily Minimum	Extreme Daily Maximum	Extreme Daily Minimum
Period of Record (yrs)	**30[(a)]**	**30[(a)]**	**30[(a)]**	**72**	**72**
January	47.3	38.2	29.2	77	-24[(c)]
February	52.3	42.4	32.4	83	-8
March	61.4	50.3	39.2	86	1
April	70.3	58.8	47.3	92	22
May	78.1	67.2	56.2	96	32
June	85.4	75.0	64.7	105[(b)]	43
July	88.2	78.4	68.7	105[(b)]	49
August	87.8	77.8	67.8	102	49
September	81.8	71.1	60.4	103	36
October	71.2	59.9	48.5	91	25
November	60.4	49.7	39.0	84	5
December	49.8	40.8	31.7	80	-6
Annual	69.5	59.1	48.8	105[(b)]	-24[(c)]

(a) 1981-2010
(b) June 2012 and July 2012
(c) January 1985
Notes: Air Temperature (°F) from 2013 Annual Knoxville Local Climatological Data.

Table 2.3-14: Air Temperatures for Oak Ridge, Tennessee

	Normal Daily Maximum	Normal Dry Bulb	Normal Daily Minimum	Extreme Daily Maximum	Extreme Daily Minimum
Period of Record (yrs)	**30[(a)]**	**30[(a)]**	**30[(a)]**	**66**	**66**
January	46.6	37.7	28.9	76	-17[(c)]
February	51.9	41.8	31.7	79	-13
March	61.4	50.4	39.3	86	1
April	70.6	58.8	46.9	92	20
May	78.3	66.8	55.2	95	30
June	85.7	75.1	64.5	105[(b)]	39
July	88.4	78.5	68.6	105[(b)]	49
August	88.0	77.6	67.2	103	50
September	81.7	70.7	59.7	102	33
October	71.1	59.5	48.0	90	21
November	59.6	48.9	38.3	83	0
December	49.6	40.3	31.1	78	-7
Annual	69.4	58.8	48.3	105[(b)]	-17[(c)]

(a) 1981-2010
(b) July 1952 and June 2012
(c) January 1985
Notes: Air Temperature (°F) from 2013 Annual Oak Ridge Local Climatological Data.

Table 2.3-15: Air Temperatures for Tower L

Month	Maximum Daily Average (°F)	Minimum Daily Average (°F)
January	46.3	25.3
February	59.0	36.0
March	58.7	36.8
April	67.4	46.6
May	76.7	64.3
June	79.1	69.0
July	79.0	70.5
August	78.0	71.5
September	77.4	69.6
October	75.6	48.8
November	53.7	34.7
December	53.8	31.3

Table 2.3-16: Humidity Values for Knoxville and Oak Ridge, Tennessee

Knoxville, Tennessee	Mean Dry Bulb Temperature	Mean Dewpoint Temperature	Mean Relative Humidity (%)	Mean Absolute Humidity (g/m^3)
January	39.2	31.1	74	4.71
February	40.7	33.6	70	4.71
March	49.8	39.6	66	6.16
April	58.5	47.6	65	8.20
May	67.4	57.8	73	12.38
June	74.1	65.3	75	15.77
July	78.1	68.7	75	17.87
August	77.1	67.9	76	17.56
September	70.8	61.5	75	14.19
October	60.1	50.9	75	9.98
November	48.3	40.9	74	6.55
December	41.1	33.9	75	5.12
Oak Ridge, Tennessee	**Mean Dry Bulb Temperature**	**Mean Dewpoint Temperature**	**Mean Relative Humidity (%)**	**Mean Absolute Humidity (g/m^3)**
January	36.8	31.8	71	4.11
February	40.1	34.0	65	4.27
March	49.2	40.7	64	5.82
April	58.3	49.8	63	7.89
May	66.2	58.8	71	11.57
June	73.9	65.8	69	14.41
July	77.4	69.7	75	17.49
August	76.7	68.9	73	16.65
September	70.2	62.3	76	14.05
October	58.7	51.8	73	9.31
November	48.1	41.7	68	6.01
December	39.9	34.1	76	4.94

Notes: Temperatures and Dewpoints (°F) from 2013 Annual Knoxville and Oak Ridge Local Climatological Data.

Table 2.3-17: Humidity Values for Tower L

Month	Average Dry Bulb Temperature (F)	Average Dewpoint Temperature (F)	Mean Relative Humidity (%)	Mean Absolute Humidity (g/m^3)
January	35.4	29.7	76	5.10
February	47.5	39.9	79	7.16
March	47.0	34.3	66	5.81
April	56.8	51.5	71	10.36
May	70.4	64.2	77	15.33
June	73.2	68.2	81	17.56
July	76.2	73.0	84	20.28
August	74.9	69.6	81	18.19
September	73.9	68.2	82	17.36
October	60.1	53.4	83	11.24
November	43.1	38.5	80	6.64
December	42.8	40.0	82	7.04
Notes: Data observed at Tower L for 2018-2019.				

Table 2.3-18: Historical Precipitation Data for Oak Ridge, Tennessee

	Normal Monthly	Maximum Monthly	Minimum Monthly	Maximum in 24 hours	Days with Precipitation (^{3} 0.01 inch)	Days with Thunderstorms[a]
Period of Record (yrs)	**30[b]**	**66**	**66**	**66**	**30[b]**	**17**
January	4.54	13.27	0.93	4.25	10.9	0.7
February	4.57	12.78	0.84	5.18	10.1	1.7
March	5.06	12.24	2.13	4.74	11.2	2.5
April	4.18	14.03	0.88	6.24	10.4	4.0
May	4.29	10.70	0.80	4.41	11.9	7.0
June	4.28	11.14	0.53	3.70	10.8	7.6
July	5.27	19.27[c]	1.23	4.91	13.0	10.4
August	2.76	10.46	0.54	7.48[e]	8.9	8.7
September	3.69	10.14	0.41	6.54	8.4	3.3
October	2.92	6.95	Trace[d]	2.66	8.3	1.3
November	4.49	12.22	1.14	5.29	9.3	1.1
December	4.86	12.64	0.67	5.12	11.3	0.8
Annual	50.91	19.27[c]	Trace[d]	7.48[e]	124.5	49.1

(a) From 1998 Annual Oak Ridge Local Climatological Data
(b) 1981-2010
(c) July 1967
(d) October 1963
(e) August 1960
Notes: Precipitation (inches) from 2013 Annual Oak Ridge Local Climatological Data.

Table 2.3-19: Precipitation Data for Towers J and L for 2018-2019

Tower J		
Month	**2018 Monthly Precipitation totals at Tower J (in)**	**2019 Monthly Precipitation totals at Tower J (in)**
January	1.92	7.03
February	11.68	15.43
March	4.82	4.74
April	6.13	4.49
May	3.06	4.23
June	5.84	9.46
July	4.40	5.17
August	3.03	6.81
September	8.59	0.18
October	2.84	8.37
November	5.60	5.86
December	7.42	6.69
Annual Sum	65.33	78.45

Tower L		
Month	**2018 Monthly Precipitation totals at Tower L (in)**	**2019 Monthly Precipitation totals at Tower L (in)**
January	2.20	7.28
February	12.01	16.53
March	5.03	5.50
April	6.22	4.88
May	3.13	4.19
June	5.35	8.59
July	4.90	4.01
August	3.13	5.87
September	7.66	0.16
October	2.98	7.98
November	6.65	6.24
December	7.73	7.05
Annual Sum	66.98	78.28

Table 2.3-20: Point Precipitation (Inches) by Recurrence Interval for Region

	Recurrence Intervals (Years)						
Duration	**1**	**2**	**5**	**10**	**25**	**50**	**100**
5 minutes	0.3	0.4	0.5	0.5	0.6	0.7	0.8
10 minutes	0.5	0.6	0.7	0.8	1.0	1.1	1.2
15 minutes	0.7	0.8	0.9	1.1	1.2	1.4	1.5
30 minutes	0.9	1.1	1.3	1.6	1.8	2.1	2.3
1 hour	1.1	1.4	1.7	2.0	2.5	2.8	3.2
2 hours	1.4	1.6	2.0	2.4	2.9	3.3	3.8
3 hours	1.5	1.8	2.2	2.5	3.1	3.5	4.0
6 hours	1.8	2.2	2.6	3.1	3.7	4.2	4.7
12 hours	2.3	2.7	3.3	3.8	4.5	5.1	5.7
24 hours	2.8	3.3	4.1	4.6	5.5	6.1	6.8
2 days	3.4	4.1	5.0	5.7	6.7	7.5	8.3
4 days	3.9	4.7	5.6	6.4	7.4	8.2	9.0
7 days	4.8	5.7	6.8	7.7	8.8	9.7	10.6
10 days	5.4	6.5	7.7	8.6	9.9	10.9	11.8

Notes:
Data is from NOAA Atlas 14 (Reference 38).
Data is for the Oak Ridge ATDL, Tennessee Station (ID 40-6750).

Table 2.3-21: Historical Snowfall for Knoxville and Oak Ridge, Tennessee

Knoxville, Tennessee	Normal Monthly (inches)	Maximum Monthly (inches)	Maximum in 24 hours (inches)	Maximum Snow Depth (inches)	Normal Number of Days with Snowfall ^{3} 0.01 inch
Period of Record (yrs)	**30(a)**	**69**	**69**	**62**	**30(a)**
January	2.7	15.1	12.0	10	1.0
February	1.6	23.3(b)	17.5	15(d)	0.6
March	0.9	20.2	14.1	15(d)	0.2
April	0.5	10.7	10.7	7	0.1
May - October	0.0	Trace	Trace	0	0.0
November	0.0	18.2	18.2(c)	10	0.0
December	0.8	12.2	8.9	6	0.3
Annual	6.5	23.3(b)	18.2(c)	15(d)	2.2

(a) 1981-2010
(b) February 1960
(c) November 1952
(d) February 1960 and March 1993

Oak Ridge, Tennessee	Normal Monthly	Maximum Monthly	Maximum in 24 hours	Maximum Snow Depth (inches)	Normal Number of Days with Snowfall ^{3} 0.01 inch
Period of Record (yrs)	**30(a)**	**51**	**51**	**62**	**30(a)**
January	4.0	9.6	8.3	8	1.4
February	3.8	17.2	11.3	6	1.3
March	0.8	21.0(b)	12.0(b)	3	0.2
April	0.2	5.9	5.4	3	0.1
May - October	0.0	Trace	Trace	0	0.0
November	0.1	6.5	6.5	1	0.0
December	2.2	14.8	10.8	10(c)	0.6
Annual	11.1	21.0(b)	12.0(b)	10(c)	3.6

(e) 1961-1990
(f) March 1960
(g) December 1963
Notes: Snowfall (inches) from 2013 Annual Knoxville and 1998 Annual Oak Ridge Local Climatological Data.

Table 2.3-22: Fog Occurrence for Knoxville and Oak Ridge, Tennessee

	Number of Days with Heavy Fog (visibility ≤ 1/4 mile)	
	Knoxville, TN	**Oak Ridge, TN**
Period of Record (yrs)	**50**	**14**
January	2.6	2.2
February	1.8	1.4
March	1.6	1.7
April	1.3	2.3
May	2.2	5.4
June	1.7	4.5
July	2.0	5.5
August	3.3	5.3
September	3.7	7.5
October	4.2	7.5
November	2.9	5.0
December	2.4	3.6
Annual	29.7	51.9

Notes: Days with heavy fog from 2013 Annual Oak Ridge and Knoxville Local Climatological Data.

Table 2.3-23: Pasquill Atmospheric Stabilities for the Tower L

Stability Class	Description	Percent Occurrence
A	Extremely Unstable	30.81
B	Moderately Unstable	0.00
C	Slightly Unstable	0.00
D	Neutral	27.04
E	Slightly Stable	26.83
F	Moderately Stable	9.79
G	Extremely stable	5.53
A,B,C	Unstable	30.81
D	Neutral	27.04
E,F,G	Stable	42.15
Notes: Atmospheric stability classes based on 15-30 m temperature difference data for 2018-2019 for Tower L.		

Table 2.3-24: Frequency Distribution of Consecutive Hours of Inversion Conditions (Page 1 of 2)

Number of Consecutive Hours	Stability Class E (-0.5<DT<=1.5)	Stability Class F (1.5<DT<=4.0)	Stability Class G (DT>4.0)	Stability Classes F and G (DT>1.5)	All Inversions (DT>-0.5)
1	1338	841	357	729	863
2	773	385	209	483	653
3	539	204	133	340	546
4	394	112	83	251	469
5	297	67	55	200	415
6	242	39	39	163	374
7	196	25	33	134	350
8	149	17	23	105	335
9	131	7	15	81	322
10	101	5	9	58	307
11	82	5	7	43	288
12	69	4	3	30	274
13	55	3	2	26	251
14	47	1	1	15	217
15	43	0	0	10	166
16	38	0	0	5	127
17	30	0	0	2	98
18	27	0	0	2	69
19	22	0	0	2	55
20	15	0	0	1	41
21	13	0	0	1	36
22	10	0	0	1	30
23	10	0	0	1	27
24	10	0	0	1	24
25	9	0	0	0	23
26	8	0	0	0	23
27	8	0	0	0	23
28	6	0	0	0	21
29	5	0	0	0	21
30	5	0	0	0	21
31	4	0	0	0	21
32	3	0	0	0	19
33	3	0	0	0	19
34	2	0	0	0	18
35	2	0	0	0	18
36	2	0	0	0	18
37	2	0	0	0	18
38	2	0	0	0	18
39	1	0	0	0	18
40	1	0	0	0	18
41	1	0	0	0	17
42	1	0	0	0	16
43	1	0	0	0	15

Table 2.3-24: Frequency Distribution of Consecutive Hours of Inversion Conditions (Page 2 of 2)

Number of Consecutive Hours	Stability Class E (-0.5<DT<=1.5)	Stability Class F (1.5<DT<=4.0)	Stability Class G (DT>4.0)	Stability Classes F and G (DT>1.5)	All Inversions (DT>-0.5)
44	1	0	0	0	15
45	1	0	0	0	13
46	1	0	0	0	11
47	1	0	0	0	11
48-57	0	0	0	0	11
58-61	0	0	0	0	10
62-67	0	0	0	0	9
68-69	0	0	0	0	8
70-71	0	0	0	0	7
72-93	0	0	0	0	6
94-96	0	0	0	0	5
97-118	0	0	0	0	4
119-138	0	0	0	0	3
139-165	0	0	0	0	2
166-215	0	0	0	0	1

Notes:
Values in each column are cumulative. For example, values in row 2 include values from row 3, row 3 includes row 4, etc.
ΔT is the 15-30 m temperature difference.
This table shows the number of cases when an inversion condition persisted for two or more hours, and the number of hours the condition lasted.
Data period: January 1, 2018- December 31, 2019.

Table 2.3-25: Classification of Atmospheric Stability

Stability Classification	Pasquill Stability Category	Ambient Temperature Change with Height (°C/100m)
Extremely unstable	A	$\Delta T \leq -1.9$
Moderately unstable	B	$-1.9 < \Delta T \leq -1.7$
Slightly unstable	C	$-1.7 < \Delta T \leq -1.5$
Neutral	D	$-1.5 < \Delta T \leq -0.5$
Slightly stable	E	$-0.5 < \Delta T \leq 1.5$
Moderately stable	F	$1.5 < \Delta T \leq 4.0$
Extremely stable	G	$\Delta T > 4.0$

Figure 2.3-1: Regional Topography

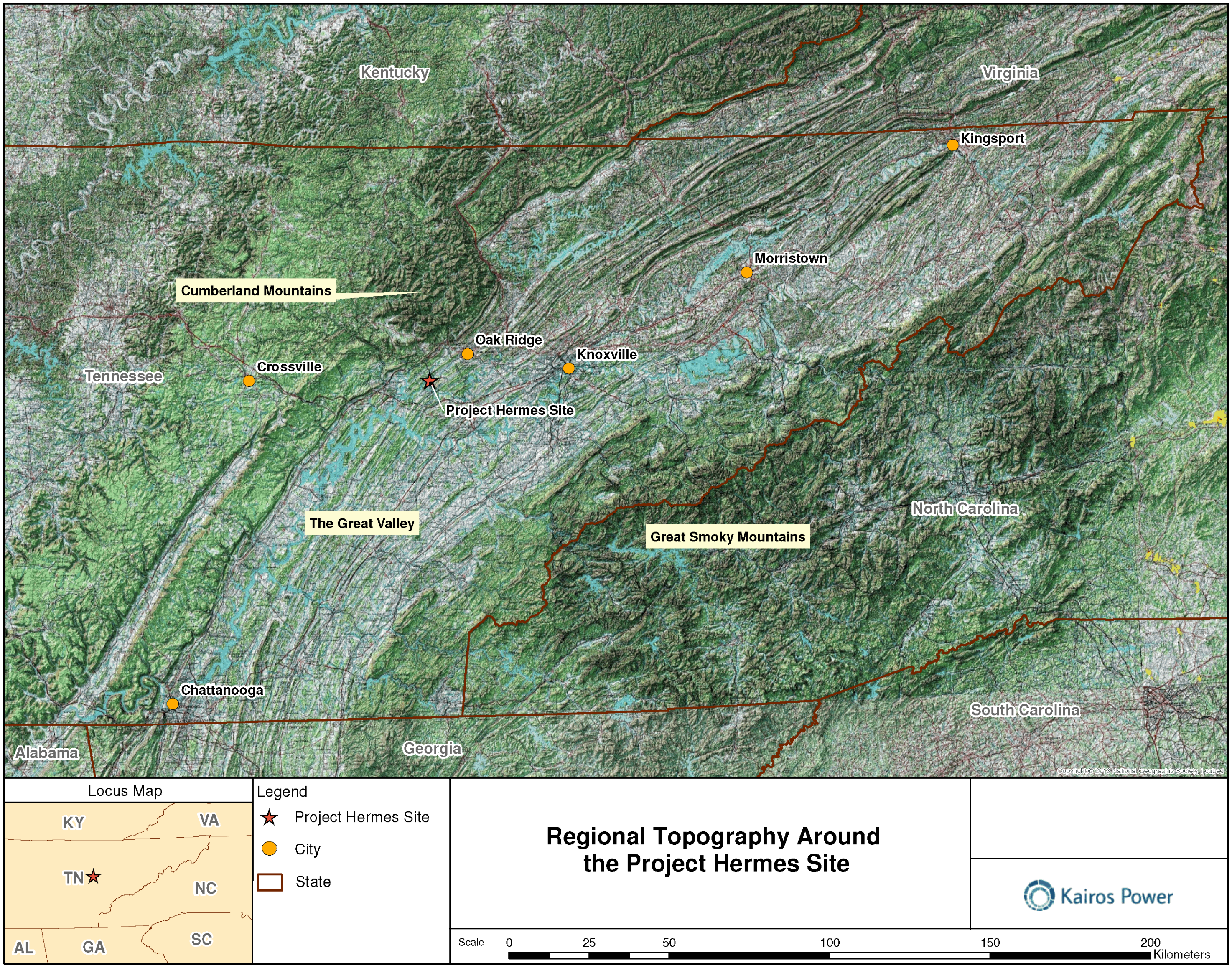


Figure 2.3-2: Local Topography and Locations of the Meteorological Towers

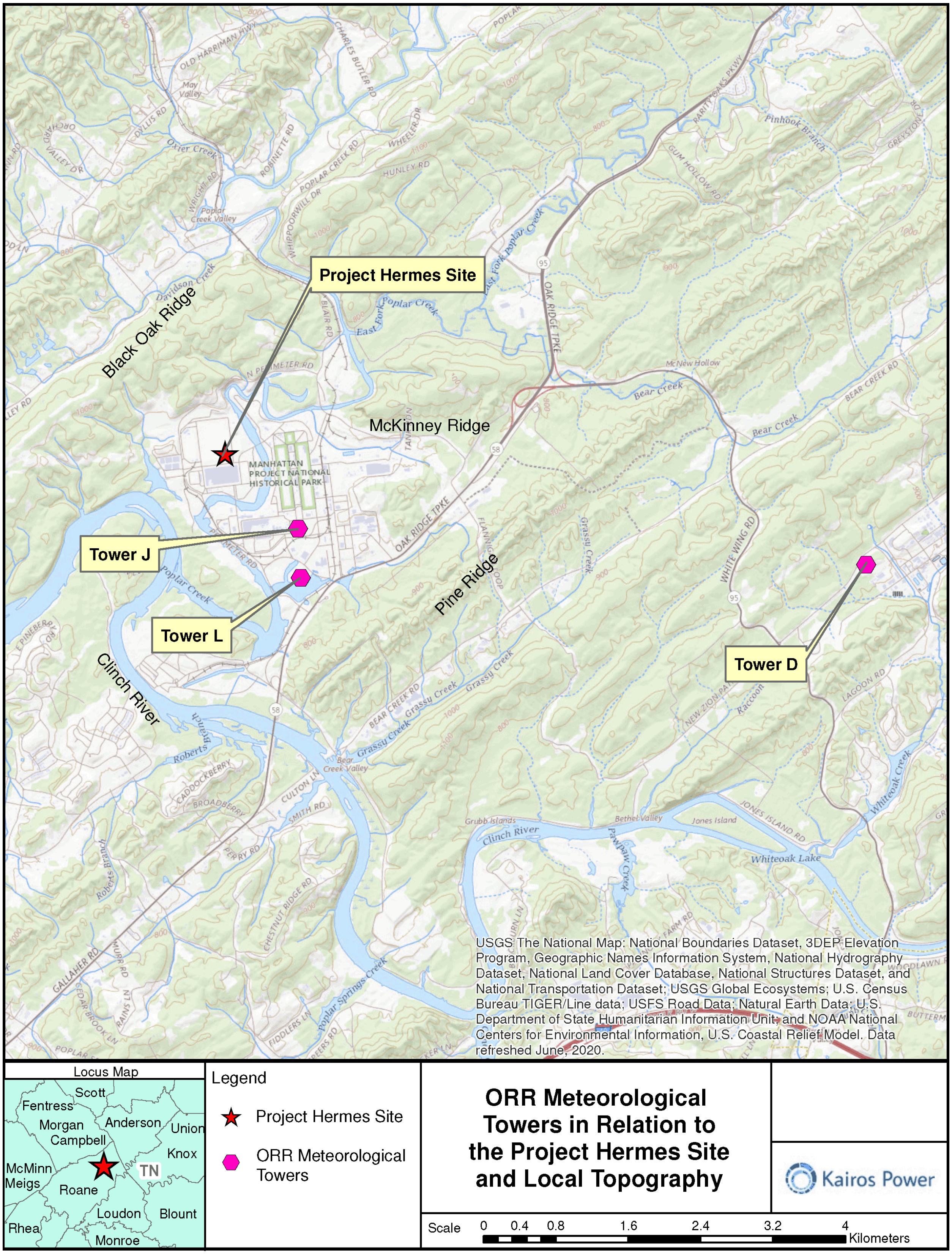


Figure 2.3-3: Topography and Locations of Meteorological Towers Within 100 km of the Site

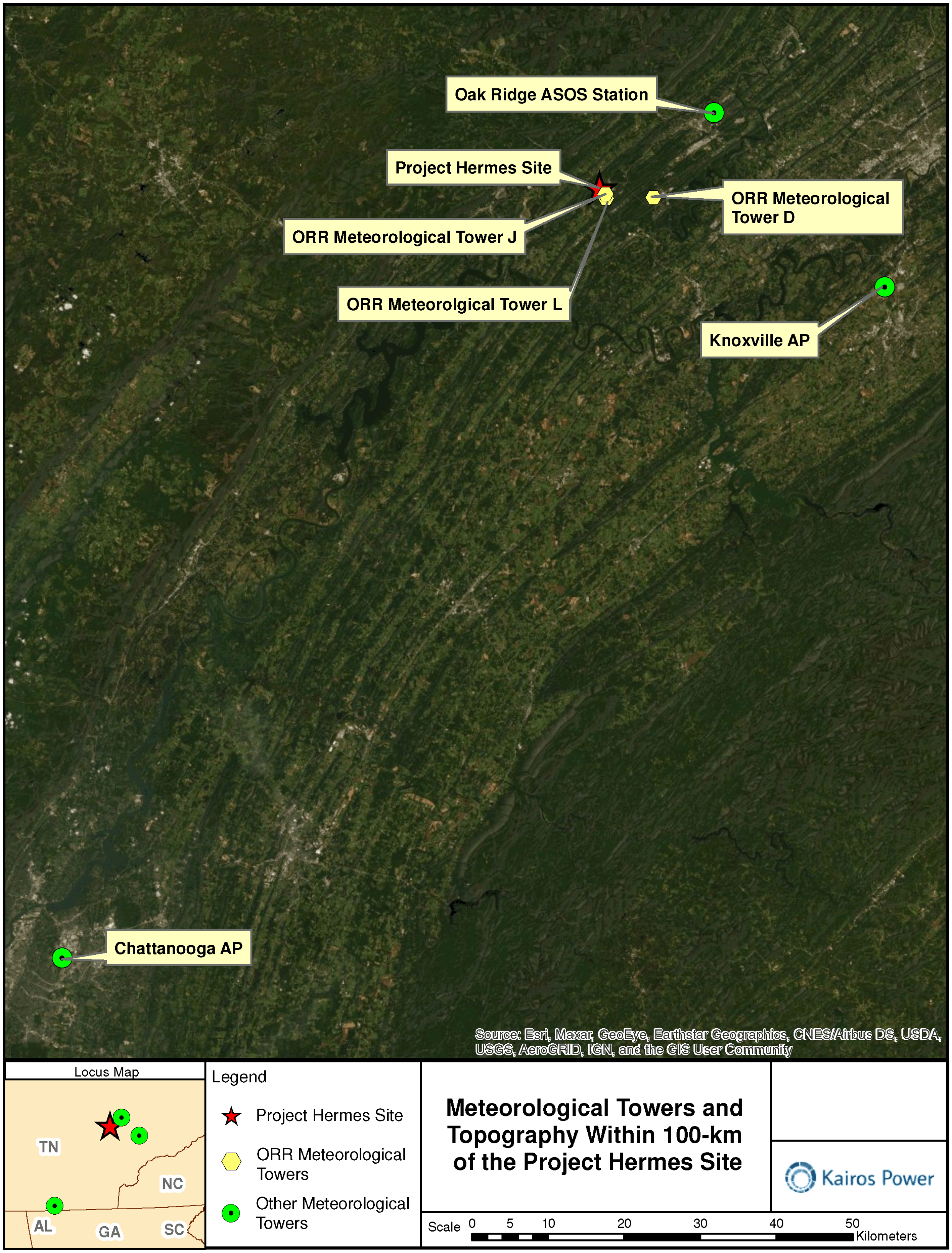
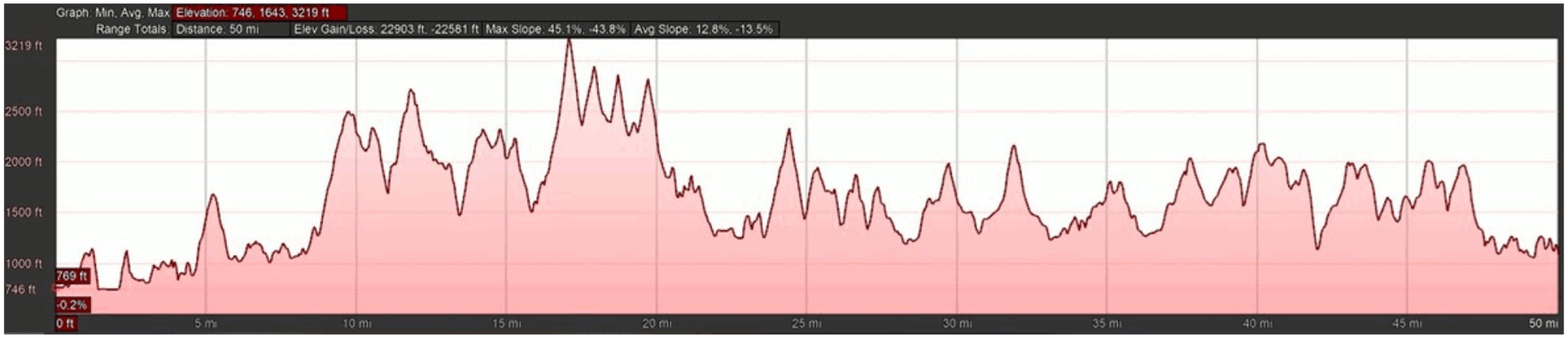
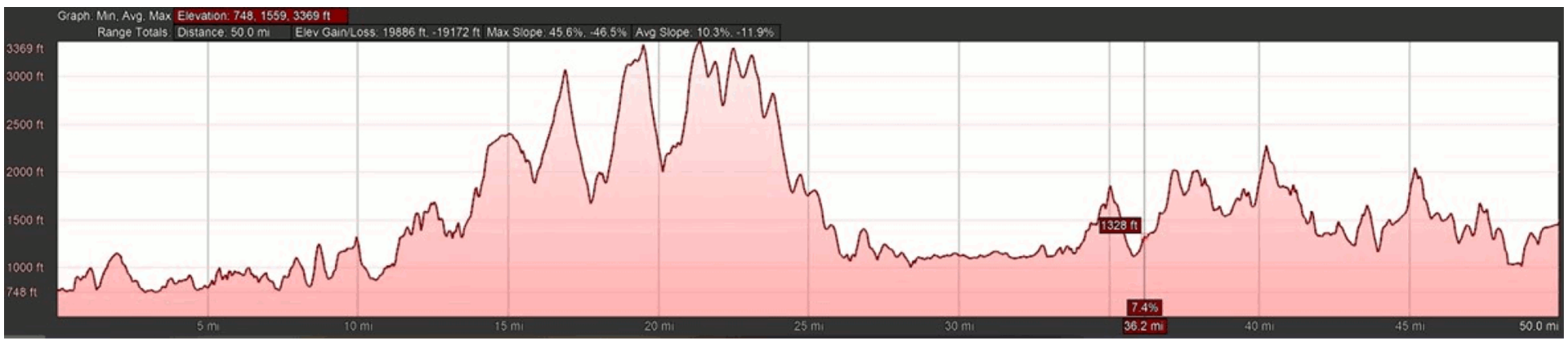


Figure 2.3-4: Terrain Elevations Within 50 miles North and North-Northeast of the Site



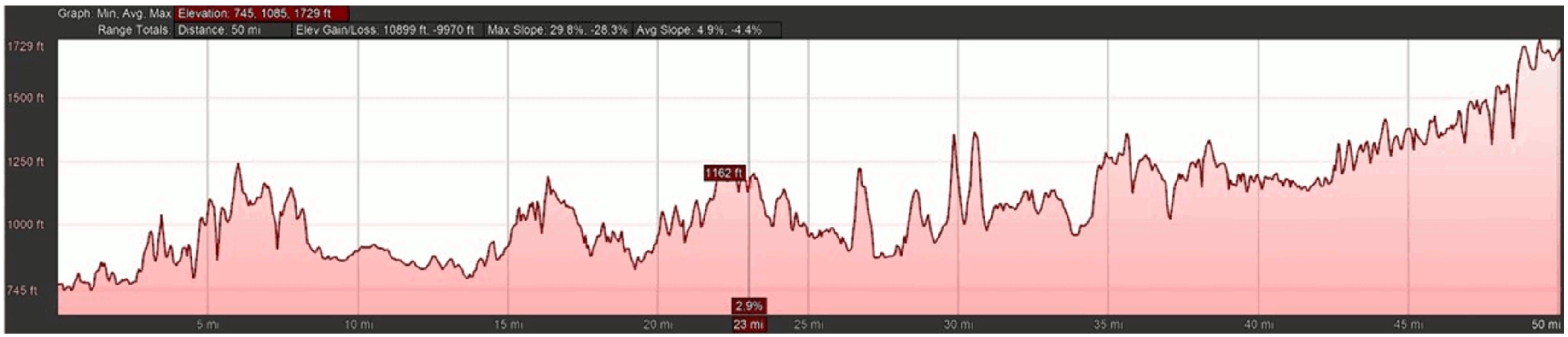
North



North-Northeast

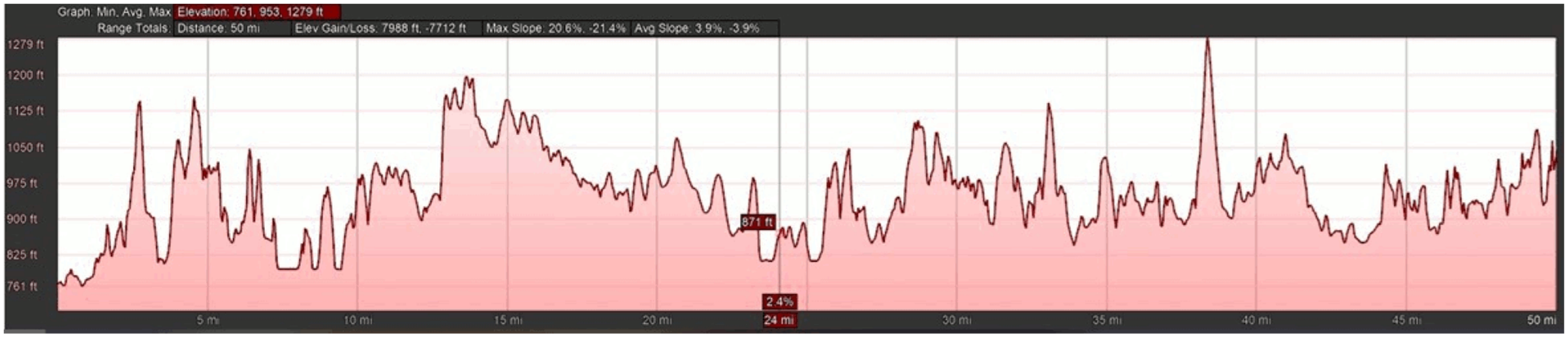
Figure 2.3-5: Terrain Elevations Within 50 miles Northeast and East-Northeast of the Site

Northeast



East-Northeast

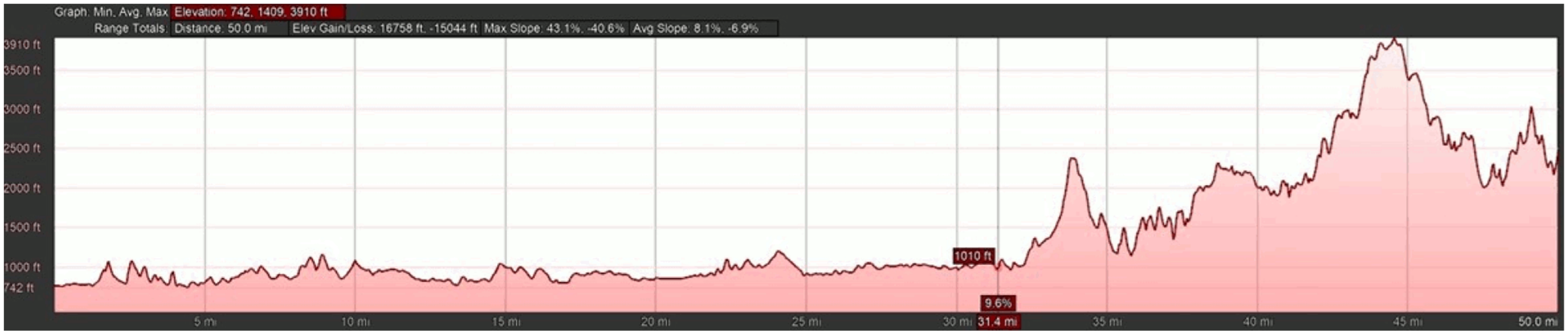
Figure 2.3-6: Terrain Elevations Within 50 miles East and East-Southeast of the Site



East

East-Southeast

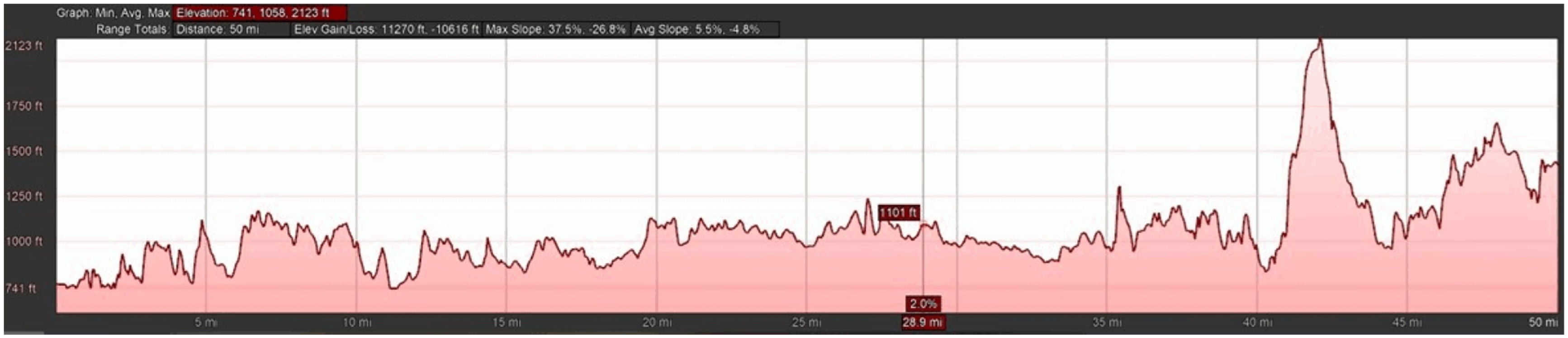
Figure 2.3-7: Terrain Elevations Within 50 miles Southeast and South-Southeast of the Site



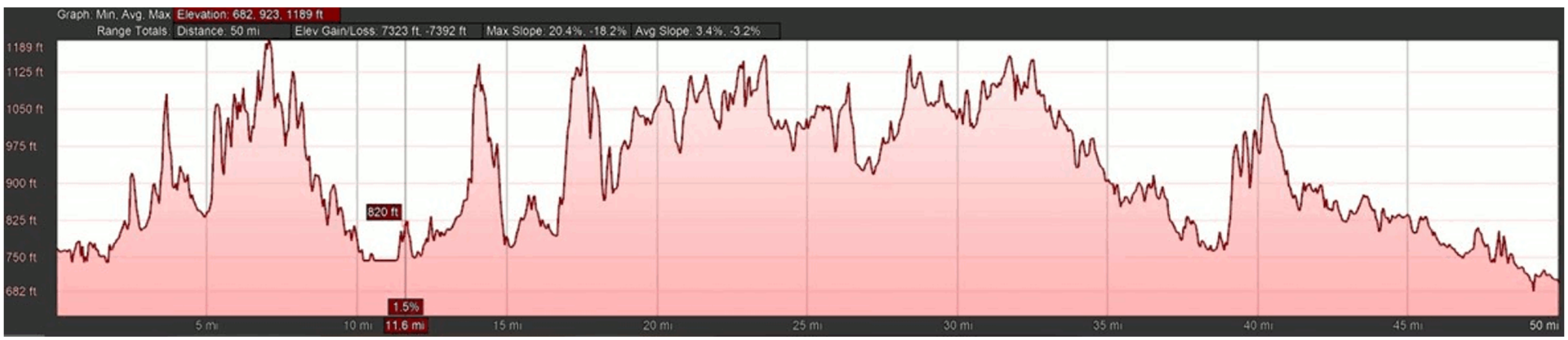
Southeast

South-Southeast

Figure 2.3-8: Terrain Elevations Within 50 miles South and South-Southwest of the Site



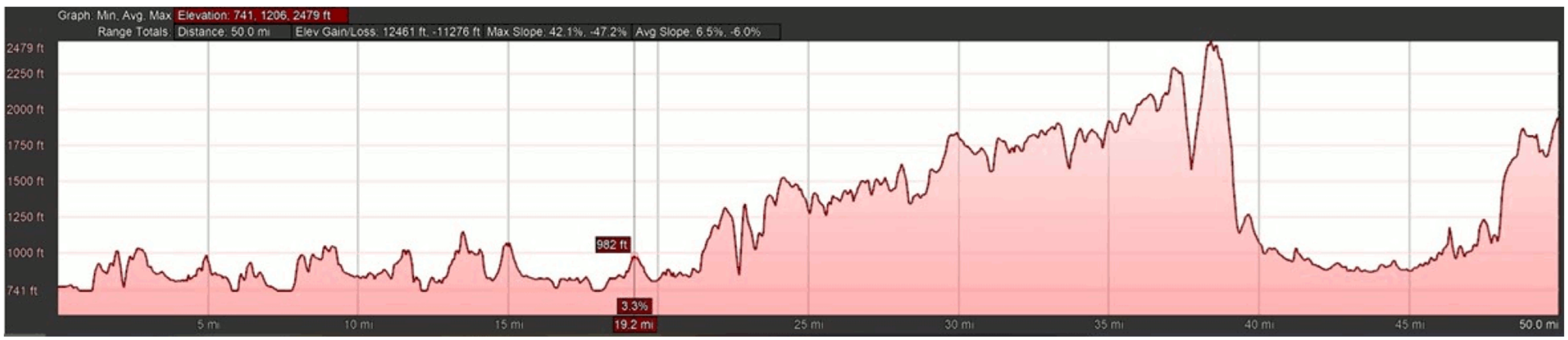
South



South-Southwest

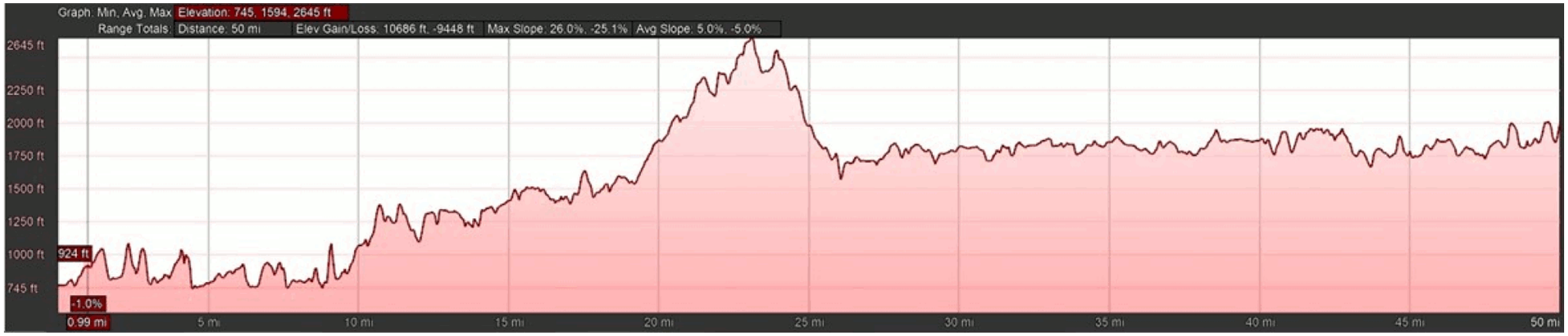
Figure 2.3-9: Terrain Elevations Within 50 miles Southwest and West-Southwest of the Site

Southwest

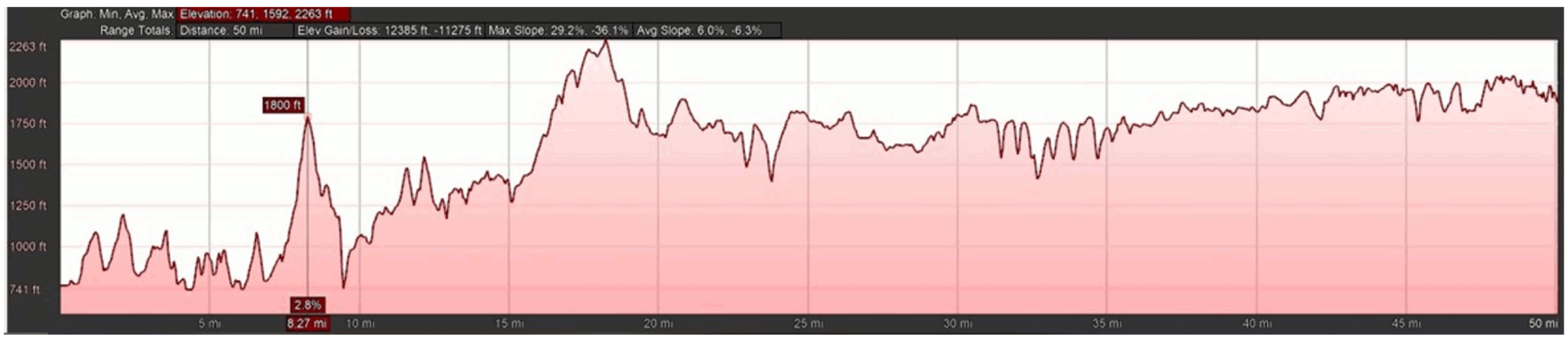


West-Southwest

Figure 2.3-10: Terrain Elevations Within 50 miles West and West-Northwest of the Site

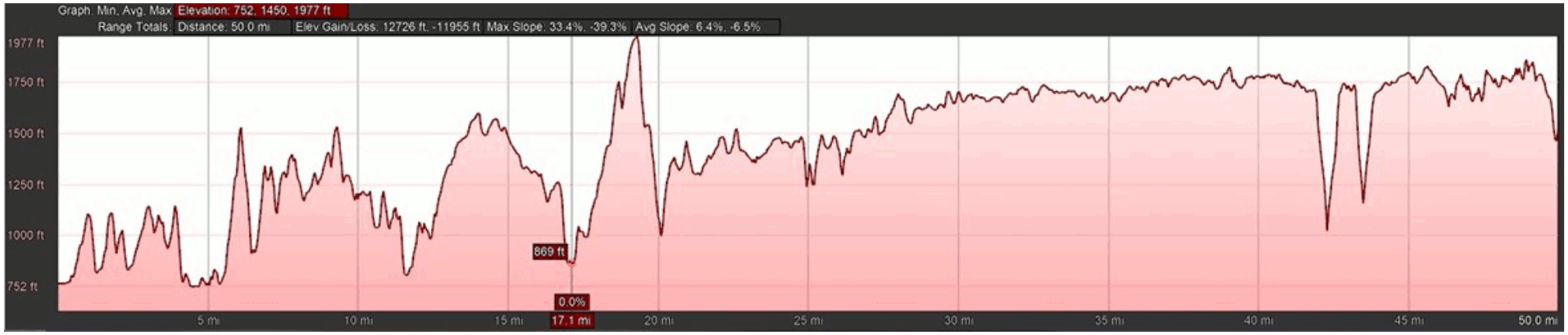


West

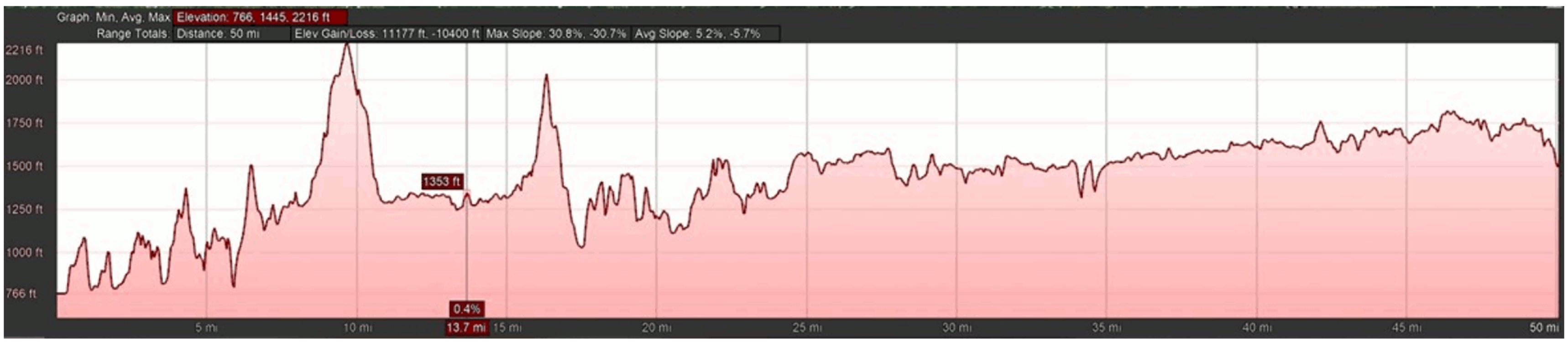


West-Northwest

Figure 2.3-11: Terrain Elevations Within 50 miles Northwest and North-Northwest of the Site



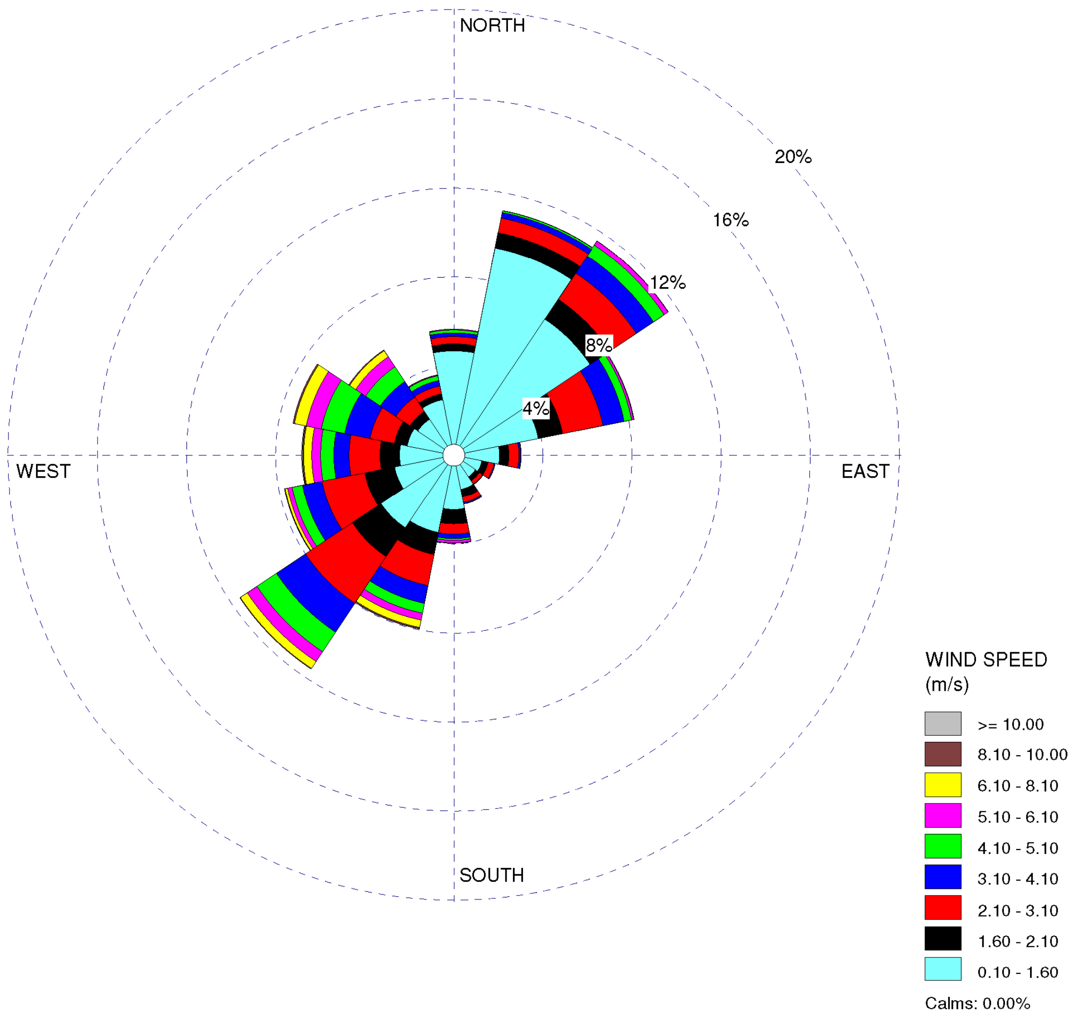
Northwest



North-Northwest

Figure 2.3-12: Tower J 20 Meter Wind Rose

WIND ROSE PLOT:	DISPLAY:
2018-2019 Wind Rose for Tower J 20 m Level	**Wind Speed** **Direction (blowing from)**



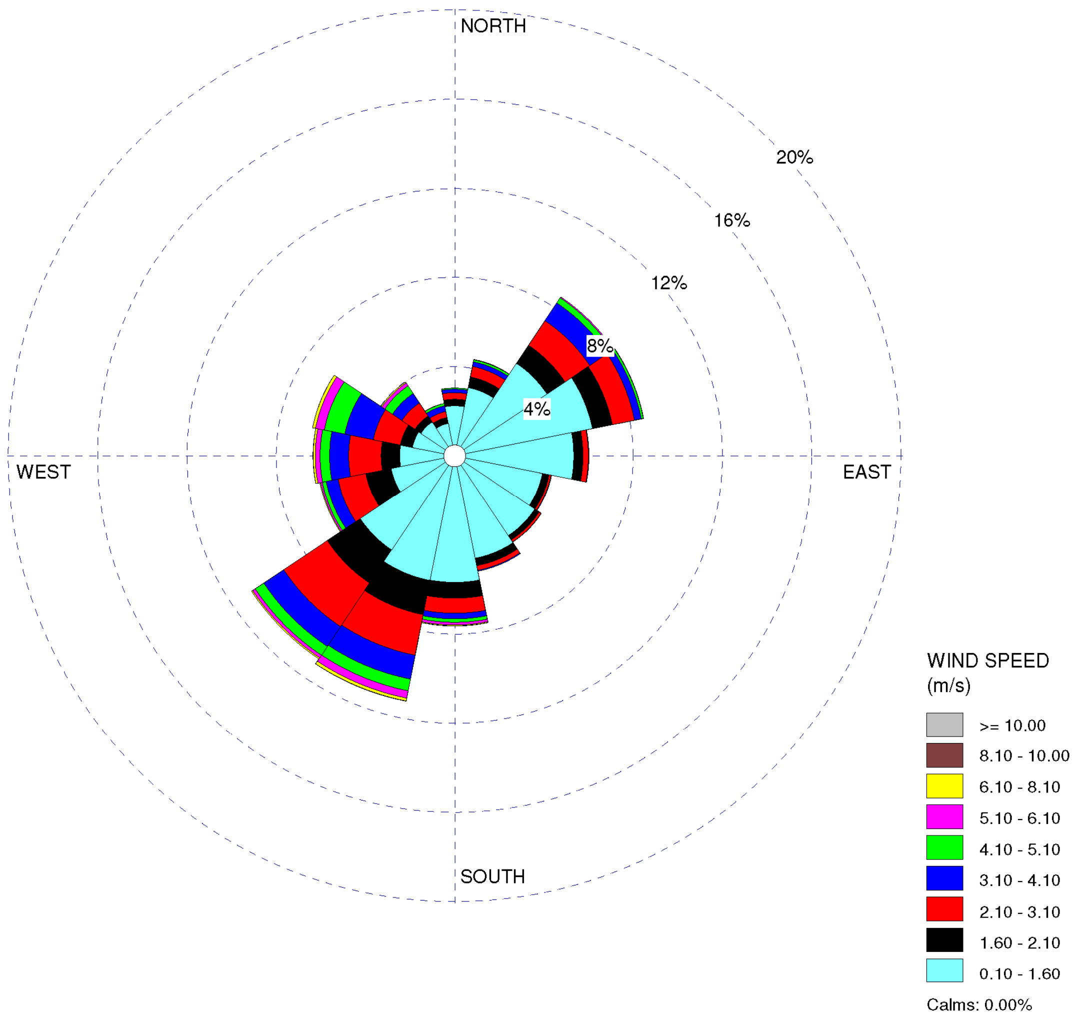
COMMENTS:	DATA PERIOD:	COMPANY NAME:	
	Start Date: 1/1/2018 - 00:00 **End Date: 12/31/2019 - 23:00**	MODELER:	
	CALM WINDS: **0.00%**	TOTAL COUNT: **17520 hrs.**	
	AVG. WIND SPEED: **1.97 m/s**	DATE: **6/23/2021**	PROJECT NO.:

WRPLOT View - Lakes Environmental Software

Figure 2.3-13: Tower L 15 Meter Wind Rose

WIND ROSE PLOT:
2018-2019 Wind Rose for Tower L 15 m Level

DISPLAY:
Wind Speed
Direction (blowing from)



COMMENTS:

DATA PERIOD:
Start Date: 1/1/2018 - 00:00
End Date: 12/31/2019 - 23:00

COMPANY NAME:

MODELER:

CALM WINDS:
0.00%

TOTAL COUNT:
17520 hrs.

AVG. WIND SPEED:
1.54 m/s

DATE:
6/23/2021

PROJECT NO.:

WRPLOT View - Lakes Environmental Software

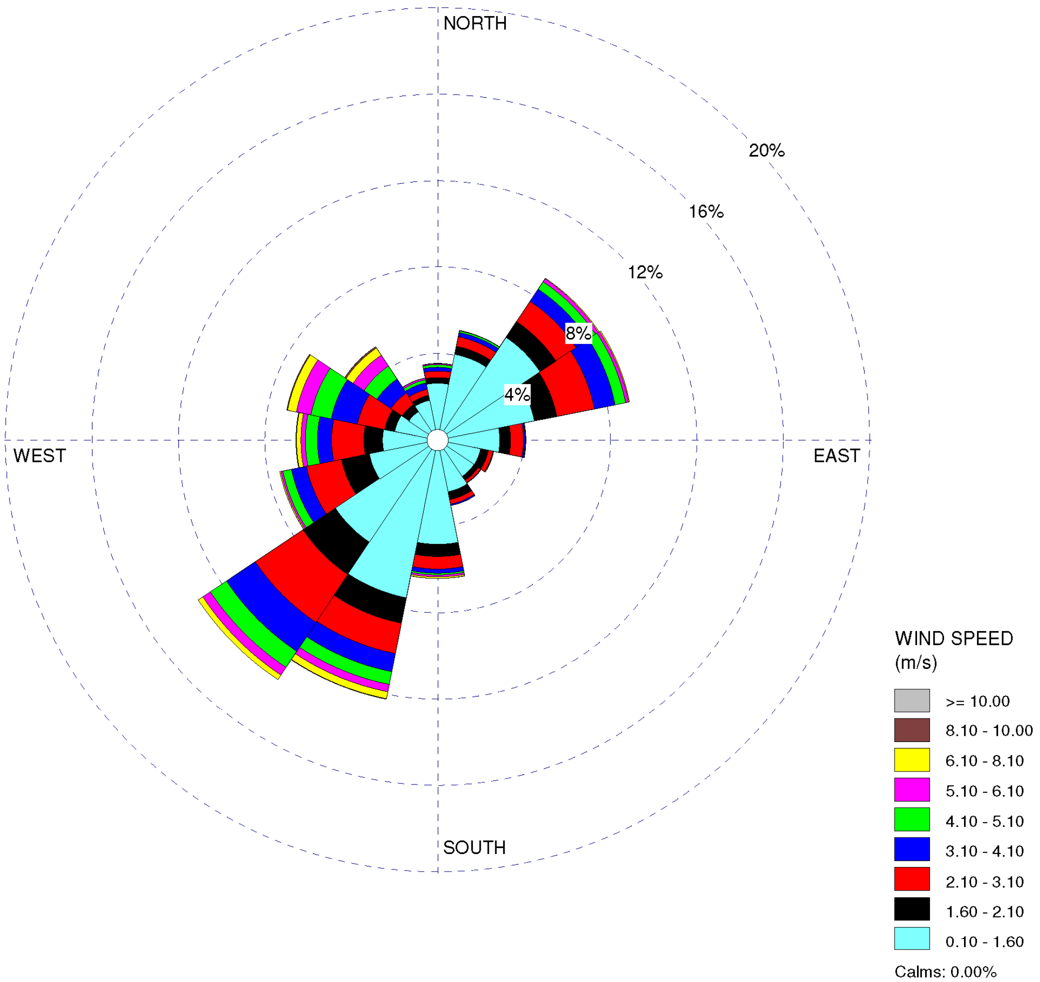
Figure 2.3-14: Tower L 30 Meter Wind Rose

WIND ROSE PLOT:

2018-2019 Wind Rose for Tower L 30 m Level

DISPLAY:

Wind Speed
Direction (blowing from)



COMMENTS:

DATA PERIOD:

Start Date: 1/1/2018 - 00:00
End Date: 12/31/2019 - 23:00

COMPANY NAME:

MODELER:

CALM WINDS:

0.00%

TOTAL COUNT:

17520 hrs.

AVG. WIND SPEED:

1.83 m/s

DATE:

6/23/2021

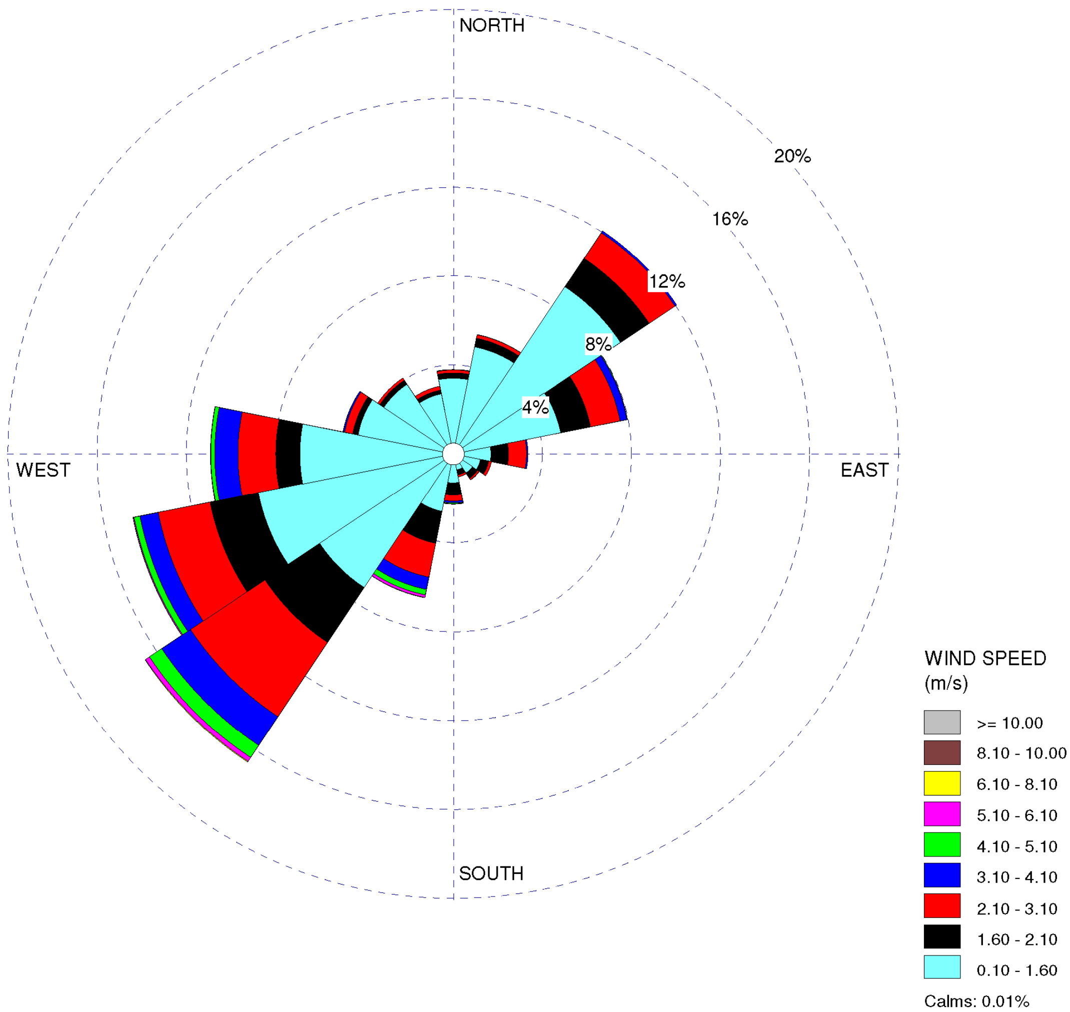
PROJECT NO.:

WRPLOT View - Lakes Environmental Software

Figure 2.3-15: Tower D 15 Meter Wind Rose

WIND ROSE PLOT:
2018-2019 Wind Rose for Tower D 15 m Level

DISPLAY:
Wind Speed
Direction (blowing from)



COMMENTS:

DATA PERIOD:
Start Date: 1/1/2018 - 00:00
End Date: 12/31/2019 - 23:00

COMPANY NAME:

MODELER:

CALM WINDS:
0.01%

TOTAL COUNT:
17520 hrs.

AVG. WIND SPEED:
1.39 m/s

DATE:
6/23/2021

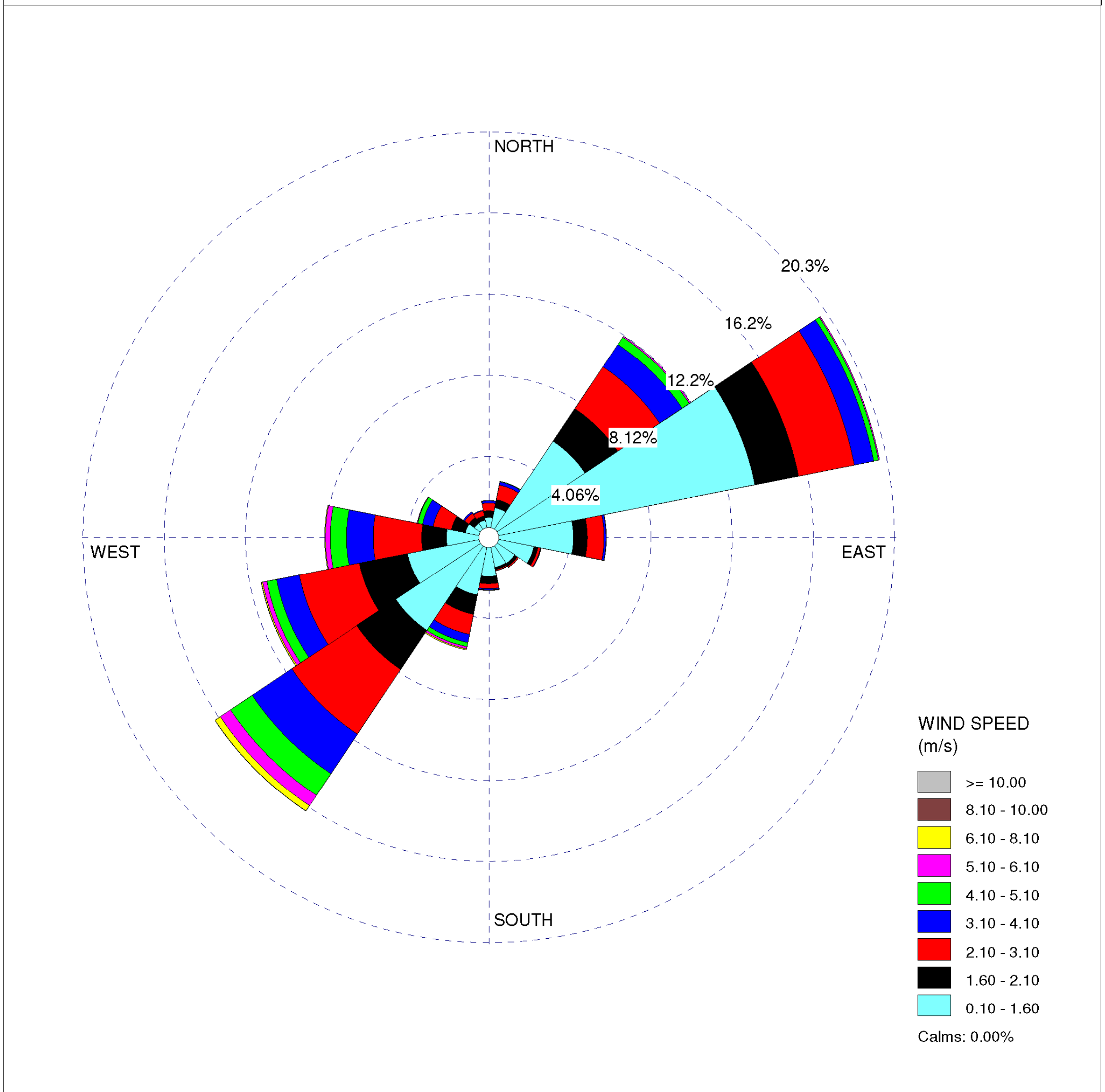
PROJECT NO.:

WRPLOT View - Lakes Environmental Software

Figure 2.3-16: Tower D 35 Meter Wind Rose

WIND ROSE PLOT:
2018-2019 Wind Rose for Tower D 35 m Level

DISPLAY:
Wind Speed
Direction (blowing from)



COMMENTS:

DATA PERIOD:
Start Date: 1/1/2018 - 00:00
End Date: 12/31/2019 - 23:00

COMPANY NAME:

MODELER:

CALM WINDS:
0.00%

TOTAL COUNT:
17520 hrs.

AVG. WIND SPEED:
1.76 m/s

DATE:
6/23/2021

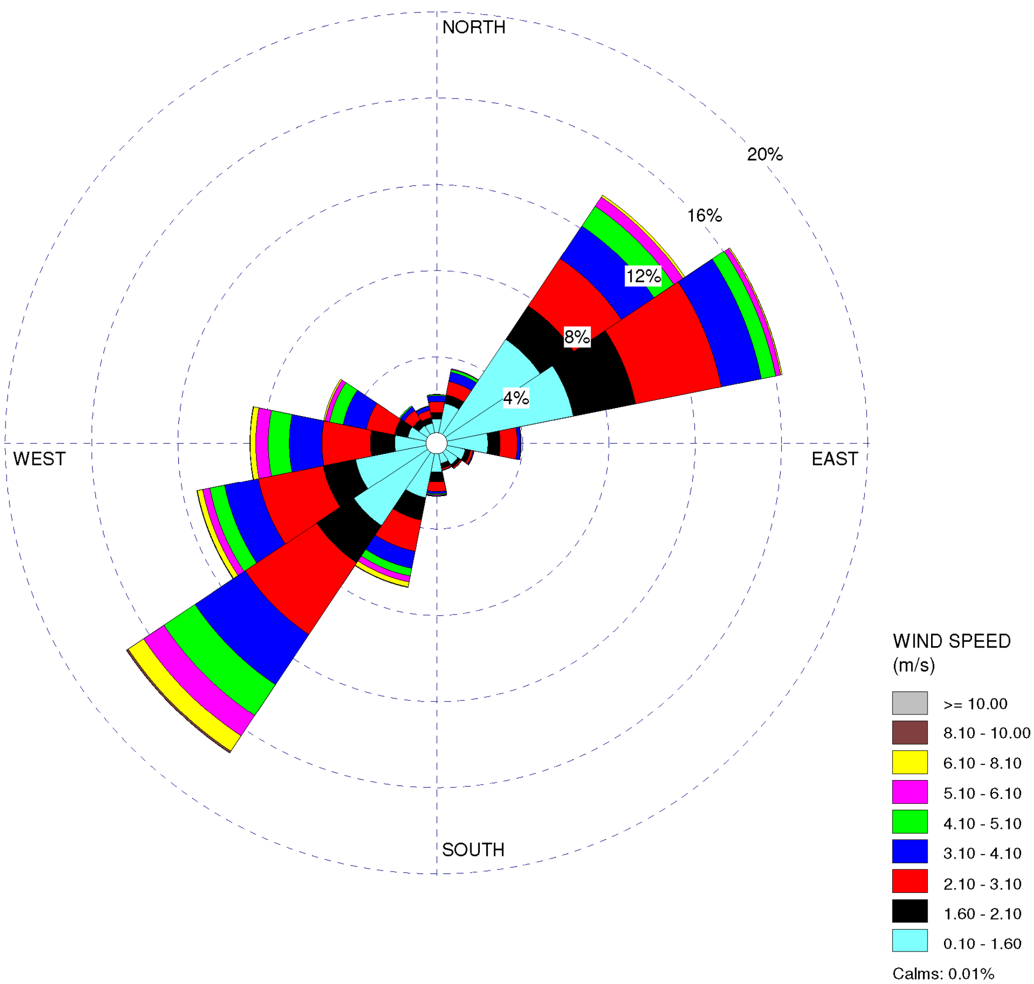
PROJECT NO.:

WRPLOT View - Lakes Environmental Software

Figure 2.3-17: Tower D 60 Meter Wind Rose

WIND ROSE PLOT:
2018-2019 Wind Rose for Tower D 60 m Level

DISPLAY:
Wind Speed
Direction (blowing from)



COMMENTS:

DATA PERIOD:
Start Date: 1/1/2018 - 00:00
End Date: 12/31/2019 - 23:00

COMPANY NAME:

MODELER:

CALM WINDS:
0.01%

TOTAL COUNT:
17519 hrs.

AVG. WIND SPEED:
2.23 m/s

DATE:
6/23/2021

PROJECT NO.:

WRPLOT View - Lakes Environmental Software

Figure 2.3-18: Chattanooga, Tennessee, 10-Year (2000-2009) Wind Rose

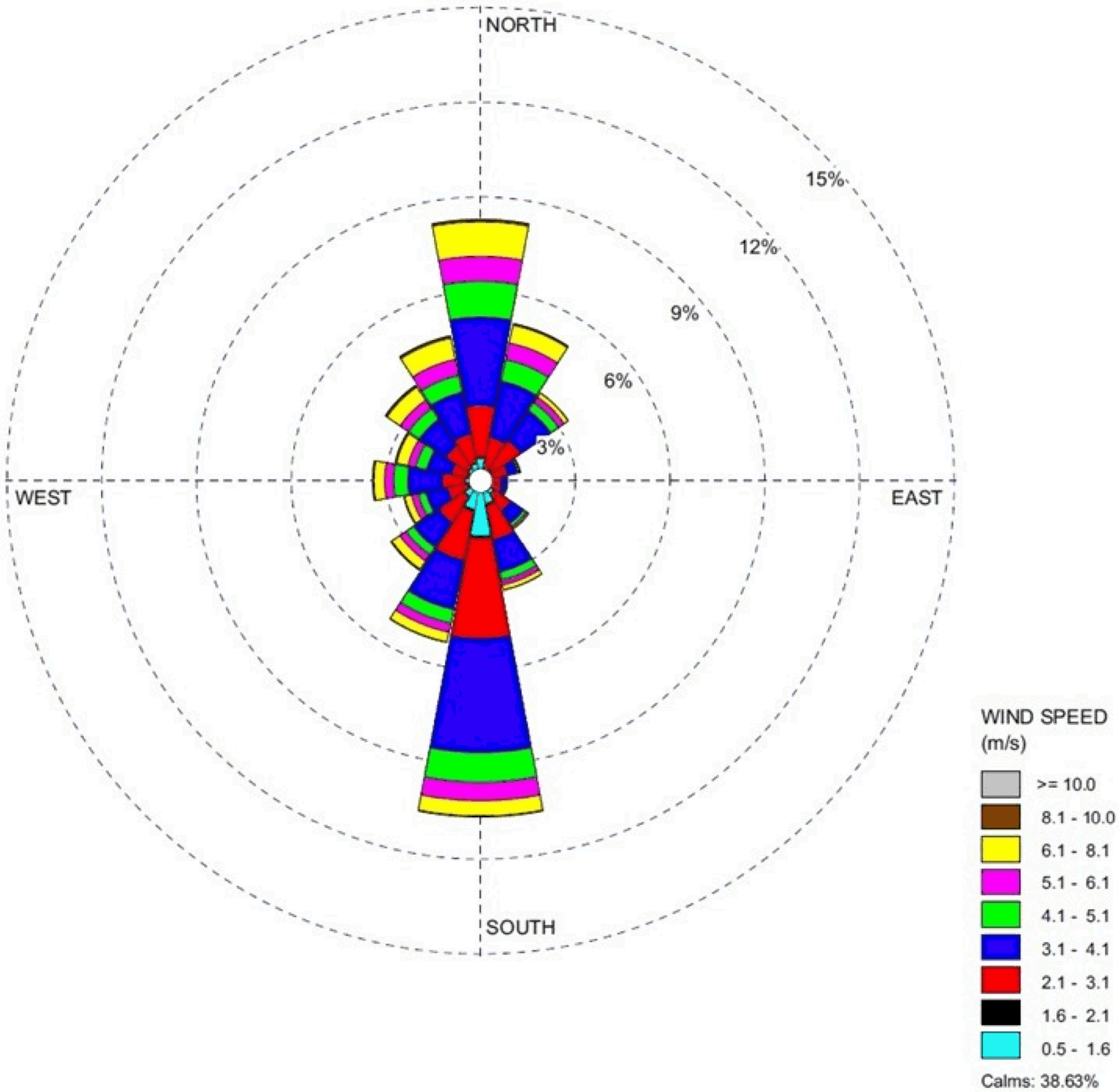


Figure 2.3-19: Oak Ridge, Tennessee, 10-Year (2000-2009) Wind Rose

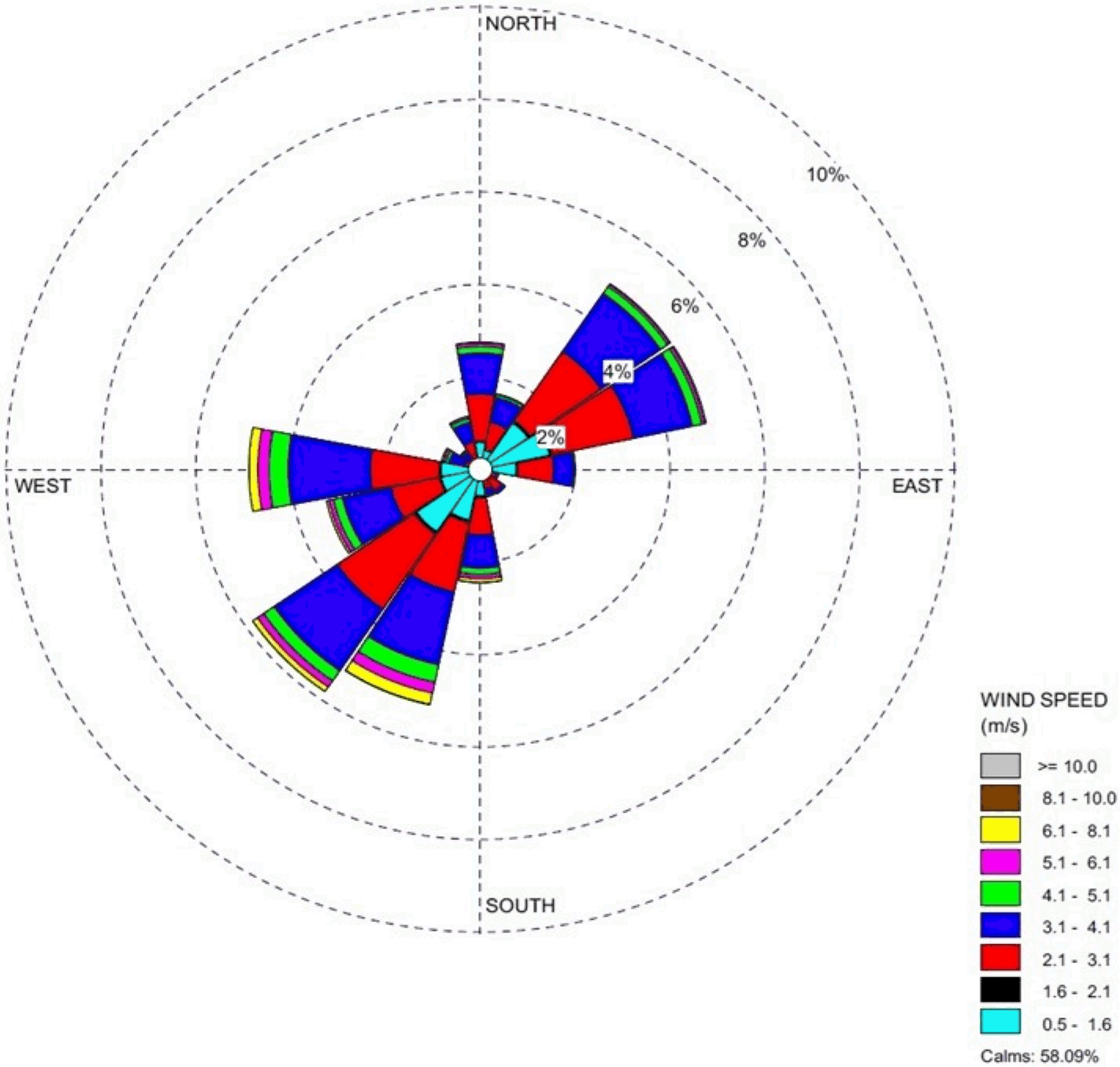


Figure 2.3-20: Wind Direction by Quarter for Tower L at 15 Meters

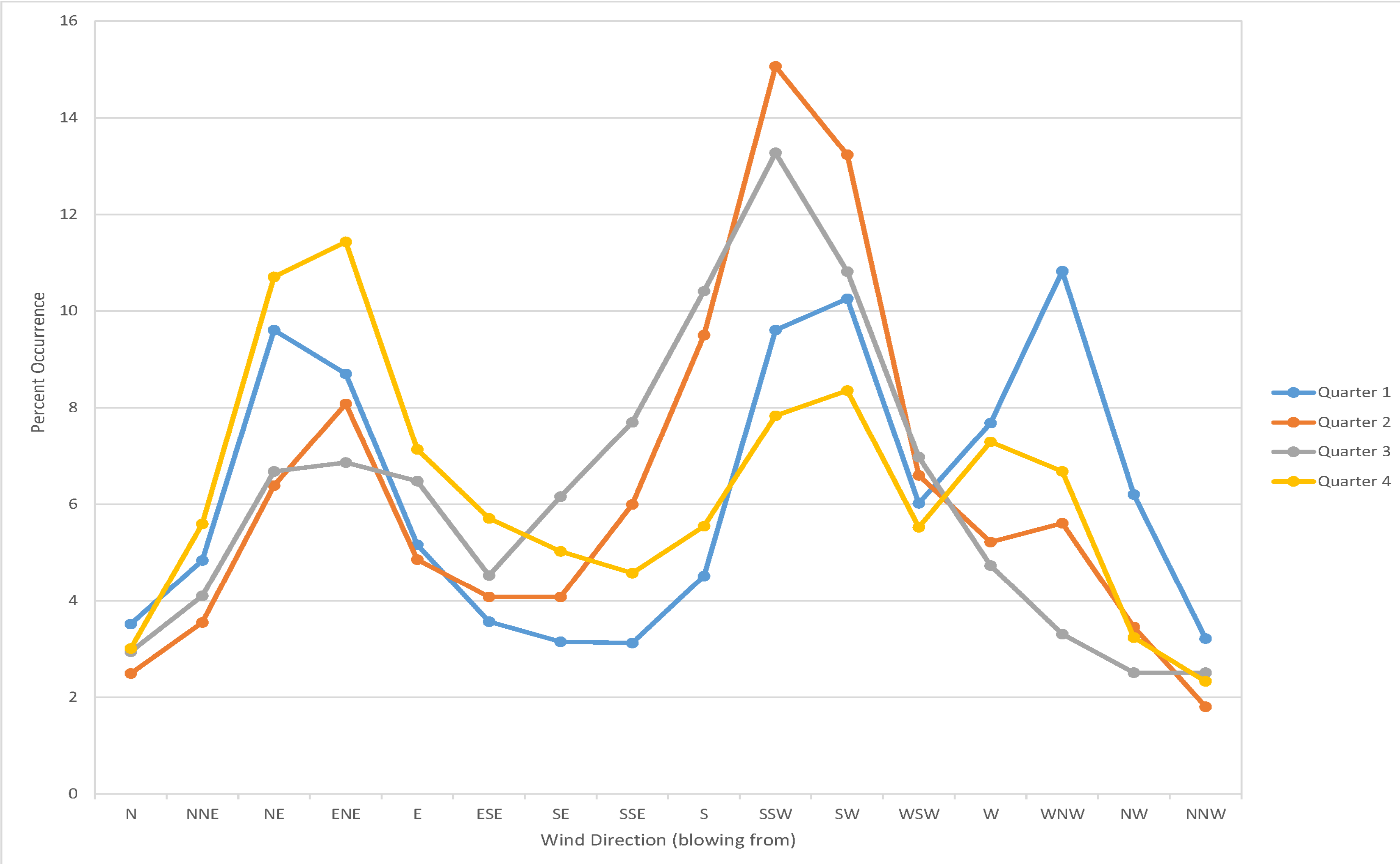
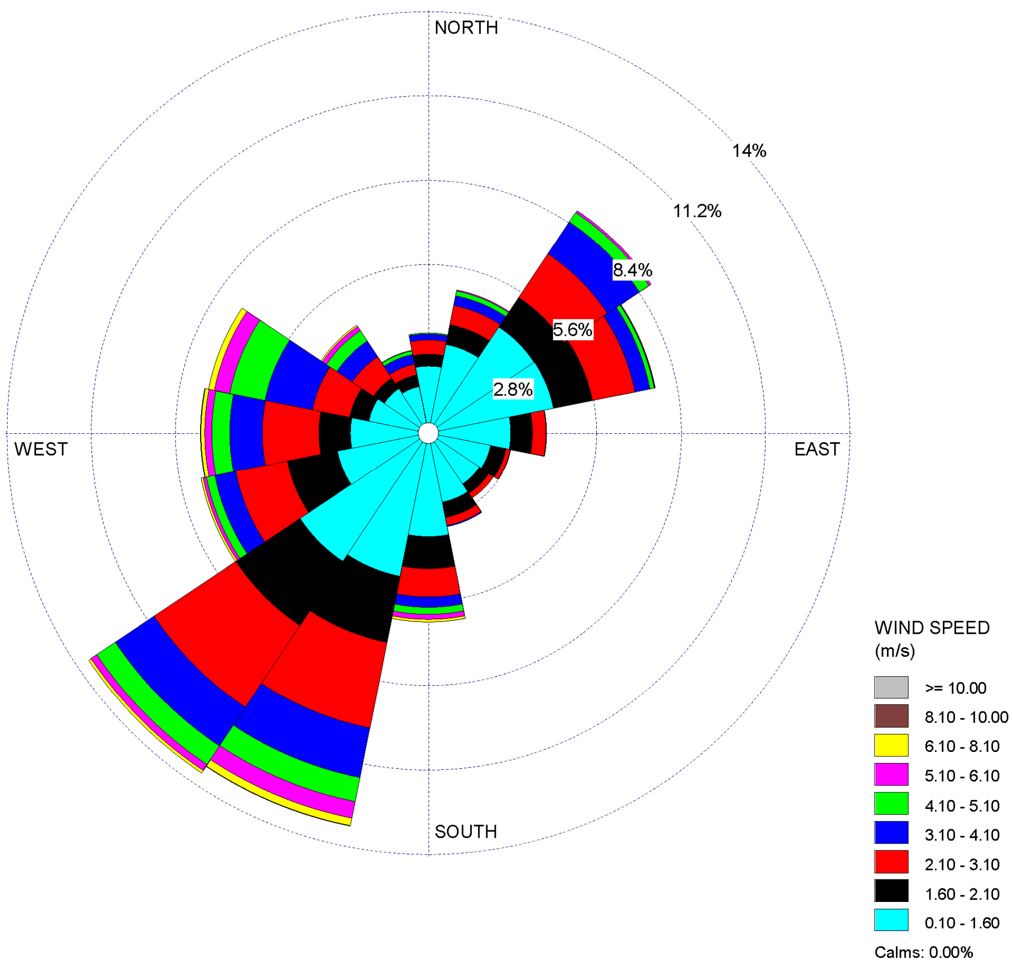


Figure 2.3-21: Daytime Wind Rose for Tower L at 15 Meters

WIND ROSE PLOT:	DISPLAY:
2018-2019 2-Year Wind Rose for Onsite Meteorological Tower at the Project Hermes Site - Daytime Hours	**Wind Speed Direction (blowing from)**

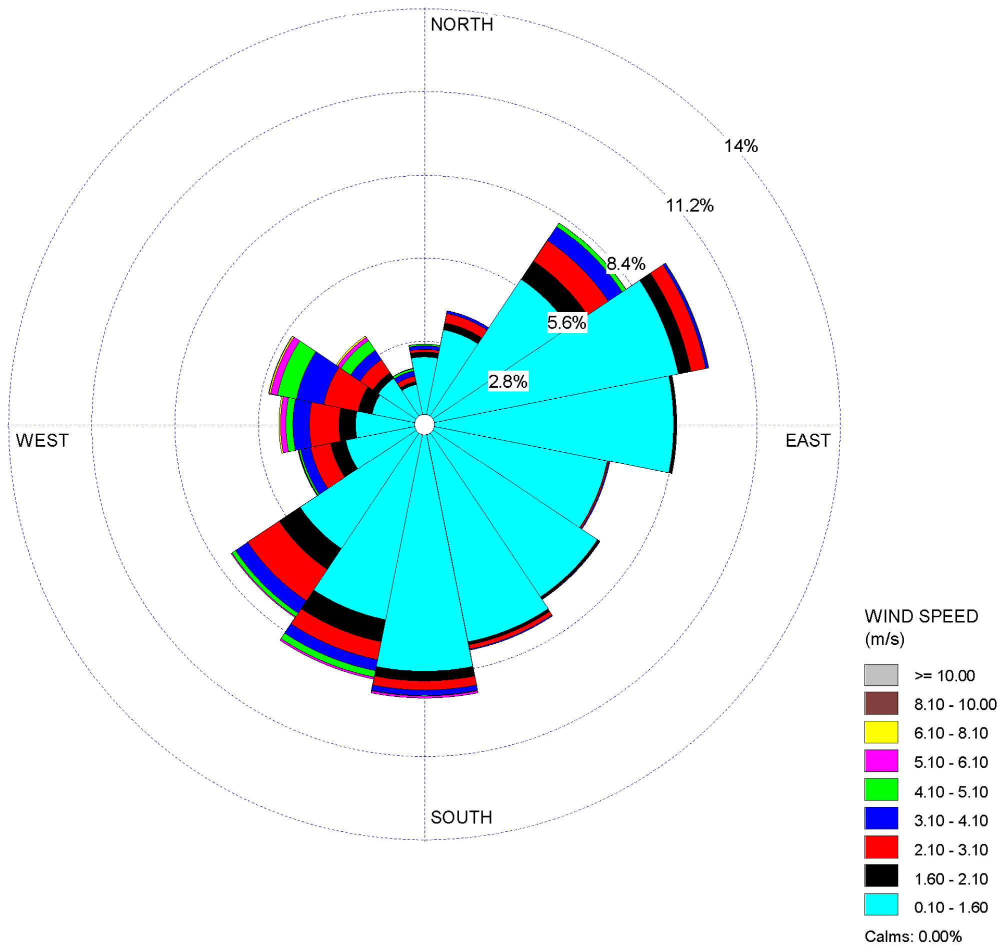


COMMENTS:	DATA PERIOD:	COMPANY NAME:	
	Start Date: 1/1/2018 - 07:00 **End Date: 12/31/2019 - 19:00**	**AECOM**	
		MODELER:	
	CALM WINDS: **0.00%**	TOTAL COUNT: **9490 hrs.**	
	AVG. WIND SPEED: **1.93 m/s**	DATE: **7/16/2021**	PROJECT NO.:

WRPLOT View - Lakes Environmental Software

Figure 2.3-22: Nighttime Wind Rose for Tower L at 15 Meters

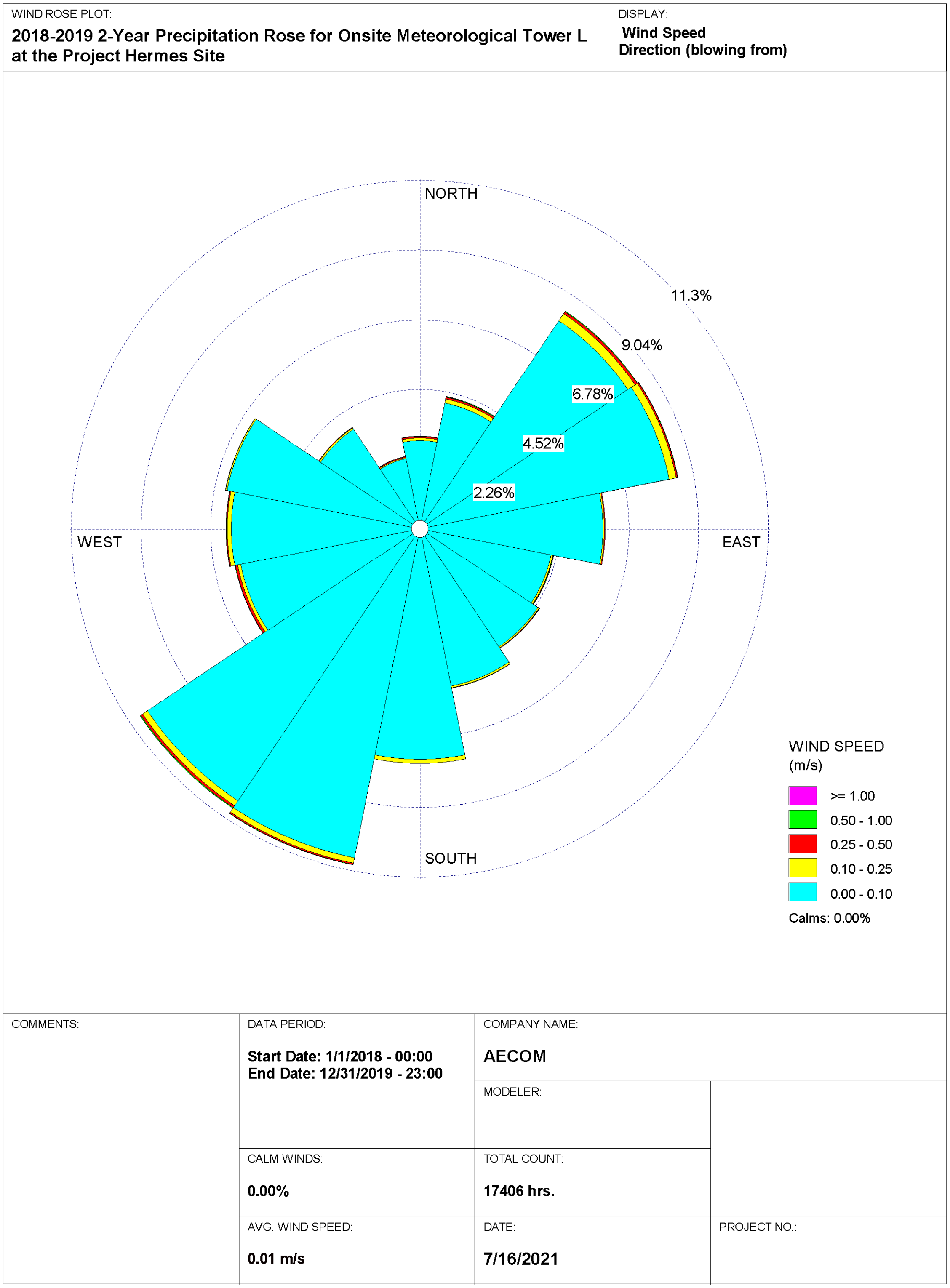
WIND ROSE PLOT:	DISPLAY:
2018-2019 2-Year Wind Rose for Onsite Meteorological Tower at the Project Hermes Site - Nighttime Hours	**Wind Speed Direction (blowing from)**



COMMENTS:	DATA PERIOD:	COMPANY NAME:	
	Start Date: 1/1/2018 - 00:00 **End Date: 12/31/2019 - 23:00**	**AECOM**	
		MODELER:	
	CALM WINDS: **0.00%**	TOTAL COUNT: **8030 hrs.**	
	AVG. WIND SPEED: **1.09 m/s**	DATE: **7/16/2021**	PROJECT NO.:

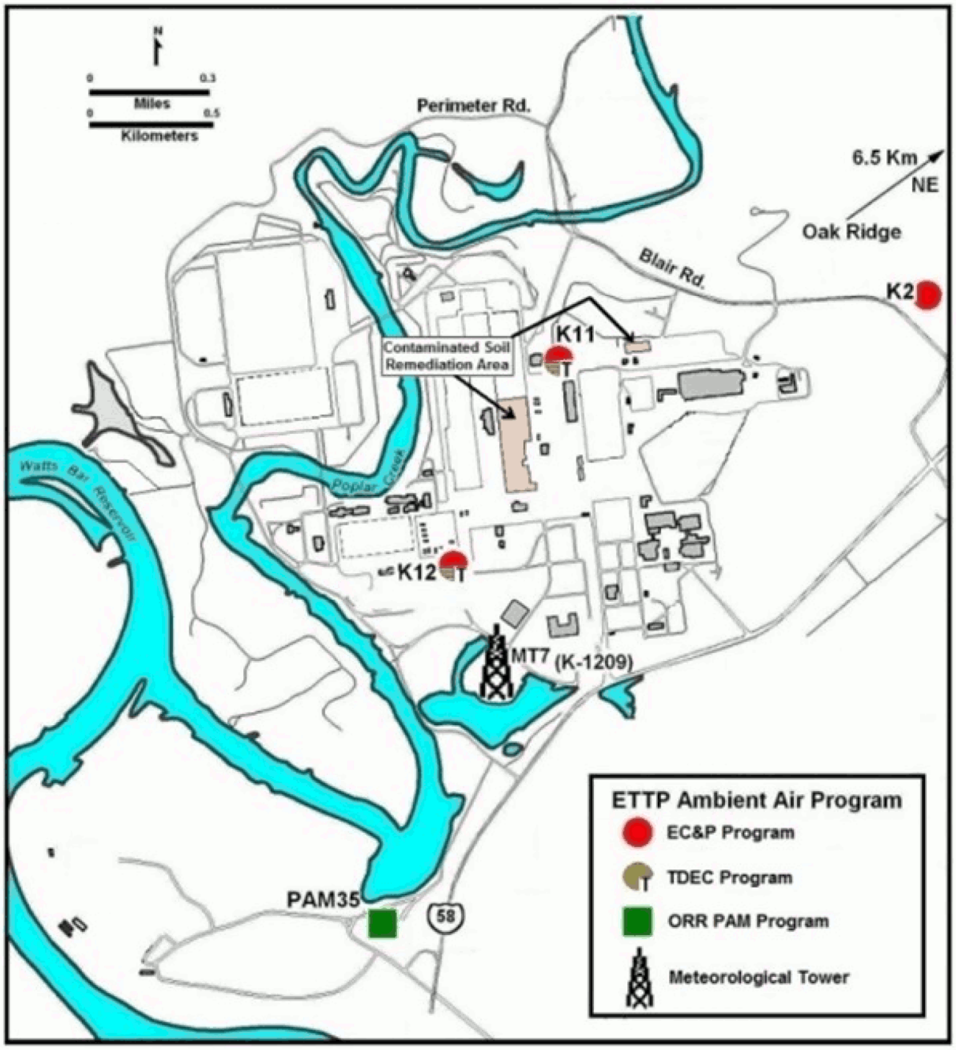
WRPLOT View - Lakes Environmental Software

Figure 2.3-23: Precipitation Wind Rose for Tower L



WRPLOT View - Lakes Environmental Software

Figure 2.3-24: East Tennessee Technology Park Ambient Air Monitoring Station Locations



Acronyms:

ETTP = East Tennessee Technology Park
MT = meteorological tower
ORR = Oak Ridge Reservation
PAM = perimeter air monitoring
TDEC = Tennessee Department of Environment and Conservation

Figure 2.3-25: Photo of Tower L with Wind Measurements at 15 and 30 Meters

Figure 2.3-26: Photo of Tower L Including Ground Cover

2.4 HYDROLOGY

Information is provided in this section as it relates to groundwater and surface water features at the facility site, to support analyses and evaluations of consequences of uncontrolled release of radioactive material.

The Hermes site is less than 3.5 miles from the Clinch River Nuclear (CRN) site, for which an Early Site Permit (ESP) was recently approved. Kairos Power has reviewed the Clinch River Early Site Permit Application (CR-ESPA), Part 2, Site Safety Analysis Report (SSAR) (Reference 8) hydrology sections, and has determined that a portion of the CR-ESPA, Part 2, SSAR hydrology information, specifically the hydrosphere information of Section 2.4.1, is applicable to the Hermes site. Accordingly, the hydrosphere information described herein does not repeat the content of the CR-ESPA, Part 2, SSAR Section 2.4.1 which is directly applicable to the site but will instead only discuss information that supplements or is different for the site. A reader of the equivalent CR-ESPA subsection can effectively substitute the term ‘Hermes’ for ‘CRN’ for direct applicability.

The effect of potential floods on sites along streams, rivers, and lakes is considered. Effects and consequences of a precipitation induced flood, seiche, surge, standing water, drainage, or seismically induced flood (such as might be caused by dam failure) are considered. At the site, hazards of tsunami, or distant or locally generated "sea waves,” are negligible and not applicable given the site’s inland location. River blockage on the Clinch River arm of the Watts Bar Reservoir, and flow diversion on Poplar Creek and the Clinch River are also considered. Additional information will be provided with the application for the Operating License.

In addition, credible hydrological events at the site are established. These events are used for design considerations of the reactors and their auxiliary facilities. Such design considerations ensure the safe operation and shutdown of the non-power reactors. Potential release of radioactive material in the event of a credible hydrologic occurrence is bounded by postulated events analyzed in Chapter 13 of this Preliminary Safety Analysis Report (PSAR).

There are existing, comprehensive, hydrological studies that were performed for the Clinch River and the East Tennessee Technology Park (ETTP), at which the facility resides. The studies considered relevant are:

- Federal Emergency Management Agency (FEMA) Flood Insurance Study (FIS) for Roane County, Tennessee, dated November 18, 2009 (FEMA FIS Number 47145CV000B) (Reference 5).
- Flood Hazard Evaluation for Y-12 (Bear Creek) and K-25 (Poplar Creek) 2015 Update prepared by Barge for URS/CH2M Oak Ridge LLC (UCOR), April 2015 (Reference 2).
- Tennessee Valley Authority, Flood Analysis for Department of Energy Y-12, ORNL, and K-25 Plants, 1991 (Reference 6).
- Tennessee Valley Authority (TVA) Clinch River Nuclear Site Early Site Permit, Part 2, Site Safety Analysis Report (Reference 8).

The hydrological descriptions described herein are based on a review of the relevant, readily available published reports and maps, and where available, records and unpublished reports from federal and state agencies. Information on the site hydrology has been acquired from a consideration of these sources and from site-specific investigations.

2.4.1 Hydrological Description

The site is located on the ETTP just west of Poplar Creek, approximately 3 miles upstream of the confluence of Poplar Creek and the Clinch River. Because of the site's location with respect to this confluence, both Poplar Creek and the Clinch River arm of the Watts Bar Reservoir are considered potential flooding sources.

The facility location is on the eastern edge of the approximately 40-acre former K-33 building footprint. The approximate grade elevation at the location is 765 feet above mean seal level (feet msl) in North American Vertical Datum of 1988 (NAVD 88). Flooding elevations from historical reports as well as TVA dam elevations are typically reported in feet msl of the National Geodetic Vertical Datum (NGVD 29). The difference in these two datums at the site is on the order of several inches. Therefore, comparisons of reported elevations in the NGVD 29 datum to the site grade in the NAVD 88 datum are qualified by the small difference in the datums and do not alter interpretations or conclusions made throughout Section 2.4. The plant grade of 765 ft msl is about 21 feet above a Poplar Creek normal water surface elevation of 744 feet msl and about 24 feet above the Clinch River arm of the Watts Bar Reservoir normal water surface elevation of 741 feet msl. A site location map is provided in Figure 2.4-1.

This section describes the hydrological processes governing the movement and distribution of water in the existing environment at and around the proposed site. Descriptions are limited to only those parts of the hydrosphere that may affect or be affected by building and operation of the non-power reactors at the site and relies on the data and analyses performed for the CRN site (Reference 8).

The Tennessee River (Watts Bar Reservoir) is the principal waterway flowing through the county. Its shoreline is dotted with summer homes and resorts but void of industry. Watts Bar Reservoir, controlled by Watts Bar Dam, is an integral part of the TVA flood control and navigation system. The Tennessee River watershed contains a total of 17,000 square miles as it flows out of Roane County (Reference 5).

The downstream four miles of the Clinch River arm of the Watts Bar Reservoir, the second largest waterway in Roane County, is in backwater from Watts Bar Lake. The Clinch River originates in southwest Virginia and passes through TVA's Norris and Melton Hill Dams before entering Roane County. The Clinch River watershed in Roane County is mainly wooded except for the ETTP and TVA Kingston Steam Plant (Reference 5).

A total of 4,413 square miles of drainage area comprises the Clinch River watershed at its mouth near Kingston, Tennessee. The Emory River, a tributary to the Clinch River, originates on the Cumberland Plateau region northwest of Roane County and flows through rugged undeveloped land before entering Roane County near the City of Harriman. Approximately 780 square miles of drainage area feeds the Emory River at Harriman. The Little Emory River joins the Emory River north of Kingston and flows through mostly forested watershed that contains a total of 42 square miles.

Whites Creek, a tributary of the Tennessee River, is contained in a natural gorge above mile 6. There is a total of 120 square miles of drainage area, 0.3 miles below Black Creek. Black Creek flows through the City of Rockwood before entering the unincorporated areas of Roane County. It flows through pastureland as it parallels U.S. Highway 72. The Black Creek's watershed contains approximately 12 square miles of drainage area at its mouth (Reference 5).

Caney Creek and Pawpaw Creek, tributaries to the Clinch River, flow through undeveloped land before entering the Clinch River and have a total of 3 and 9 square miles of drainage area, respectively. Indian Creek heads along the southern slope of the Cumberland Plateau divide around elevation 790 feet msl NGVD at the northern edge of the Town of Oliver Springs. Indian Creek flows through a restrictive gap just upstream of Mineral Springs Branch (Reference 5). The watershed above the gap is heavily strip mined so flood flows on Indian Creek are heavily laden with silt, which in general contributes to increased flood damage and significant stream channel sedimentation (Reference 5).

Poplar Creek heads out of Walden Ridge northeast of the Town of Oliver Springs and flows south parallel to Tennessee Highway 118 before entering the Town of Oliver Springs. The total drainage area of Poplar Creek at the upstream limits of the study, is 26.6 square miles (Reference 5). On the Oliver Springs side of Poplar Creek, the floodplain is mainly agricultural, with housing well above the 100-year flood, and commercial development inside the floodway at Tennessee Highway 61 and Highway 62 bridges (Reference 5).

East Fork Poplar Creek has its origin on Chestnut Ridge, south of the residential area of Oak Ridge. The creek flows generally northwesterly into Oak Ridge and parallels Tennessee Highway 62 in this reach. Near the intersection of Tennessee Highways 95 and 62, at an elevation of approximately 850 feet msl NGVD, it is joined by a tributary that drains the western portion of the populated section of Oak Ridge. From here, the main stream flows approximately 12.5 miles southwest to enter Poplar Creek approximately 5.5 miles above its mouth in Watts Bar Reservoir backwater (Reference 5).

Historical records of flooding for Poplar Creek and East Fork Poplar Creek have been documented in the FIS (Reference 5):

Poplar Creek – Since 1902, the June 29, 1928 flood is the highest known flood of record. The estimated discharge was 17,000 cubic feet per second (cfs) at mile 13.8 with a recurrence interval of 40 years (Reference 5). On September 29, 1944, a severe flood on Poplar Creek caused extensive damage to crops in the flood plain. The estimated discharge was 13,000 cfs with a recurrence interval of 25 years at the Highway 61 Bridge. During July 5-7, 1967, a total of 9.5 to 11 inches of rain fell on Oliver Springs. Field crops and gardens were heavily damaged and roads were badly washed. At the USGS gaging station at Highway 61 and 62 the flood crest on July 6, 1967 was 3.8 feet below the June 1928 flood. The flood crest on July 12, 1967 was 2.2 feet lower than the July 6, 1967 crest, and the July 29, 1967 crest was 4.4 feet below the July 6, 1967 crest. On November 26-28, 1973, a total of 8.7 inches of rain fell on the Oak Ridge gage, producing the highest gage reading of record (27.1 feet or elevation 770.6 feet msl NGVD) at the USGS stream gage at mile 13.94. At Highway 61 and 62 this flood was about 1.8 feet below the June 1928 and 2 feet higher than the July 1973 floods (Reference 5).

East Fork Poplar Creek – Major floods occurred on June 29, 1928, September 29, 1944, November 28, 1973, and April 4, 1977. Elevations, discharges, and recurrence intervals for the 1928 and 1944 floods are not cited because they cannot be compared directly to flooding under current conditions, due to channel changes and watershed urbanization. The November 28, 1973, and April 4, 1977, floods were about equal in magnitude. These floods reached an elevation of 770.2 feet msl NGVD with a recurrence interval of approximately 30 years at 3.3 miles upstream of the confluence with Poplar Creek. Only minor damage occurred as a result of these floods (Reference 5). Based on modeling results for East Fork Poplar Creek and Poplar Creek in Reference 6 for the 100-year return period, a flooding elevation of 771.2 feet msl at East Fork Poplar Creek mile 3.32US projected downstream to Poplar Creek mile 3.17US east of the site would be approximately 749.6 feet msl, which is more than 15 feet below site grade. Therefore, flooding levels similar to the 1970s floods on East Fork Poplar Creek would have no impact at the site.

A schematic of the Clinch River and Poplar Creek watersheds is shown in Figure 2.4-2.

2.4.1.1 Surface Water

The Clinch River originates in western Virginia and flows generally to the southwest, joining the Tennessee River near Kingston, Tennessee. Along with its tributaries, the Clinch River drains an area of about 4,416 square miles (mi^2) in the Upper Tennessee River basin. The drainage pattern in the Clinch River watershed is characterized by both long straight river reaches and frequent sharp bends, which are a consequence of the long parallel ridges and valleys of the Valley and Ridge Physiographic Province through which the Clinch River and its tributary streams flow. The site is bordered on the south, north and east by Poplar Creek, which immediately connects to the Clinch River arm of the Watts Bar Reservoir at the south, at about Clinch River Mile (CRM) 14.5, which is about 14.5 river miles upstream from the confluence with the Tennessee River (Figure 2.4-3). The drainage area of the Clinch River watershed above the location of the site is 3,370 square miles, about 76% of the total watershed area. Two dams, owned and operated by TVA, are located on the Clinch River upstream of the site: the Melton Hill Dam is located at about CRM 23 and Norris Dam is located just downstream from the confluence with the Powell River at about CRM 80 (Figure 2.4-4). Releases from each of these dams influence Clinch River flows at the site. Norris Dam is operated for flood control and hydroelectric power generation of 110 megawatts electric (MWe). The reservoir provides 1,113,000 ac-ft of flood storage and has a water-surface elevation that varies 29 ft from summer to winter during a year with normal rainfall. Melton Hill Dam does not provide significant flood storage, but it does provide 79 MWe of hydroelectric power generation, and it includes a navigation lock that allows barge traffic 38 miles upstream to Clinton, Tennessee. Both reservoirs provide significant shoreline and in-water recreational opportunities (Reference 3).

Two dams located on the Tennessee River influence flows in the Clinch River at the site: Watts Bar Dam and Fort Loudoun Dam, both owned and operated by TVA (Figure 2.4-4). Watts Bar Dam is located at Tennessee River mile 530, about 38 miles downstream from the Clinch River confluence and about 52 river miles downstream from the site. The reach of the Clinch River downstream from Melton Hill Dam, which includes the river adjacent to the site, is part of the Watts Bar Reservoir and is referred to as the Clinch River arm of the Watts Bar Reservoir. Fort Loudoun Dam is located at Tennessee River mile 602.3, about 35 miles upstream from the Clinch River confluence, and releases water into the Watts Bar Reservoir. Watts Bar and Fort Loudoun Dams are operated for hydroelectric power generation, flood control, and navigation. Both reservoirs provide significant shoreline and in-water recreational opportunities. Some characteristics of the reservoirs that influence flows at the site are listed in Table 2.4-1. Because the Clinch and Tennessee Rivers near the site are regulated by releases from reservoirs operated by TVA, relevant information about the flows adjacent to the site were obtained from TVA. Releases from reservoirs are determined by rainfall, runoff, and management objectives (e.g., flood control). Reservoirs are drawn down in the winter to provide flood storage, and minimum elevations are established to maintain a navigation channel. Reservoir elevations are maintained at higher levels during the summer and fall (generally May through October) (Reference 3).

2.4.1.2 Groundwater, and Groundwater Extraction/Injection

The facility design does not include groundwater withdrawal or injection. No planned future injection or withdrawal of groundwater is expected to have an impact on facility operation or safety.

2.4.2 Floods

The following paragraphs provide brief descriptions of the previous flood studies and estimated flooding elevations in the vicinity of the site.

<u>FEMA Flood Insurance Study for Roane County, Tennessee (Reference 5)</u>

Four Clinch River return period flood profiles were provided in this FIS: 10-year (10% probability of occurring in a given year), 50-year (2% probability of occurring in a given year), 100-year (1% chance of occurring in a given year) and 500-year (0.2% chance of occurring in a given year). Approximate flooding elevations at the confluence of the Clinch River and Poplar Creek and resulting flooding depths at the site are provided in Table 2.4-2.

Flood Hazard Evaluation for UCOR dated April 2015 (Reference 2)

This study was performed for the purpose of evaluating flooding risk at Department of Energy (DOE) critical facilities including ETTP. The previous NPH was performed by TVA in 1991. The Flood Hazard Evaluation addressed both Poplar Creek and the Clinch River. Flooding events evaluated in this analysis ranged from the 4% (25-year return interval) to a Probable Maximum Flood (PMF) (Reference 2).

The Clinch River flooding event results reported in the 2015 study were taken from different models. TVA developed a Clinch River hydraulic model in 2003 using U.S. Army Corps of Engineers (USACE) Hydrologic Engineering Centers River Analysis System (HEC-RAS) software. The 4% to 0.001% flooding elevations reported in the 2015 study were taken from that model and shown in Table 2.4-3.

For the UCOR 2015 evaluation of Poplar Creek, watershed precipitation and hydraulics were evaluated. A hydraulic model of Poplar Creek developed by TVA in 1991 was converted to HEC-RAS. The model geometry was not revised as it was not part of the scope of work for the 2015 update. The period of record precipitation datasets in the watershed were also reviewed to evaluate changes since 1991.

Based on the 2015 study (Reference 2), flooding elevations at the site are controlled by Poplar Creek for the 4% to 0.01% flood events and by the Clinch River for the 0.005% to the PMF. Estimated flooding elevations and depths at the site based on the UCOR study are provided in Table 2.4-3. A PMF study will be discussed with the application for an Operating License.

2.4.2.1 Rainfall Frequency Curve Development

Rainfall frequency curves were developed for local area rainfall, using estimates of the 5-, 10-, 25-, 50-, and 100-year rainfall and the TVA maximum probable precipitation (TVA Storm) and probable maximum precipitation (PMP). Order-of-magnitude estimates of the probability of the TVA Storm and PMP were made based on extrapolating flood data, watershed rainfall, and extraordinary storm occurrences (Reference 6).

Rainfall-frequency estimates for durations from 5 to 60 minutes, and return periods up to 1000 years, were obtained from National Oceanic and Atmospheric Administration's (NOAA) Atlas 14, Volume 2 Version 3 (Reference 2). TVA Storm and PMP estimates were obtained from Hydrometeorologic Report No. 56 (Reference 6).

To establish the exceedance probability of the TVA Storm, earlier estimates of flood-frequency curves at 36 long-record stream gaging stations, extrapolated to computed TVA maximum probable flood (TVA Flood) estimates, were reviewed. The 36 watersheds were within the Tennessee Valley watershed and ranged from 31.9 square miles to 21,400 square miles. The TVA Flood exceedance probabilities ranged from 1×10^{-3} to 10^{-9} with a median and mode of 1×10^{-5}. The exceedance probability of the TVA Storm was assumed equal to that of the TVA Flood.

Earlier estimates of the exceedance probability of the PMF were reviewed; in particular, estimates based upon two rainfall frequency analyses, which considered (a) rainfall on watersheds within the Tennessee Valley and (b) storms occurring east of the 105th meridian.

The exceedance probability of the PMF can be assumed equal to the exceedance probability of the PMP. This is because the PMP is defined as an event approaching the physical upper limit of precipitation.

To determine the chance of a PMP storm striking a selected area, observed 6- and 24-hour rainfalls for storms covering 10 square miles and for 72-hour rainfalls covering 5,000 square miles east of the 105th meridian greater than or equal to 50 percent of the PMP (Reference 6) were evaluated. Although no PMP storms have occurred, data are available from storms that were from 50 to 90 percent of the PMP storm and struck storm areas of 10 and 5,000 square miles. Extrapolation of these data to the PMP indicate that exceedance probabilities range from 2.4×10^{-7} to 5.6×10^{-8}. Based on this information, an exceedance probability of 1×10^{-8} was assumed for the PMP (Reference 6).

Determination of confidence intervals for the rainfall-frequency curves requires knowledge of the population distribution. The population distribution was assumed to be the Fisher-Tippett Type I distribution with application as described by Gumbel, herein referred to as the Gumbel distribution. This is consistent with NOAA procedures which use the Gumbel frequency distribution. A least-squares regression analysis was used to fit the Gumbel distribution to the sample points for the 5-minute and 1-hour rainfall. A Smirnov-Kolmogorov (S-K) goodness-of-fit test accepted the hypothesis that the sample points were from the Gumbel distribution. However, the S-K goodness-of-fit test is not robust at small exceedance probabilities. Therefore, to be conservative, the upper bound of the 99 percent confidence interval (10^{-6}) was adopted as the exceedance probability of the PMP. Extrapolation of the rainfall frequency curves to the PMP with a probability of 1×10^{-6} results in an exceedance probability of 5×10^{-5} for the TVA Storm (Reference 6).

2.4.2.2 Dam Failures Floods

In the 1991 TVA study listed in Section 2.4 (Reference 6), Norris and Melton Hill Dams (separately) were postulated to fail seismically, concurrent with the one-half PMF, and in non-flood conditions. Dam failures were treated as hypotheticals and TVA neither implied or conceded that its dams are inadequate to withstand great floods and/or earthquakes that may be reasonably expected to occur in the region under consideration.

TVA has a program of inspection and maintenance carried out on a regular schedule to keep its dams safe. Instrumentation of the dams to help keep check on their behavior was installed in many of the dams during original construction. Other instrumentation has been added since and is still being added as the need may appear or as new techniques become available.

In short, TVA has confidence that its dams are safe against catastrophic destruction by any natural forces that could be expected to occur.

Failure of Norris and Melton Hill Dams during one-half the PMF was assumed to occur at peak reservoir levels; at Norris, this elevation was 1036.9 and at Melton Hill, 799.3. Reservoir levels for the non-flood failure were assumed at normal maximum pool elevation 1020 for Norris and 795 for Melton Hill. Failure of Norris Dam in both events would overtop and fail the Melton Hill Dam. Unsteady flow techniques were used to route the floods resulting from the dam failures.

The stability of Norris Dam was reanalyzed for various scenarios in 2014 (Reference 2). The analysis concluded that the concrete sections and the earthen embankment were stable under seismic conditions analysis (Reference 2). Therefore, the postulated failure analysis is different than the 1991 study. The controlling seismic event producing the highest elevations on Watts Bar reservoir was used for the seismic postulated failure evaluation. This includes a postulated failure of Melton Hill Dam. A "sunny day" postulated failure scenario was developed for Norris Dam as part of the TVA studies.

2.4.2.3 Landslide Induced Flooding

Flooding may occur as the result of waves generated from of landslides downstream or upstream of the site. The site is adjacent to Poplar Creek, which is a body of water that is not subject to significant riverbank landslides.

Flood waves from landslides into upstream reservoirs required no specific analysis. Based on the review of CR-ESPA, Part 2, SSAR, the borders of the Watts Bar and upstream reservoirs indicate the absence of major elevation relief in nearby reservoirs. The volume of material entering the nearby reservoirs from potential landslides is not significant compared to the available detention space in reservoirs. Any waves created from landslides would not result in site flooding due to the large difference in elevation between the maximum normal pool elevation at the site.

2.4.3 Credible Hydrological Events and Design Basis

Based on the prior studies discussed above, the credible hydrological events for the siting and design of the reactors are set according to the site-specific study performed for the ETTP (Reference 2). The credible hydrological event for the design basis is defined for a probability of $4x10^{-5}$ (25,000 year return period). This return period is appropriate for structures, systems, and components (SSCs) of Flooding Design Category 4 (FDC-4) (Reference 4). For such events, the design basis flooding level elevation at the site is 759.9 feet msl (5.1 feet below plant grade of 765.0 feet msl). The PMF is not used in the design basis of SSCs, however, a PMF analysis will be discussed with the application for an Operating License.

2.4.3.1 Design Bases for Flooding in Streams and Rivers

The Design Basis Flood elevation is 759.9 feet msl.

2.4.3.2 Design Bases for Site Drainage

The maximum flooding level for site drainage is set at 765.0 feet msl (plant grade), 5.1 feet above the $4x10^{-5}$ credible event flooding elevation.

2.4.3.3 Other Site Criteria Design Bases

The site relies on the existing topography so that runoff water naturally drains to the east, south, and west with flow directed to Poplar Creek. The final grading plan of the site takes full advantage a favorable topography by employing a number of measures, including grading slopes and diversion ditches to divert runoff water to Poplar Creek. Detailed design of the site layout and facilities at the site, including the storm water drainage system will be conducted and the final site grading and site layout designed such that safety-related SSCs are able to function.

2.4.4 Groundwater

Subsurface investigations encountered groundwater starting at depths approximately 10 feet below the ground surface. The depth to saturated groundwater will vary with seasonal conditions. These water table level variations will be addressed in the application for an Operating License.

2.4.5 Groundwater Contamination

The primary coolant for the reactors is salt based and not water based. Secondary support systems containing water (i.e., the Decay Heat Removal System and the Component Cooling Water System) could experience small amounts of tritiated water migration. Tritium contamination and the potential for liquid effluent releases from secondary support systems are monitored through periodic sampling and tritium concentration measurements in support system water inventory.

Tritium is controlled in the facility by the Tritium Management System (TMS) as described in Section 9.1.3. Total tritium inventory is monitored to comply with inventory limits set by the maximum hypothetical accident analysis assumptions and dose limits in 10 CFR 100.11. The TMS maintains a level of overall tritium capture capacity to minimize tritium releases from the plant and satisfy PDC 60. Tritium releases in effluents are controlled within the effluent limits in 10 CFR 20.

Additionally, tritium capture is carried out in the environments surrounding the primary heat transport system to collect permeating tritium as well as any tritium released from limited gas leakage out of interfacing systems during normal operations or maintenance activities.

Section 11.1.7 provides additional information regarding the environmental monitoring program.

2.4.6 References

1. Not used
2. BARGE, *Flood Hazard Evaluation for Y-12 (Bear Creek) and K-25 (Poplar Creek)*. 2015.
3. U.S. Nuclear Regulatory Commission, Environmental Impact Statement for an Early Site Permit (ESP) at the Clinch River Nuclear Site, Final Report, NUREG-2226, Volume 1, April 2019.
4. Department of Energy, *Natural Phenomena Hazard Analysis and Design Criteria for DOE Facilities*, Standard 1020-2016, DOE. 2016.
5. Federal Emergency Management Agency, *Flood Insurance Study (FIS) for Roane*, FEMA FIS Number 47145CV000B. 2009.
6. Tennessee Valley Authority, *Flood Analysis for Department of Energy Y-12, ORNL, and K-25 Plants*, TVA. 1991.
7. Not used
8. Tennessee Valley Authority, *Clinch River Nuclear Site Early Site Permit Application Part 2 Safety Analysis Report Revision 2*, TVA, ADAMS Accession No. ML19030A358. 2019.
9. Tennessee Valley Authority, *Hydroelectric, Knoxville, Tennessee,* ADAMS Accession No. ML18036A967. TN5241, TVA. 2017.
10. Tennessee Valley Authority, *Lake Levels, Knoxville, Tennessee*. ADAMS Accession No. ML18036A968. TN5242, TVA. 2017.

Table 2.4-1: Reservoirs that Influence Flows at the Confluence of Clinch River and Poplar Creek

Reservoir	Water Body	Purpose	Flood Storage (acre-ft)[1]	Area (Acre)[1]	Operating Elevation (feet msl)[2]	Date[1] Completed
Norris	Clinch & Powell Rivers	Power generation, flood control, recreation	1,113,000	33,840	992-1,020	1936
Melton Hill	Clinch River	Power generation, navigation, recreation, water supply	Negligible	5,470	793-795	1963
Watts Bar	Tennessee, Clinch, & Emory Rivers	Power generation, flood control, navigation, recreation, water supply	379,000	39,090	735-741	1942
Fort Loudoun[3]	Tennessee River	Power generation, flood control, navigation, recreation, water supply	111,000	14,600	807-812.8	1943

NOTES:

(1) Reference 9

(2) Reference 10

(3) Fort Loudoun Reservoir is connected by a canal to Tellico Reservoir on the Little Tennessee River. A regulated spillway on Tellico Dam is used only during extreme flooding

Table 2.4-2: Roane County FEMA FIS Flooding Elevation (Projected to Site)

Annual Exceedance Probability	Flood Elevation (*) (feet msl)	Estimated Depth at Hermes 2(**) (feet)
0.1 (10 %)	744.8	-20.2
0.02 (2%)	746.0	-19.0
0.01 (1%)	747.0	-18.0
0.002 (0.2%)	749.5	-15.5

(*) Flood elevations from the FIS Study are from Clinch River at the mouth of Poplar Creek in feet msl NGVD 29 datum.

(**) Site grade is 765 feet msl NAVD 88 datum. Estimating flood depths shown for the Hermes 2 site do not incorporate a conversion and are qualified due to a small difference in the vertical datums, on the order of several inches. A negative number indicates dry site based on plant grade El 765 NAVD 88.

Table 2.4-3: UCOR Poplar Creek and Clinch River Flooding Elevations (Projected to Site)

Annual Exceedance Probability	Flood Elevation (*) (feet msl)	Estimated Depth at Hermes 2 (**) (feet)
0.04 (4%)	747.2	-17.8
0.01 (1%)	749.7	-15.3
0.002 (0.2%)	752.7	-12.3
0.0005 (0.05%)	755.2	-9.8
0.0001 (0.01%)	758.2	-6.8
$5x10^{-5}$ (0.005%)	759.4***	-5.6
$4x10^{-5}$ (0.004%)	759.9***	-5.1
$1x10^{-5}$ (0.001%)	766.6***	1.6

(*) Flood elevations from UCOR Study are feet msl NGVD 29 datum.

(**) Site Grade is 765 feet msl NAVD 88 datum. Estimated flooding depths shown for the site do not incorporate a conversion and are qualified due to a small different in the vertical datums, on the order of several inches. A negative number indicates a dry site.

(***) Flood elevations for higher annual exceedance probabilities up to 0.01% are controlled by Poplar Creek. Flood elevations for lower annual exceedance probabilities at or below 0.0005% are controlled by backwater from the Clinch River.

Figure 2.4-1: Location of Site

Figure 2.4-2: Poplar Creek and Clinch River Watersheds

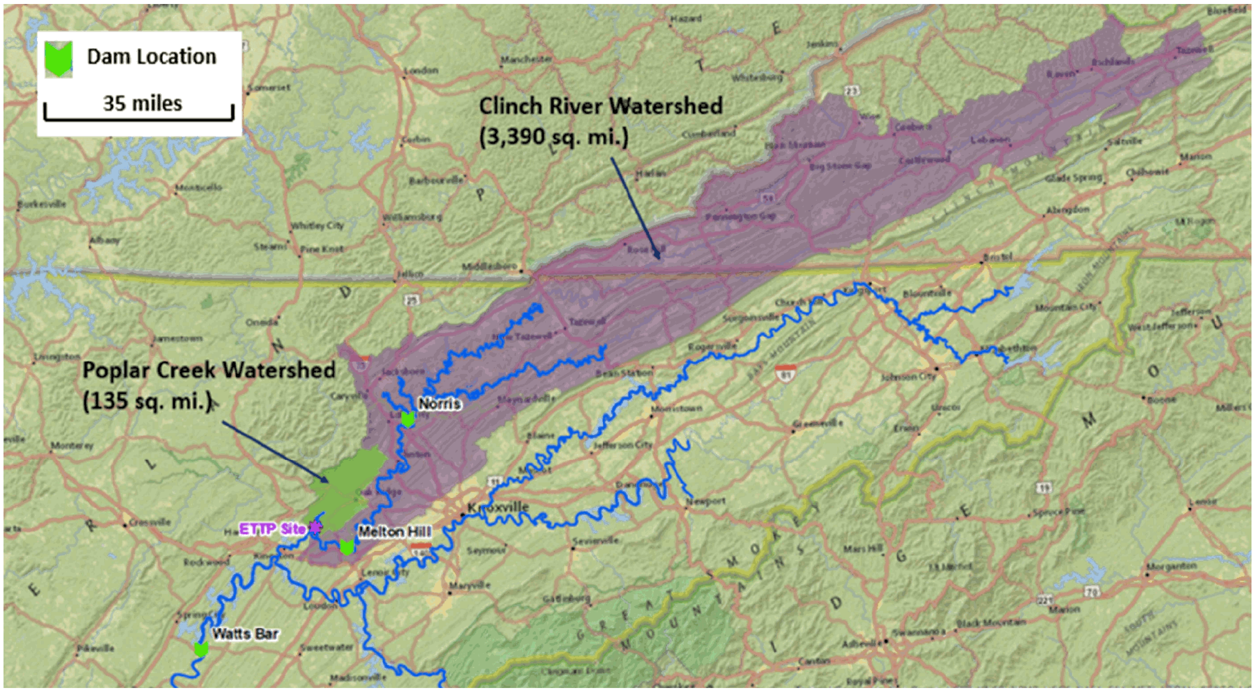


Figure 2.4-3: Streams and Rivers near the Site

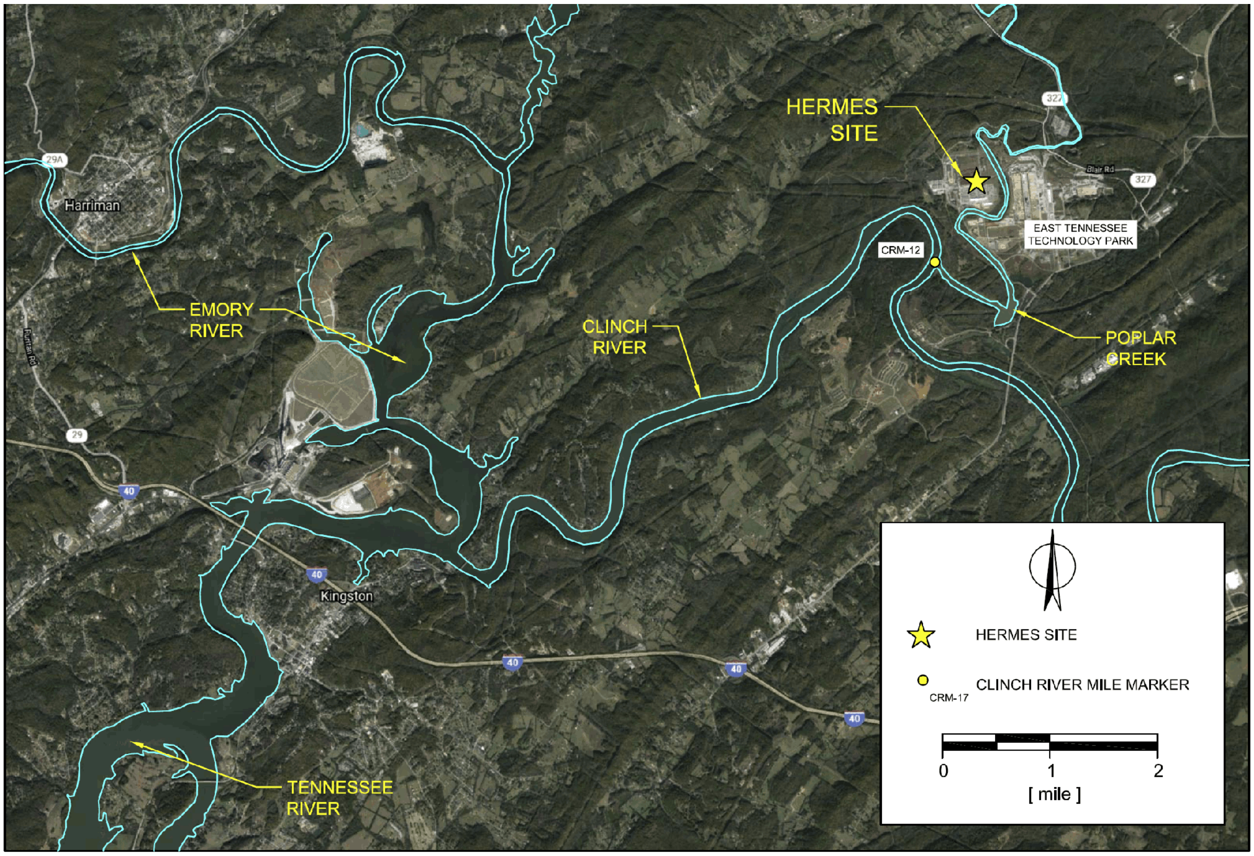


Figure 2.4-4: Location of Dams that Influence Flows at Site

2.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

This section describes the geologic, geophysical, seismic, and geotechnical characteristics of the site and the surrounding region. Site characteristics are developed that provide the basis for the required design input for SSCs. The seismic design basis is based on existing information that includes the CRN detailed Probabilistic Seismic Hazard Analysis (PSHA) and current seismic hazard publications. The design basis will later be supplemented with the site-specific data retrieved from a detailed geophysical and geotechnical investigation.

As described further in Section 2.5.3, the facility PSHA is adapted from the PSHA presented in the Clinch River Early Site Permit Application, Part 2, Site Safety Analysis Report (Reference 1). The PSHA approach is an enhancement to the existing guidance in NUREG-1537 and follows the approach delineated by the American Nuclear Society in ANS 2.29, “Probabilistic Seismic Hazard Analysis” (Reference 2), and the American Society of Civil Engineers in ASCE 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities” (Reference 3). Because the vibratory ground motion at the site is determined with the PSHA approach, the organization of material in this section differs from that of NUREG-1537. The information is organized to reflect the general process of the PSHA.

This section includes a general regional characterization of the seismicity that can potentially impact the ground acceleration demands at the site. Although NUREG-1537 does not explicitly require the specification of a radius of a region to account for seismicity, the regional description of the site investigation covers a radius close to 200 miles around the site.

Because the site is approximately 3.5 miles from the CRN site, for which an Early Site Permit was recently approved, Kairos Power has reviewed the CR-ESPA, Part 2, SSAR regional geology sections and determined that the CR-ESPA, Part 2, SSAR regional geology information is directly applicable to the site. Accordingly, the regional geology information described herein does not repeat the content of the CR-ESPA, Part 2, SSAR Section 2.5.1 which is directly applicable to the site but will instead only discuss information that supplements or is different for the site. A reader of the equivalent CR-ESPA, Part 2, SSAR subsection can effectively substitute the term ‘Hermes’ for ‘CRN’ for direct applicability.

Similar to the approach described above for the regional geology, portions of the site geology discussion of the CR-ESPA, Part 2, SSAR Section 2.5.2 have also been determined to be applicable for the site. Accordingly, Section 2.5.2 will not repeat all the content of the CR-ESPA, Part 2, SSAR Section 2.5.2 but will instead only discuss information that is different or supplemental for the site.

To a lesser extent, the remaining 2.5 sections, Section 2.5.3 Vibratory Ground Motion, Section 2.5.4 Potential for Surface Deformation, and Section 2.5.5 Foundation Interface, also derive content from the CR-ESPA, Part 2, SSAR and only present new information that is different or supplemental for the site.

Section 2.5.2 presents the site geology and discussions related to the potential for subsurface deformation from the action of karstic dissolution. The presence of karst is not uncommon throughout the Valley and Ridge physiographic province. Previous investigations, particularly those from the CR-ESPA, Part 2, SSAR, have documented the presence of karst. Karst is specifically addressed in Section 2.5.2.1 and Section 2.5.4.3 within the context of potential for surface deformation.

The site geology description is complemented with a site-specific geologic and geotechnical investigation. The investigation, consisting of a series of geotechnical borings, ground water monitoring wells, and a laboratory testing program, yields the subsurface stratigraphic information necessary to develop a design of foundation systems adequate to support safety. The site investigation and its findings are provided in Section 2.5.2.2 and 2.5.2.3.

The vibratory ground motion analysis, Section 2.5.3, was performed to obtain the seismic design basis applicable to the site.

Vibratory ground motion analysis includes a PSHA, and the development of the seismic Design Response Spectra (DRS). Vibratory ground motion is presented in Section 2.5.3 and the resulting DRS in Section 2.5.3.4.6.

Section 2.5.4 addresses the potential for surface deformation that may be brought upon by hazards such as sinkholes, faults, and/or soil liquefaction. Given the subsurface conditions, and foundation interface plans along with fill placement, there is no potential for liquefaction at the site. Active surface faults have not been documented within the site area. There are several inactive faults in the site area. These are being addressed as part of the site area mapping and published literature. The potential for karst formations and sinkholes requires additional confirmatory investigations, and these are discussed in Section 2.5.4.

Foundation interfaces are presented in the form of subsurface profiles showing the elevation and placement of the proposed facilities and the engineered fills. It is demonstrated that the current site plans and foundation designs provide the necessary bearing support for the reactors and their auxiliary facilities. The foundation interface setting is discussed in Section 2.5.5.2.

2.5.1 Regional Geology

The regional geological information is documented in the NRC approved CR-ESPA, Part 2, SSAR (Reference 1) and is not repeated herein. This regional information is directly applicable to the site. The site is in the Valley and Ridge physiographic province. The area within a 200-mile radius of the site includes six physiographic provinces. These include, from west to east: the Central Lowlands province, Interior Low Plateaus province, the Appalachian Plateaus province, including the Cumberland Plateau at the latitude of the site region, the Valley and Ridge province, the Blue Ridge province, and the Piedmont province. Each of these six physiographic provinces is described in detail in the CR-ESPA, Part 2, SSAR (Reference 1).

2.5.2 Site Geology

The site lies within the Appalachian Valley and Ridge Physiographic Province of East Tennessee. This Province is characterized by elongated, northeasterly-trending ridges formed on highly resistant sandstone and shale. Between ridges, broad valleys and rolling hills are formed primarily on less resistant limestone, dolomite, and shale.

Published geologic information, Geologic Map of the Elverton Quadrangle, Tennessee, by Limiszki, P.J., Tennessee Geological Survey, 2015 (Reference 4), indicates that this site is underlain by bedrock from three geologic formations. The Mascot Dolomite Formation underlies the northwest side of the site, the Murfreesboro Limestone Formation underlies the southeast side of the site, and the Pond Springs Limestone Formation underlies the center of the site. These formations trend in a northeast-southwest direction through the site and parallel to the regional trend of the Valley and Ridge province.

The Mascot Dolomite formation of the Knox Group is generally composed of well-bedded light-gray dolomite containing minor amounts of limestone. The Murfreesboro Limestone consists of thin- to thick-bedded gray limestone and the Pond Springs Formation, of thin- to medium-bedded maroon to olive-gray limestone. These formations are each generally composed of fine-grained, siliceous dolomite interbedded with limestone or predominately limestone. These formations typically weather to produce a thick reddish or orangish-brown clay overburden soil. The formations also contain trace silica nodules in the form of chert, that is resistant to weathering and typically scattered throughout the residuum.

A subsurface stratigraphy was developed for the site from a geotechnical boring program. Details of the boring program, along with subsurface profiles are included in Section 2.5.2.3.

2.5.2.1 Karst

Since the bedrock formations underlying this site contain carbonate rock (e.g., limestone/dolomite), the site could be susceptible to the carbonate hazards of irregular weathering, cave and cavern conditions, and overburden sinkholes. Carbonate rock, while appearing very hard and resistant, is soluble in slightly acidic water. This characteristic, plus differential weathering of the bedrock mass, is responsible for the hazards. Of these hazards, the occurrence of sinkholes is potentially the most damaging to overlying soil-supported structures. Sinkholes primarily occur due to differential weathering of the bedrock and flushing or raveling of overburden soils into the cavities in the bedrock. The loss of solids creates a cavity or dome in the overburden. Growth of the dome over time or excavation over the dome can create a condition in which rapid, local subsidence, or collapse of the roof of the dome, occurs.

Borings B-3 and B-6 were drilled at the facility location, in the area at which bedrock is closest to the ground surface.

The geotechnical investigation at the site encountered indications of karstic activity. Surface signs of sinkhole activity at the site were not detected. Remnants of the old K-33 building foundations remain undisturbed and fully integrated within the soil matrix. As discussed in 2.5.4.3, it is noted that although zones of soft soils were encountered beginning at a depth of approximately 28 feet in Boring B-1, which may be indicative of the potential for karstic activity, Boring B-1 is located more than 1000 feet away from the proposed location of the reactors.

The karst investigation will be complemented with a set of tests and surveys. These include site reconnaissance, analysis of LiDAR imaging, inventory of surface depressions in the site area, deeper borings at the reactor locations, laboratory analyses of rock cores, and the elaboration of the karst model. This information will be provided with the application for an Operating License.

2.5.2.2 Site Subsurface Stratigraphy

Subsurface conditions were explored between March 22 and March 30, 2021, with six soil test borings (designated B-1 through B-6) and six observation trenches. The boring plan is presented in Figure 2.5-1. Location of the borings, existing piezometers, and observation trenches were established with field Global Positioning System (GPS) handheld devices.

2.5.2.3 Soil Borings

The borings were advanced in the overburden soil using hollow stem augers with an inside diameter of 3¼ inches with a Diedrich D-50 drill rig. The drill crew worked in accordance with the American Society for Testing and Materials International (ASTM) D6151 (Reference 5), the Standard Practice for Using Hollow-Stem Augers for Geotechnical Exploration and Soil Sampling. Split-spoon sampling and standard penetration tests (SPTs) were performed in accordance with ASTM D1586, the Standard Test Method for Standard Penetration Test (SPT) and Split-Barrel Sampling of Soils (Reference 6). Split-spoon samples were obtained and SPT performed with a standard 1.4-inch inside diameter (ID), 2-inch outside diameter (OD) split-spoon sampler at 2½-foot intervals to depths of 10 feet and on 5-foot intervals thereafter. The sampler was first seated 6 inches and then driven an additional foot with blows of the 140-pound hammer falling 30 inches. The number of hammer blows required to drive the sampler the final foot was recorded and is designated as the standard penetration resistance (N-value) with units of blows per foot (bpf). The N-value provides a general indication of in-situ soil conditions and has been correlated with certain engineering properties of soils. An automatic trip drop hammer was used for the standard penetration resistance testing. The automatic hammer has a higher efficiency than a manual hammer and may yield lower N values. The N values reported on boring logs are the field values without any adjustments or corrections. In addition, one thin-walled tube sample was obtained in Boring B-1 at a depth of 30 to 32 feet.

Figure 2.5-23 and Figure 2.5-24 provide profile sections of the soil borings and observation trenches with collected boring data summary. Section 2.5.2.3.2 describes the stratigraphic column based on the boring findings.

The soil samples obtained during the field activities were visually classified by members of the field engineering staff in accordance with ASTM D2488, the Standard Practice for Description and Identification of Soils (Visual-Manual Procedure) (Reference 7). Laboratory testing was performed to classify soils. The extent of the laboratory testing was limited to basic index testing for site characterization. A more comprehensive evaluation will be provided with the application for an Operating License.

A description of the overburden soils is provided in Section 2.5.2.3.2.

2.5.2.3.1 Observation Trenches

The observation trenches were advanced using a CASE CX210D tracked excavator. All trenches except OT-3 were excavated in one direction adjacent to a remnant foundation and in a direction perpendicular to the foundation. OT-3 was excavated as one long trench approximately 120 feet long. Soils were logged in accordance with ASTM D2488 (Reference 7). The foundations were cleaned, observed, and photographed. Depths of soil strata, foundations, and ground water were recorded.

2.5.2.3.2 Subsurface Stratigraphy

Table 2.5-1 summarizes the subsurface stratigraphy at the site. Each of the borings and trenches encountered topsoil at the ground surface with thickness ranging from 4 to 12 inches thick.

Fill soil was encountered beneath the surface cover in each of the borings and trenches. The fill ranged in depth from about 12 to 21 feet. In trenches OT-2, OT-4, OT-6A, and OT-6B the entire interval of fill soil was not penetrated. Typically, the fill consisted of red to yellow, red fat clay (CH) with limestone and rock fragments. Strata of crushed stone were encountered in several of the observation trenches in thin layers of about 6 to 8 inches thick throughout the fill, and occasionally in layers of 3 to 5 feet thick closer to the ground surface. Concrete foundations were also encountered as deep as about 12 feet. Additional discussions related to the old foundations and site history are presented in Section 2.5.5.1. The SPT N-values for the fine-grained fill ranged from 6 to 100 bpf indicating soil consistencies of medium firm to very hard, although SPT N-values of 100 were likely amplified by rock fragments in the samples. The SPT value for the coarse-grained fill was 26 bpf indicating a medium dense relative density.

Alluvial soils were encountered beneath the fill in Borings B-1 and B-4 to depths of 22 to 31 feet, respectively. Alluvial soils are soils transported to their present location by flowing water. The SPT N-values for the fine-grained alluvium ranged from 8 to 11 bpf indicating soil consistencies of medium firm to firm. The SPT values for the coarse-grained alluvium was 15 bpf indicating a medium dense relative density.

Residual soils were encountered beneath the alluvial soil in Borings B-1 and B-4. Residual soil was encountered beneath the fill in each of the observation trenches, except for trenches OT-01, OT-02, OT-04, OT-05A, OT-06A and OT-06B. Residual soils are soils weathered from the underlying parent bedrock. Residual soils extended to auger refusal at depths ranging from 14.1 to 54.4 feet in the borings and about 13 to 19.5 feet in the observation trenches. The residual soils consisted of red brown, yellow brown, light gray, to dark brown fat clays with varying amounts of chert, fine sand, and weathered rock.

The SPT N-values of the residual soils ranged from 2 bpf to 100 bpf, indicating soil consistencies of very soft to very hard. Based on observations of the cutting, the soil consistencies ranged from very soft to firm. Samples with higher blow counts were typically amplified by refusal material within the samples. Residual soils were encountered to depths ranging from 11 feet to 54.4 feet. Trenches OT-05B and OT-05C were terminated in residual soil.

Moisture content of the boring samples was determined to range from 4.5 to 50.9 percent.

Bedrock at the site consists of dolomitic limestones of different nature. The north portion of the site is underlain by the Mascot formation, a gray, medium to thickly bedded, fresh, hard rock. The bedrock is directly underneath the residuum and presented a Rock Quality Designation (RQD) of 70% to 100%. At the north end of the site, around Boring B-1, the Mascot bedrock was encountered at a depth of about 55 feet.

The midsection of the site, near the area of Boring B-2, is underlain by the Pond Springs formation, which is described as a limestone, light gray, medium bedded, medium jointed. It presented an approximately 5 feet thick weathered layer and quickly transitions to fresh hard rock with RQD of 70%. The Pond Springs bedrock was encountered at a depth of about 35 feet below the ground surface.

The south end of the site is underlain by the Murfreesboro dolomitic limestone. Encountered at depths of about 20 feet near Boring B-5, this formation is light gray, medium, close jointed, with an approximately 3 feet weathered layer. Below the weathered zone, RQD is greater than 80%.

Figure 2.5-2 and Figure 2.5-3 provide the subsurface profiles that are mapped in Figure 2.5-1.

2.5.3 Vibratory Ground Motion

The CRN site is only 3.5 miles away from Hermes. The seismic hazard study for CRN is documented in the CR-ESP application, Part 2, SSAR. The study evaluated new data, methods, and models developed since publication of the 2012 Central and Eastern United States (CEUS) Seismic Source Characterization model (Reference 8). Relevant updates to the CEUS Seismic Source Characterization were incorporated into the site-specific evaluation of the seismic hazard at CRN. The update team performed interviews of experts who have developed data and/or interpretations of seismic sources in the site region, reviewed an updated seismicity catalog developed for CRN, and performed site-specific studies, as needed, to assess the quality of data and uncertainty associated with recently published studies. The updates include geologic/paleoseismic studies within the Eastern Tennessee Seismic Zone (ETSZ); (2) investigations of the Mineral, Virginia earthquake that occurred in or near the Central Virginia Seismic Zone (CVSZ); and (3) revisions to the maximum magnitude distributions for seismic zones in the CEUS Seismic Source Characterization model. The PSHA for CRN incorporates the post CEUS-Seismic Source Characterization updates. Since the site is only separated from CRN by less than 3.5 miles, CEUS Seismic Source Characterization updates performed for CRN are applicable.

The goal of the Senior Seismic Hazard Analysis Committee (SSHAC) process (for the SSHAC or PSHA Levels) is to provide a methodology for developing Seismic Source Characterization and Ground Motion Characterization (GMC) models that capture the center, body, and range of technically defensible interpretations of available data, methods, and models. The terminology “center, body, and range” refers to the complete characterization of epistemic uncertainty. By following the structured methodology of the SSHAC process, reasonable regulatory assurance is provided that the goal of representing the center, body, and range of the characterizations has been met, and thus provides the basis for developing seismic hazard estimates that are reproducible, defensible, transparent, and stable.

The Hermes PSHA is adapted from the PSHA in the CR-ESPA, Part 2, SSAR and the resulting DRS meets the guidance of ASCE 43-19. The motivation for development of a PSHA to establish the seismic design basis originates from the recommendations of ASCE 43-19. ANSI/ANS 2.29 is indicated as the recommended standard to perform the PSHA. The primary objective of ASCE 43-19, in combination with ANSI/ANS 2.29 is to provide a consistent risk-informed design of a nuclear facility that protects the workers, the public, and the environment from the effects of earthquakes, consistent with the intent of NUREG-1537.

Use of the CR-ESPA, Part 2, SSAR PSHA is both appropriate and reasonable given the proximity between both sites. As observed in Figure 2.5-4 (Reference 1), the site region for both CRN and Hermes are essentially equivalent, covering a radius of 200 miles. The seismic hazard for hard rock conditions at the site is therefore equivalent to that reported in the CR-ESPA, Part 2, SSAR. The CR-ESPA, Part 2, SSAR catalog and source model update, however, does not yet include the application of the Next Generation Attenuation (NGA)-East Ground Motion Prediction Equations (GMPE). As discussed in Section 2.5.3.4.5 and Section 2.5.3.4.6, additional margin is built into the design basis to account for changes that could originate from this update.

2.5.3.1 Seismicity

The CEUS Seismic Source Characterization (Reference 8) earthquake catalog, covering the period from 1568 through 2008, is plotted in Figure 2.5-4. The seismicity shown in the figure was further updated to the year 2013 during the CR-ESPA, Part 2, SSAR investigation. The M 5.8 August 23, 2011, Mineral, Virginia, earthquake brought significant new data that warranted a comprehensive catalog update. The nature and activities performed for the update are detailed in the CR-ESPA, Part 2, SSAR.

2.5.3.2 Seismic Source Model

The CR-ESPA, Part 2, SSAR performed a comprehensive review of available geological and seismological data for the site region, as well as for portions of seismic sources that extend beyond the site region. The Seismic Source Characterization is based on the CEUS Seismic Source Characterization report.

It is accepted practice that seismic sources used in a PSHA may be identified based on existing databases and models, with the provision that new information relevant to a seismic source be evaluated and incorporated as appropriate. The baseline for the PSHA is the regional seismic source model developed by the CEUS Seismic Source Characterization. The model was developed using SSHAC Study Level 3 methodology, incorporating the evaluation of uncertainty by capturing the knowledge and contribution of the learned expert community in the field. A full and detailed description of the CEUS Seismic Source Characterization, as it is applied to CRN (or Hermes) can be obtained from the CR-ESPA, Part 2, SSAR.

For a new PSHA, there is the expectation that the adopted Seismic Source Characterization model is to be updated and adapted to the specific site. Distributed seismicity source updates are tied to the revision of the earthquake catalog and the resulting recurrence rates for a given source. Distributed seismicity sources have been updated for the CR-ESPA, Part 2, SSAR.

Table 2.5-2 lists the distributed seismicity sources considered for the PSHA. Another adaptation of a regional Seismic Source Characterization to a site-specific case is the incorporation of the repeated large magnitude earthquakes (RLME) that can potentially impact the hazard at the study site. In several places throughout the CEUS, historical and paleo-earthquake records point to the repeated occurrence of large-magnitude ($M \geq 6.5$) earthquakes that are attributed to previously identified, characterized, and physically delineated tectonic sources. The CR-ESPA, Part 2, SSAR incorporated RLME sources that lie within 640 km of site. The only RLME sources that were found to contribute significantly to a hazard at the site were the New Madrid Seismic Zone (NMSZ) and the Charleston earthquake (Reference 1). The Eastern Tennessee Seismic Zone (ETSZ) is also of special interest, due to its proximity to the site (Figure 2.5-5). The ETSZ, however, after detailed examination, was found not to meet the characteristics of an

RLME source and its seismic activity was therefore incorporated within the effect of distributed sources as part of the CEUS Seismic Source Characterization. The ETSZ and RLME sources are described in the following paragraphs.

2.5.3.2.1 New Madrid Seismic Zone

The three largest historical earthquakes in the CEUS region occurred in the New Madrid area. These earthquakes occurred on December 16, 1811, January 23, 1812, and February 7, 1812 (Reference 1). Considerable uncertainty exists regarding their exact magnitudes. The CEUS Seismic Source Characterization model defines the NMSZ as an RLME to account for large prehistoric earthquakes and the three large events that occurred in 1811–1812 (Reference 8). The NMSZ is approximately 400 km from the site (see Figure 2.5-6). A detailed description of the NMSZ characterization is included in the CR-ESPA, Part 2, SSAR. The characterization of the NMSZ used for CR-ESPA, Part 2, SSAR and Hermes adopts the same uncertainty logic tree of the CEUS-Seismic Source Characterization (Reference 8).

2.5.3.2.2 Charleston

The largest historical earthquake along the eastern U.S. seaboard occurred in Charleston, South Carolina, in 1886. Estimates of the magnitude of this earthquake are based on liquefaction data and isoseismal area regressions and vary from the high-6 to mid-7 range (Reference 8). Charleston is modeled as an RLME source in the CEUS Seismic Source Characterization model. The Charleston RLME source is approximately 420 km from Hermes (Figure 2.5-6). A detailed description of the Charleston characterization is included in the CR-ESPA, Part 2, SSAR. The characterization used for CR-ESPA, Part 2, SSAR and Hermes adopts the same uncertainty logic tree of the CEUS-Seismic Source Characterization (Reference 8).

2.5.3.2.3 Post CEUS-Seismic Source Characterization Study of the Eastern Tennessee Seismic Source Zone

Evaluation of the ETSZ Post CEUS-Seismic Source Characterization as discussed in the CR-ESPA, Part 2, SSAR is applicable to Hermes.

2.5.3.2.4 Updated Seismic Source Parameters

Recent, post CEUS Seismic Source Characterization updates of seismic source parameters were performed as part of the CR-ESPA, Part 2, SSAR, as described in Section 2.5.3.2. These parameters included maximum magnitude and earthquake recurrence rates and are applicable at the site.

2.5.3.3 Correlation with Earthquake Activity

The CEUS Seismic Source Characterization earthquake catalog (Reference 8) includes earthquakes in the CEUS from 1568 through the end of 2008, and its development is discussed in Section 2.5.3.1. The entire CEUS Seismic Source Characterization earthquake catalog comprises 10,984 dependent and independent earthquakes of uniform moment magnitude E[M] > 2.2, and 3,298 dependent and independent events of E[M] > 2.9. The catalog includes 6,965 and 2,563 independent events of magnitude E[M] > 2.2 and E[M] > 2.9, respectively. For seismicity rate calculations, dependent and small events are removed, which results in fewer earthquakes. However, patterns of seismicity are better illustrated when these events are included, as shown Figure 2.5-7 and Figure 2.5-8. As described in Section 2.5.3.1, the catalog has been updated for the CRN Site to include events through mid-September 2013. The updated CEUS Seismic Source Characterization catalog is applicable for Hermes. Detailed discussions related to correlation between earthquake activity and the distributed seismicity and RLME zones considered for Hermes are described in the CR-ESPA, Part 2, SSAR. As discussed in

Section 2.5.3.4.5 and Section 2.5.3.4.6, additional margin is built into the design basis to account for changes that could originate from the update to the earthquake catalog.

2.5.3.4 Design Response Spectra

The DRS is defined using the following process:

1. The hard-rock uniform hazard response spectra (UHRS) is established at the ASCE 43-19 relevant annual probability of exceedance levels:

    • For Seismic Design Category 3 (SDC-3), applicable to the reactors and surrounding safety-related structures:

    - Performance goal (P_f) = 1 x 10^{-4}
    - Annual frequency of exceedance (H_D) to establish the Scale Factor (SF) to account for the slope of the hazard curve = 1 x 10^{-3}

    • Other SSCs that are non-safety related are designed to the local building code, the 2012 International Building Code (IBC), which is consistent with NUREG 1537.

2. The subsurface profile derived from the geotechnical subsurface investigation is compared to the soil columns at CRN Location A (Reference 9). The site-specific shear wave velocity profiles are estimated based on the findings from the geotechnical investigation. Equivalencies in expected site response are inferred as shown in Section 2.5.3.4.2, such that the UHRS at the control point elevation at CRN is also applicable at the control point elevation at the site.
3. The UHRS at the site elevation control point is established using the ratio of the CRN UHRS at Location A to the UHRS at hard rock for 1 x 10^{-4}, and applying the same ratio to the other three probability levels as defined in Step 1. This approach is applicable because the CRN site response is practically linear given the nature of the rock in the subsurface. The set of UHRS is established using the spectral shape reported in the CR-ESPA, Part 2, SSAR for Location A for the annual frequency of exceedance of 1 x 10^{-4}.
4. The ASCE 43-19 SF and DRS for SDC-3 are established based on the UHRS and Location A spectral shape, for the four annual probabilities of exceedance listed in Step 1 above.
5. The hazard levels reported in the USGS National Seismic Hazard Mapping Project (NSHMP) for rock sites (Site Category A) at 4 x 10^{-4} are compared to the 4 x 10^{-4} UHRS in Step 1. This comparison establishes an upper bound of the hazard at the site by using hazard levels from USGS that are meant for applications that are engineered under higher assumed risk of failure.
6. The SDC-3 DRS is increased to define an upper bound enveloping DRS to account for the uncertainty associated with the approximated shear wave velocities at the site, the NGA East GMPE, and post 2013 Seismic Source Characterization catalog updates.

The DRS will be supplemented with site response spectra analyses that rely on in-situ shear wave velocity measurements derived from the PSHA and updated as appropriate in the application for an Operating License.

2.5.3.4.1 Uniform Hazard Response Spectra

The UHRS for hard rock conditions reported in the CR-ESPA, Part 2, SSAR is applicable at Hermes. The PSHA resulting hard rock hazard curves are plotted in Figure 2.5-9. The UHRS for hard rock conditions is derived from the hazard curves and plotted Figure 2.5-10. The four (4) annual probabilities of exceedance are presented in the plots.

2.5.3.4.2 Soil Columns at the facility and CRN

The reactors are located towards the southeast corner of the original K-33 footprint (See Figure 2.5-11). Section 2.5.2.2 details the foundation interface. The foundation mat of the SDC-3 structures is deployed at a depth of about 20 feet below grade. The safety-related structures are founded on concrete fill on competent bedrock of the Murfreesboro dolomitic limestone. Shear wave velocity values for limestone like the one encountered at the site are in the range of 2,500 to 3,000 meters per second (m/s) (Reference 10 and Reference 11). This range is equivalent to the reported values in the CR-ESPA, Part 2, SSAR, which originate from site-specific velocity measurements. Figure 2.5-12 provides a comparison between the CRN Location A (Reference 1) and assumed site velocity profiles. Since the reactors are deployed over the Murfreesboro limestone, it is possible to define the site ground motion control point as an outcropping ground motion at the elevation of the bedrock horizon. Therefore, the UHRS at the site is considered equivalent to the UHRS at CRN Location A. As discussed in Section 2.5.3.4.5 and Section 2.5.3.4.6, additional margin is incorporated into the DRS to account for the uncertainty associated with the lack of site-specific shear wave velocity measurements.

2.5.3.4.3 UHRS at Hermes

ASCE 43-19 guidelines for development of SDC-3 DRS require UHRS at the hazard levels specified in Step 1 of 2.5.3.4.

The 1×10^{-4} UHRS is directly obtained from the CR-ESPA, Part 2, SSAR. The other three annual probability of exceedance levels are developed by scaling the 1×10^{-4} UHRS times a spectral amplification ratio calculated for each frequency (Figure 2.5-13). The resulting UHRS are plotted in Figure 2.5-14.

2.5.3.4.4 Hermes ASCE 4-19 DRS

The SF to establish the DRS is computed using Spectral Accelerations (SA) in the UHRS, as follows (Reference 3):

$$SF = max[SF_1, SF_2, SF_3] \qquad \text{(Equation 2.5-1)}$$

Where: $SF_1 = A_R^{-1.0}$; $SF_2 = 0.6 \cdot A_R^{-1.0}$; $SF_3 = 0.45$

For SDC-3:

$$A_R = \frac{SA_{1\times10^{-4}}}{SA_{1\times10^{-3}}}$$

2.5.3.4.5 USGS NSHMP and Hermes

Figure 2.5-15 compares the hazard levels reported in the USGS NSHMP for rock sites (Site Category A) at 4×10^{-4} to the facility Location A at 4×10^{-4} (Conterminous U.S. 2014 Update) (Reference 12). This comparison establishes an upper bound of the hazard at the site by using the risk levels of USGS. The USGS accelerations are higher than the site-specific study counterpart. The following section utilizes the difference between the USGS and the SDC-3 curves to add margin for the spectra. The margin accounts for the uncertainty associated with the pending shear wave velocity measurements.

2.5.3.4.6 Enveloping Design Response Spectrum

The NGA-East report (PEER Report 2018/08) (Reference 13) indicates that the use of NGA-East (instead of EPRI, 2013, which was the GMPE used at the CRN site) at the Chattanooga, Tennessee, test site results in a factor of 1.5 to 1.7 increase in the 1E-4 UHRS at 1 Hz. This increase is accommodated by introducing additional margin to the DRS at low frequencies.

The final DRS is established by:

a) Increasing the DRS obtained in the previous step by a factor of one (1) plus 40% of the relative difference between the USGS NSHMP and Hermes SDC-3 curves. The factor increases the spectral acceleration levels to define an enveloping DRS that addresses uncertainties associated with the ongoing shear wave velocity measurements,

b) Further increasing the DRS in the low frequency range below 6 Hz by a factor of up to 1.6 at 1.0 Hz to account for the findings of the new 2018 NGA East report, and

c) Multiplying the horizontal DRS by the Vertical to Horizontal (V/H) ratio recommended in the CR-ESPA, Part 2, SSAR.

The resulting DRS for SDC-3 is plotted in Figure 2.5-16 and the spectral values listed in Table 2.5-3.

2.5.4 Potential for Subsurface Deformation

Potential causes for subsurface deformation are surface faults or discontinuities in the foundation bedrocks, liquefaction of saturated sand deposits, or voids in the bedrock formations resulting from karstic limestone dissolution. The following paragraphs describe these hazards.

2.5.4.1 Surface Faulting

This information will be provided in the application for the Operating License.

2.5.4.2 Liquefaction Potential

Each safety-related reactor foundation mat is deployed over a concrete fill placed directly on competent bedrock. Surrounding structures rest either over bedrock or engineered soils after excavation and backfill operations. Section 2.5.5.2 describes the foundation interface conditions for the reactor foundations. Liquefaction at the site is accordingly not an issue for the safety-related reactor foundations. This conclusion and the effects of liquefaction on the surrounding nonsafety-related structure foundations will be addressed in the application for an Operating License.

2.5.4.3 Karst

The geotechnical subsurface investigation encountered limited evidence of voids or karstic dissolution at or near the reactor building locations. Boring B-5 encountered an open void between 21-22.5 feet. As discussed in Section 2.5.2.1, signs of karstic activity at the bedrock/overburden interface were encountered in the area of Boring B-1, located at the Northwest corner of the site, more than 1000 feet away from the reactor foundations. Residuum clays were not encountered south of Boring B-6. The location for the reactors is in the area of Boring B-3 and Boring B-6 and has been selected based on the findings of the geotechnical investigation. The foundation rock for the reactors will be at depths at which no evidence of karstic dissolution is encountered. Over-excavation will be performed at areas at which the compromised bedrock/overburden interface is encountered.

2.5.5 Foundation Interface

This section presents the foundation interface for the reactors and their auxiliary facilities. The foundation layout has been established based on knowledge of the site subsurface conditions gathered from both historical documentation and the subsurface boring exploration campaign. Subsurface profiles are provided in Section 2.5.2.3.

2.5.5.1 Site History

Site preparations for the construction of the original K-33 building involved significant amounts of earth movement and fill placement. Figure 2.5-17 shows topographic maps developed at the site (a) prior to construction of some the Oak Ridge Gaseous Diffusion Plant (ORGDP) facilities (1949), and (b) during construction of the ORGDP and prior to the erection of K-33 (1951) (Reference 14). For the construction of K-33, the site was releveled and graded to foundation footprint elevation. Currently, the site grade is El. 765 feet North American Vertical Datum of 1988 (NAVD 88) (Figure 2.5-1). The historic maps point out an area at which rock was encountered at higher elevations. This observation coincides with the findings of the geotechnical investigation, which encountered rock at the highest near Boring B-3 (Figure 2.5-1). It appears that, in this zone, the excavation for the construction of K-33 reached the top of bedrock horizon. The area was then backfilled with a rock/soil fill material of crushed limestone and reddish clay soil (Reference 15). This observation is also consistent with the findings of the geotechnical investigation at Boring B-3.

Building K-33 was a two-story, 25 meter tall structure with approximately 260,000 square meters of floor space (Figure 2.5-18). Subsequent to its demolition, there is no sign of the above ground remnants of the K-33 building and decontamination has been completed. Figure 2.5-19 shows a present day above ground image of the site.

The K-33 building consisted of a steel braced frame two-story structure resting on isolated spread footing reinforced concrete foundations. The foundations were not removed during demolition and remain underneath the ground surface. There are more than 3,000 isolated spread footing foundations. These range in depths of approximately 4’ to 18’ below the ground surface. Footing footprints are square with dimensions ranging from 4’4” to over 14’ (Reference 16). The width of the columns over the footings varies, on average, between 24” and 42”. The thickness of the footings varies from 14” to 40” and spacing between footings ranges 10’ to 15’. Figure 2.5-20 is an image extract of the original foundation plan showing the north portion of the site (Reference 16). The figure indicates the nature of the density of foundation remnants throughout the subsurface. The geotechnical investigation included observation trenches.

Figure 2.5-21 shows a photograph of footings encountered at OT-2 and OT-6. The OT-6 case shows the column rising from the footing.

2.5.5.2 Plant Layout and Foundation Interface

Plant grade is set at El. 765 NAVD 88. The location for the reactors is in the area of Boring B-3 and Boring B-6 (see Figure 2.5-11). From geotechnical stability and constructability perspectives, at this location, the bedrock interface is just above the depth of the foundation. This condition provides an adequate bearing stratum while reducing the amounts of excavation of hard rock. The foundation surface is to be carefully examined and subjected to inspection after excavation. Weathered zones are to be over-excavated and backfilled with adequate sub-base.

Figure 2.5-22 presents a cross section of the reactors foundation interface. The safety-related structures and the non-safety related structures do not fully share the same foundation system. The bearing system for the safety-related structures is a foundation mat resting directly over sound rock or over a thin concrete fill. It is anticipated that sound bedrock will be very close to the elevation of the bottom of the basemat. Engineered fill supports the lighter portions of the surrounding non-safety related facility. Foundation and structural design aspects are described in Section 3.5.

2.5.5.2.1 Bearing Capacity

Because the facility is supported in bedrock, ample bearing capacity for mat foundations is available. Foundation settlement is expected to be minimal and limited to immediate elastic response of the supporting rock.

Settlement of the non-safety related structure can be controlled because the supporting media is engineered fill. Response is expected to be elastic, and settlement limited to immediate displacement.

Additional details on bearing capacity, settlement and lateral pressure will be provided in the application for an Operating License.

In conclusion, both failure and settlement-controlled bearing capacities are sufficient to safely support the facility at the repurposed site.

2.5.6 References

1. Tennessee Valley Authority, Clinch River Nuclear Site Early Site Permit Application Part 2 Safety Analysis Report Revision 2, ML19030A358, TVA. 2019.
2. American National Standards Institute/American Nuclear Society, “Probabilistic Seismic Hazard Analysis, ANSI/ ANS 2.29. 2020.
3. American Society of Civil Engineers, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," ASCE 43-19. 2019.
4. Limiszki, P., "Geologic Map of the Elverton Quadrangle, Tennessee," Tennessee Geologic Survey. 2015.
5. American Society for Testing and Materials, "Standard Practice for Using Hollow-Stem Augers for Geotechnical Exploration and Soil Sampling," D6151, ASTM. 2015.
6. American Society for Testing and Materials, "Standard Test Method for Standard Penetration Test (SPT) and Split-Barrel Sampling of Soils," D1586, ASTM. 2018.
7. American Society for Testing and Materials, "Standard Practice for Description and Identification of Soils (Visual-Manual Procedure ASTM)," D2488, ASTM. 2017.
8. Electric Power Research Institute, "Technical Report: Central and Eastern United States Seismic Source Characterization for Nuclear Facilities, DOE, NRC," NUREG 2115, EPRI Report 1021097. 2012.
9. Giffels and Valet, Inc., Foundation Plan, Project K Oak Ridge Area. 1954.
10. Earle, S., Physical Geology, s.l.: University of British Columbia. 2019.
11. United States Geologic Survey, "Petrologic and Mineral Physics Database for Use with the U.S. Geological Survey National Crustal Model," USGS, Open-File Report. 2019-1035.
12. United States Geologic Survey, National Seismic Hazard Mapping Project, Web access: https://earthquake.usgs.gov/hazards/interactive/, USGS. 2014.
13. United States Geologic Survey, "Central and Eastern North America Ground-Motion Characterization, NGA-East Final Report," Pacific Earthquake Engineering Research Center, PEER Report 2018/08, USGS. 2018.
14. Giffels and Valet, Inc., "Plan of Fill Placement", Project K Oak Ridge Area. 1952.
15. URS | CH2M Oak Ridge, LLC., "Data Extracted from Confidential Restricted Data, Document GAT-Z-282, Expansion Program at Oak Ridge, Tennessee, Foundations and Soil Bearing Investigations." UCOR. 2014.
16. Giffels and Valet, Inc., "Footings Schedules and Details," Project K Oak Ridge Area. 1954.
17. Hatcher, R. D., Vaughn, J. D. & Obermeier, S. F., Large Earthquake Paleoseismology in the East Tennessee Seismic Zone: Results of an 18-month pilot study, Geological Society of America Special Paper 493, Boulder, CO, 2012.

18. Howard, C. W. et al., Detailed geologic maps of two sites south of Dandridge, Tennessee, Record Evidence of Polyphase Paleoseismic Activity in the East Tennessee Seismic Zone, Geological Society of America Abstracts with Programs, 43(2):31, 2011.

19. King, D. A., The Final Demise of East Tennessee Technology Park Building K-33, Oak Ridge Associated Universities, 2011.

20. Obermeier, S. F., Vaughn, J. D. & Hatcher, R. D., Field Trip Guide: Paleoseismic Features in and near Douglas Reservoir, East Tennessee Seismic Zone, Northeastern Tennessee, 2010.

21. Vaughn, J. D. et al., Evidence for One or More Major Late Quaternary Earthquakes and Surface Faulting in the East Tennessee Seismic Zone, Seismological Research Letters, 81(2):323, 2010.

22. Warrell, K. F., Detailed Geologic Studies of Paleoseismic Features Exposed at Sites in the East Tennessee Seismic Zone: Evidence for Large, Prehistoric Earthquakes, Master's Thesis, University of Tennessee, 2013.

23. American Society for Testing and Materials, Standard Practice for Rock Core Drilling and Sampling of Rock for Site Explorations D2113, ASTM, 2014.

24. Chapman et al, The Eastern Tennessee Seismic Zone - Summary after 20 Years of Network Monitoring, Seismological Research Letters 73 (2):245, 2002.

Table 2.5-1: Subsurface Stratigraphy

Unit		Depth [ft]	Description	N-Value
1	Fill	12 to 21	Red fat clay with gravel, medium stiff to stiff; presence of old K-33 foundation remnants	6 to >50
2	Alluvial soils	22 to 31	Alluvium, medium firm to firm, medium dense	8 to 11
3	Residuum	14 to 55	Residual soils, brown fat clay w/chert and fine sand, very soft to soft	2 to >50
4	Bedrock Mascot (North)	Top at ~55	Dolomite, gray, medium to thickly bedded, fresh, hard RQD 70% to 100%	NA
5	Bedrock Pond Springs (Mid-Section)	Top at ~35	Limestone, light gray, medium bedded, medium jointed, 5 ft of moderately weathered to highly, then fresh hard, RQD 70%	NA
6	Bedrock Murfreesboro (South)	Top at ~20	Limestone, light gray, medium, close jointed, 60°, 3 ft weathering, then fresh, hard, clay filled fracture at 30.5'. RQD 80%	NA

NOTES:
- RQD: Rock Quality Designation
- NA: Not applicable

Table 2.5-2: Distributed Seismicity Sources included in PSHA

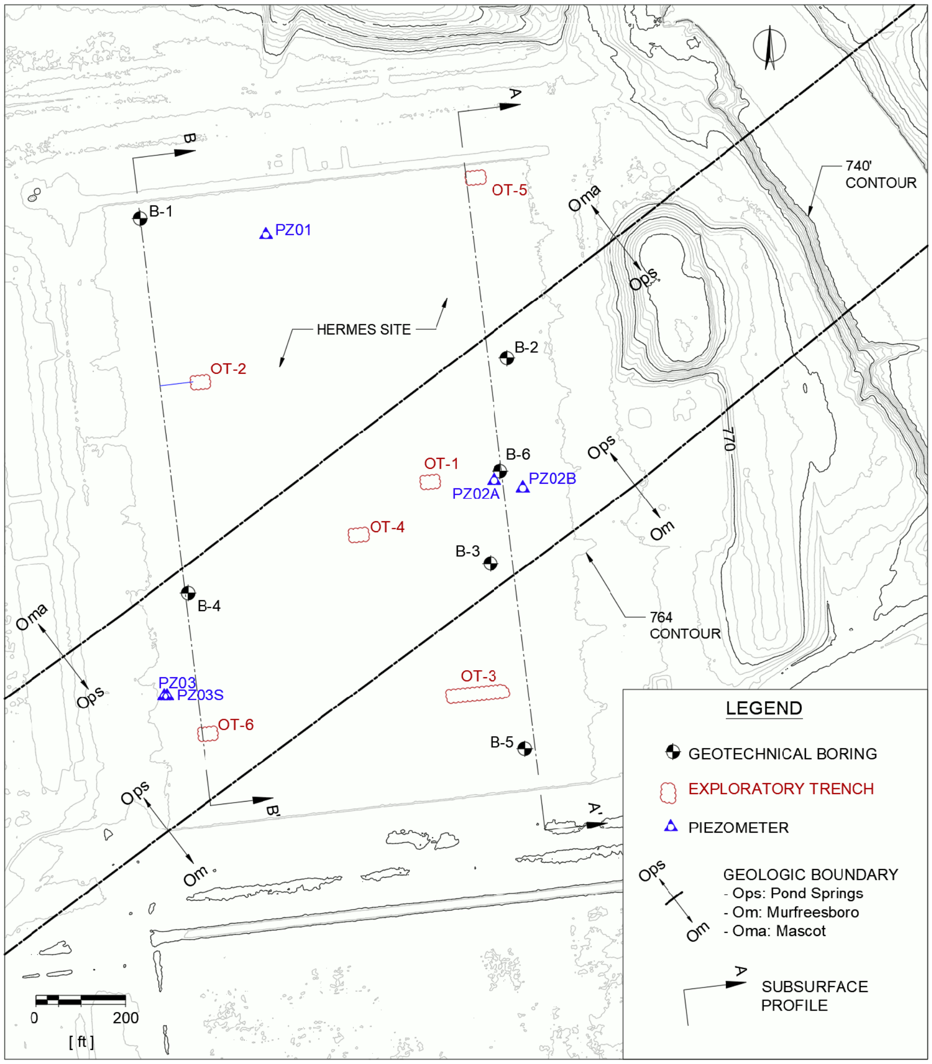
Zone Type	Zone Acronym	Zone Name
Maximum Magnitude Zones	MESE-N and MESE-W	Mesozoic and Younger Extended Crust, narrow and wide geometries
	NMESE-N and NMESE-W	Non-Mesozoic and Younger Extended Crust, narrow and wide geometries
	STUDY_R	CEUS study region
Seismotectonic Source Zones	ECC-AM	Extended Continental Crust – Atlantic Margin
	ECC-GC	Extended Continental Crust – Gulf Coast
	IBEB	Illinois Basin Extended Basement
	MidC-A, B, C, D	Midcontinent Craton
	PEZ-N, PEZ-W	Paleozoic Extended Crust (Narrow and Wide)
	RR, RR-RCG	Reelfoot Rift, Reelfoot Rift with Rough Creek Graben

Table 2.5-3: Design Response Spectra

f [Hz]	Horizontal (g)	Vertical (g)
	SDC-3	SDC-3
0.10	0.009	0.008
0.13	0.012	0.010
0.15	0.014	0.011
0.20	0.019	0.015
0.30	0.029	0.023
0.40	0.039	0.032
0.50	0.050	0.040
0.60	0.061	0.049
0.70	0.073	0.058
0.80	0.085	0.067
0.90	0.097	0.077
1.00	0.110	0.087
1.25	0.114	0.090
1.50	0.119	0.095
2.00	0.135	0.111
2.50	0.143	0.118
3.00	0.161	0.137
4.00	0.200	0.175
5.00	0.236	0.212
6.00	0.258	0.237
7.00	0.288	0.271
8.00	0.317	0.304

f [Hz]	Horizontal (g)	Vertical (g)
	SDC-3	SDC-3
9.00	0.346	0.337
10.00	0.375	0.371
12.50	0.389	0.404
15.00	0.384	0.412
20.00	0.375	0.428
25.00	0.357	0.425
30.00	0.341	0.408
35.00	0.324	0.393
40.00	0.306	0.376
45.00	0.286	0.361
50.00	0.267	0.350
60.00	0.230	0.317
70.00	0.215	0.293
80.00	0.208	0.267
90.00	0.206	0.245
100.00	0.204	0.231

Figure 2.5-1: Boring Layout



Note - Sectional Views A-A’ and B-B’ are provided in Figures 2.5-23 and 2.5-24.

Figure 2.5-2: Subsurface Profile A-A'

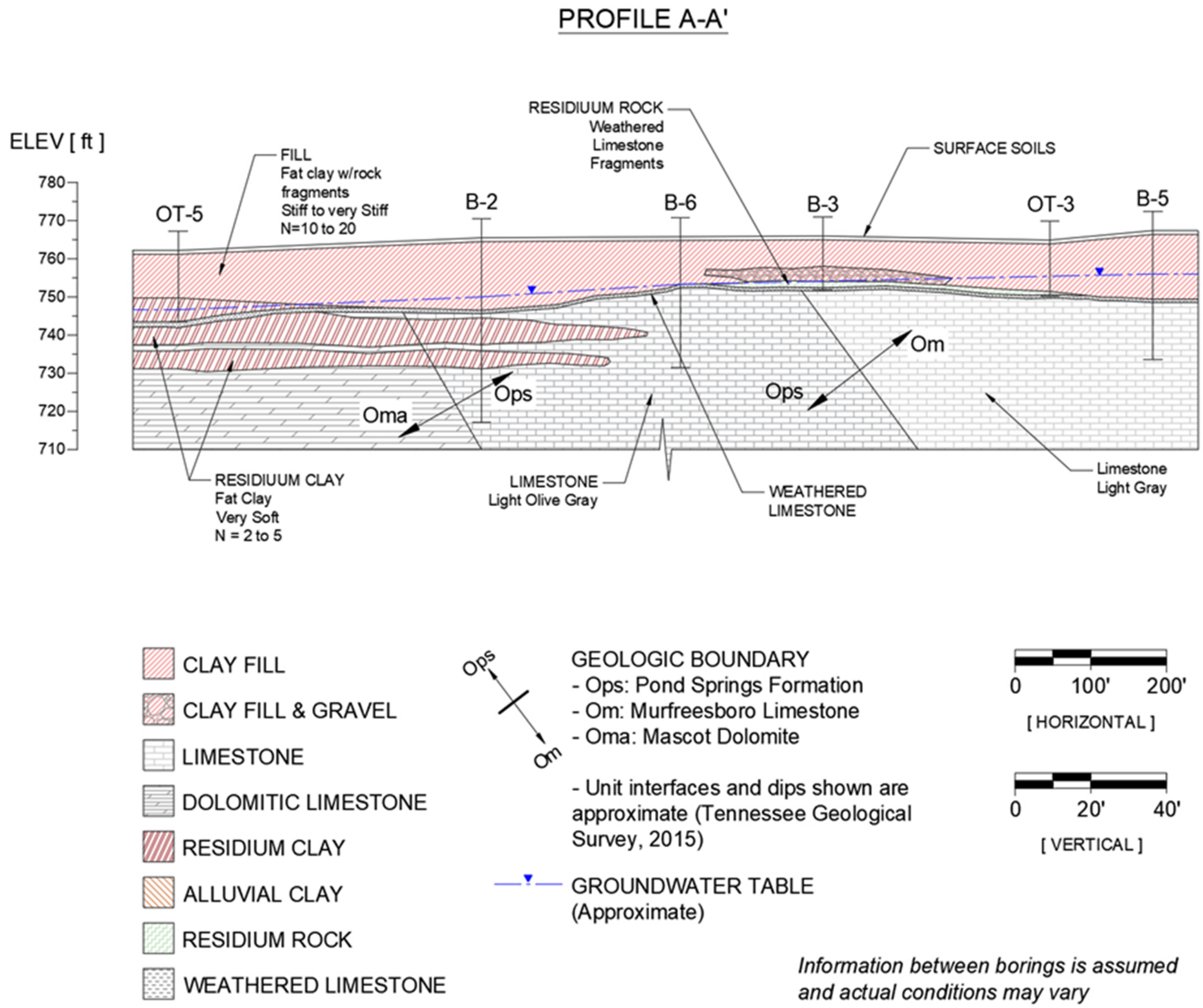


Figure 2.5-3: Subsurface Profile B-B’

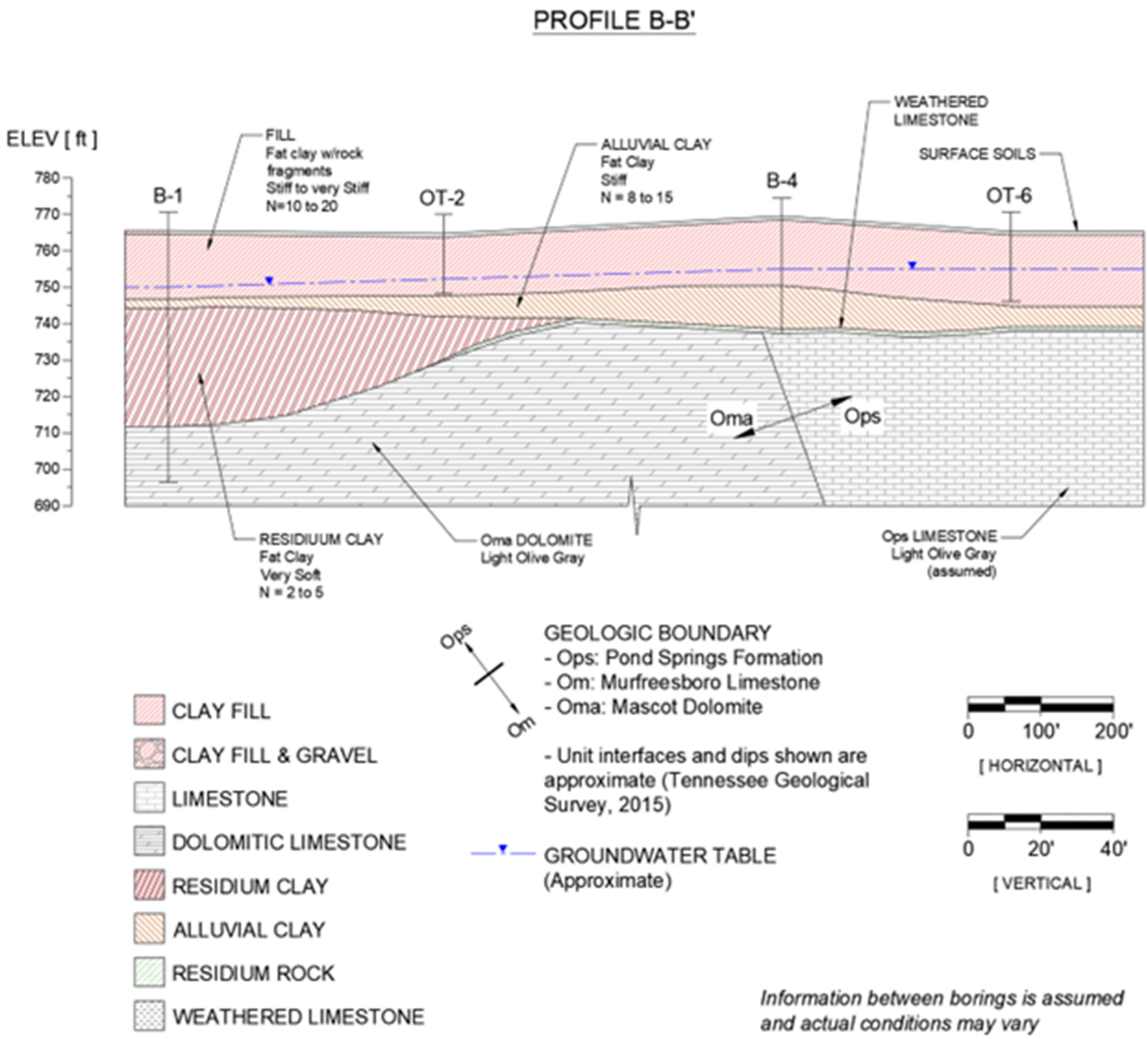


Figure 2.5-4: Plot of Seismicity Within 320 km of the Facility

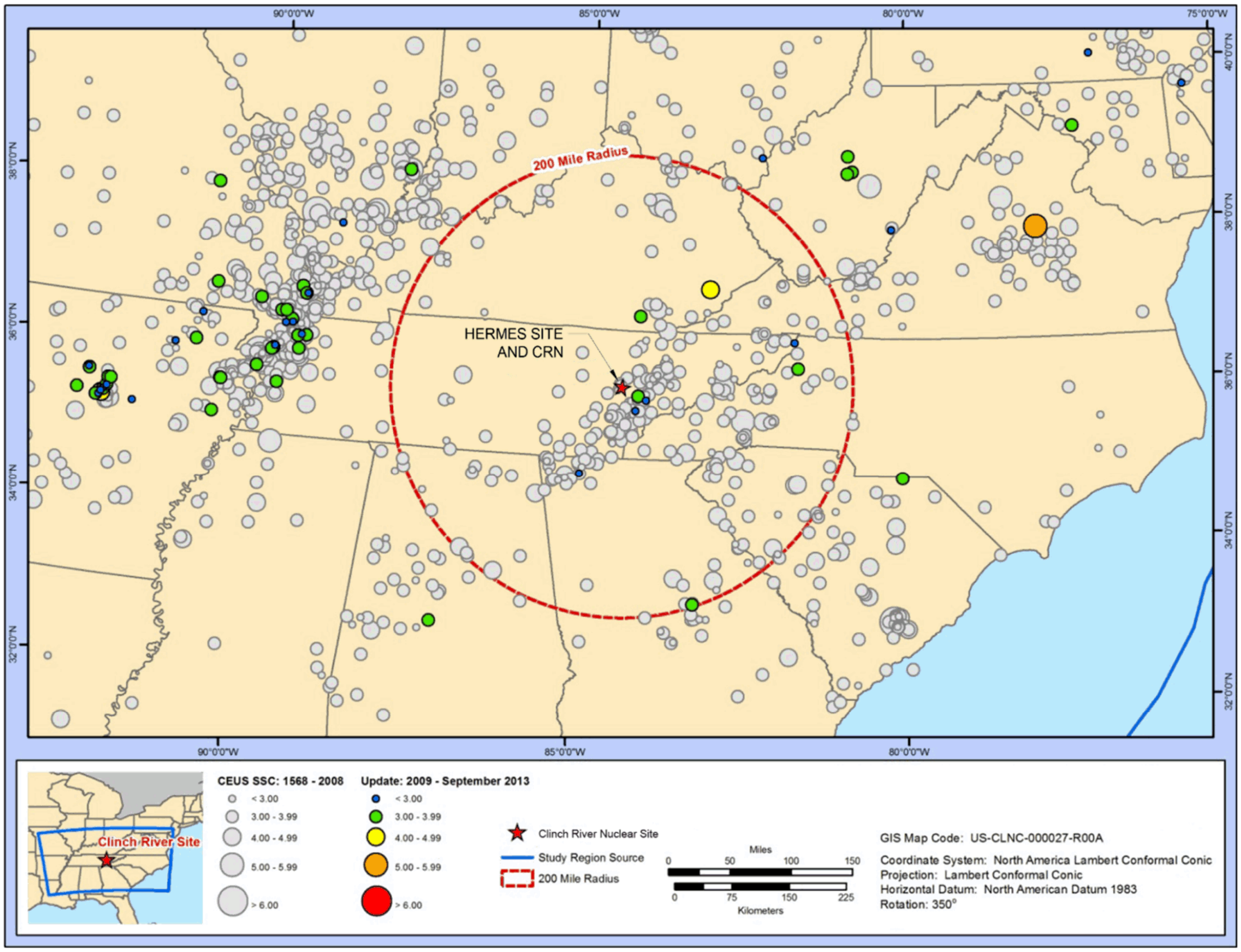


Figure 2.5-5: Eastern Tennessee Seismic Zone

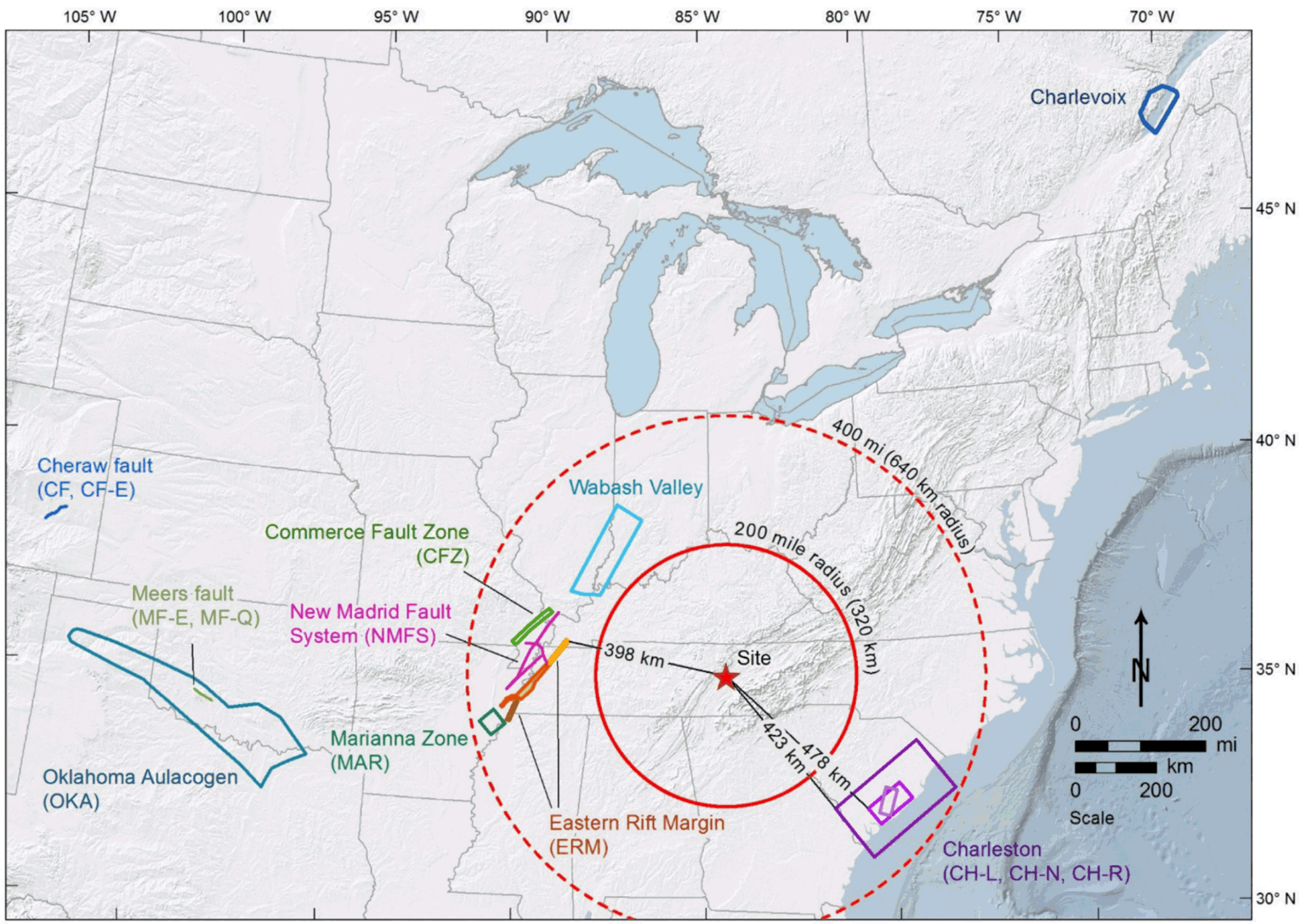


Figure 2.5-6: RLME source zones in the CEUS-Seismic Source Characterization

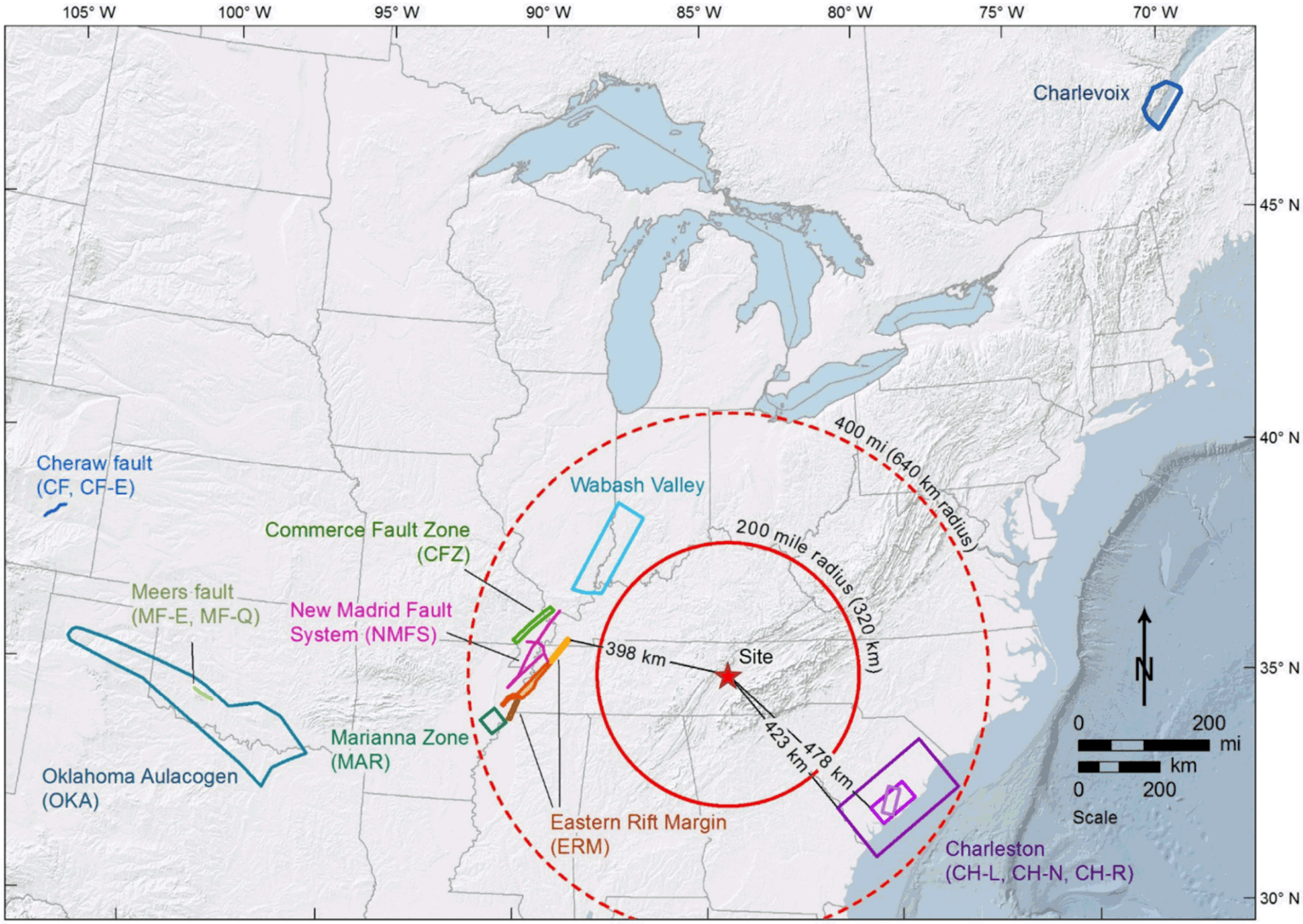


Figure 2.5-7: Maximum Magnitude and Repeated Large Magnitude Earthquake Source Zones

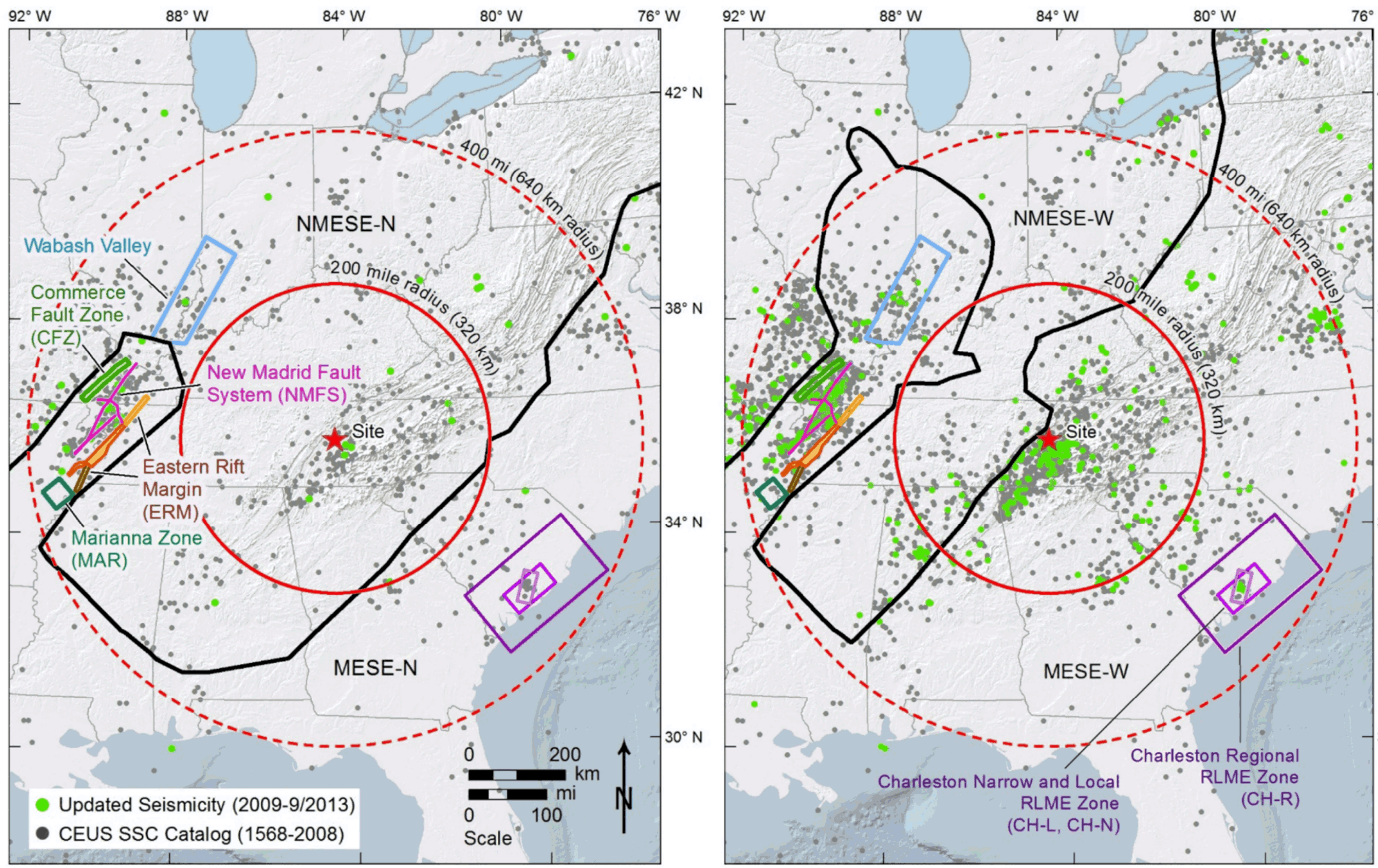


Figure 2.5-8: Seismotectonic and Repeated Large Magnitude Earthquake Source Zones

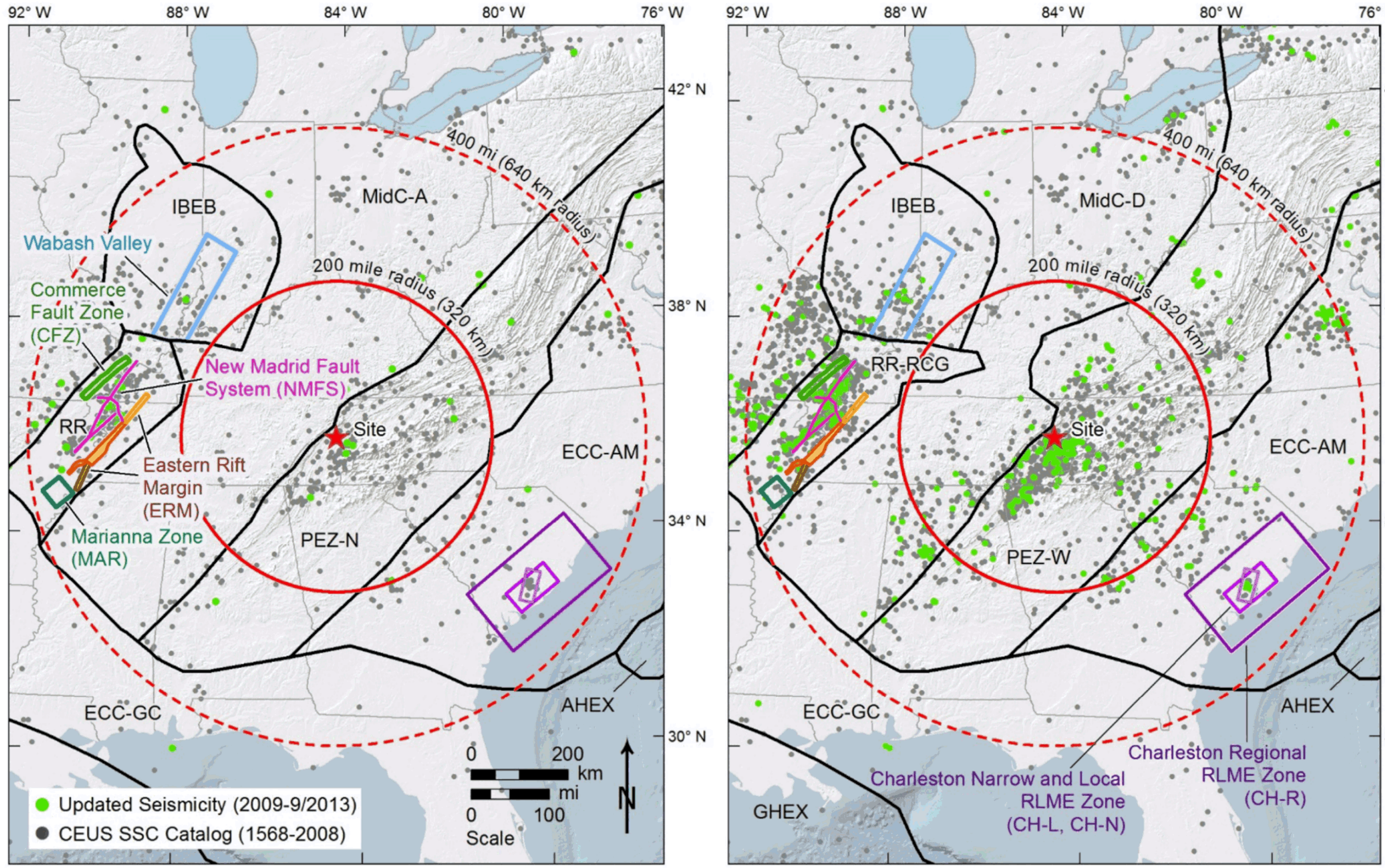
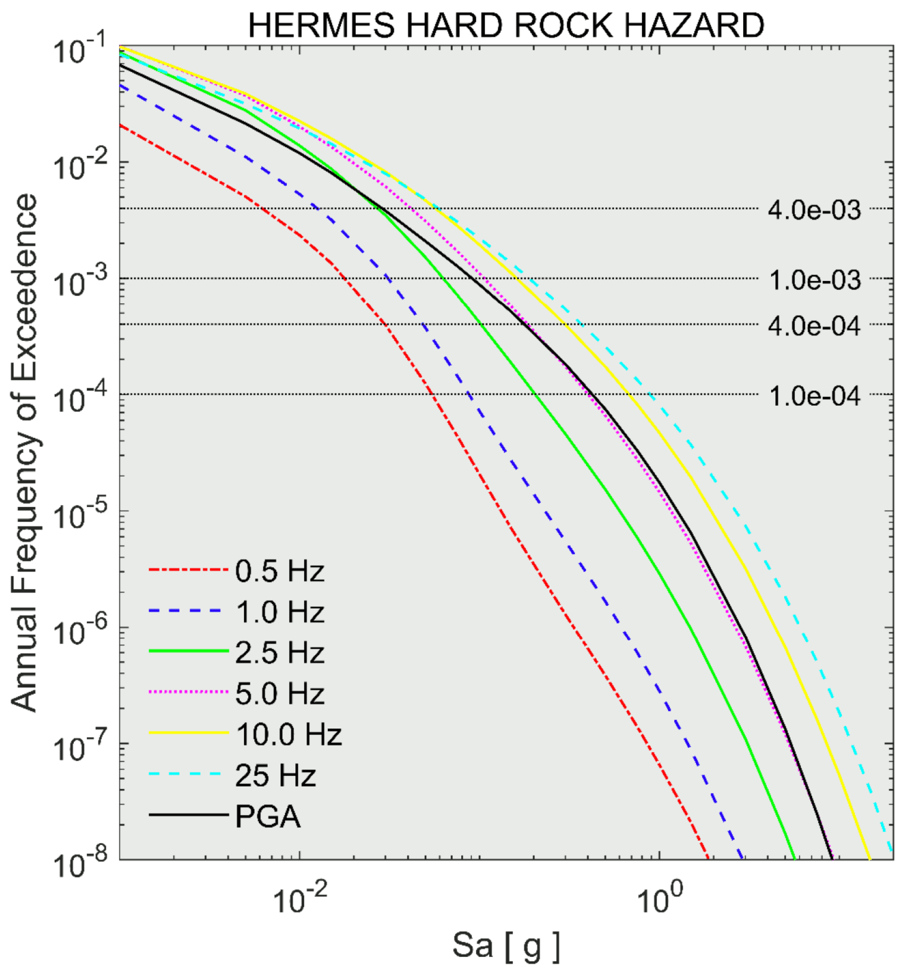


Figure 2.5-9: Mean Total Rock Hazard Curves



Where: PGA = Peak Ground Acceleration

Reference 1

Figure 2.5-10: Uniform Hazard Response Spectrum for Hard Rock Conditions (Log and Semi-log)

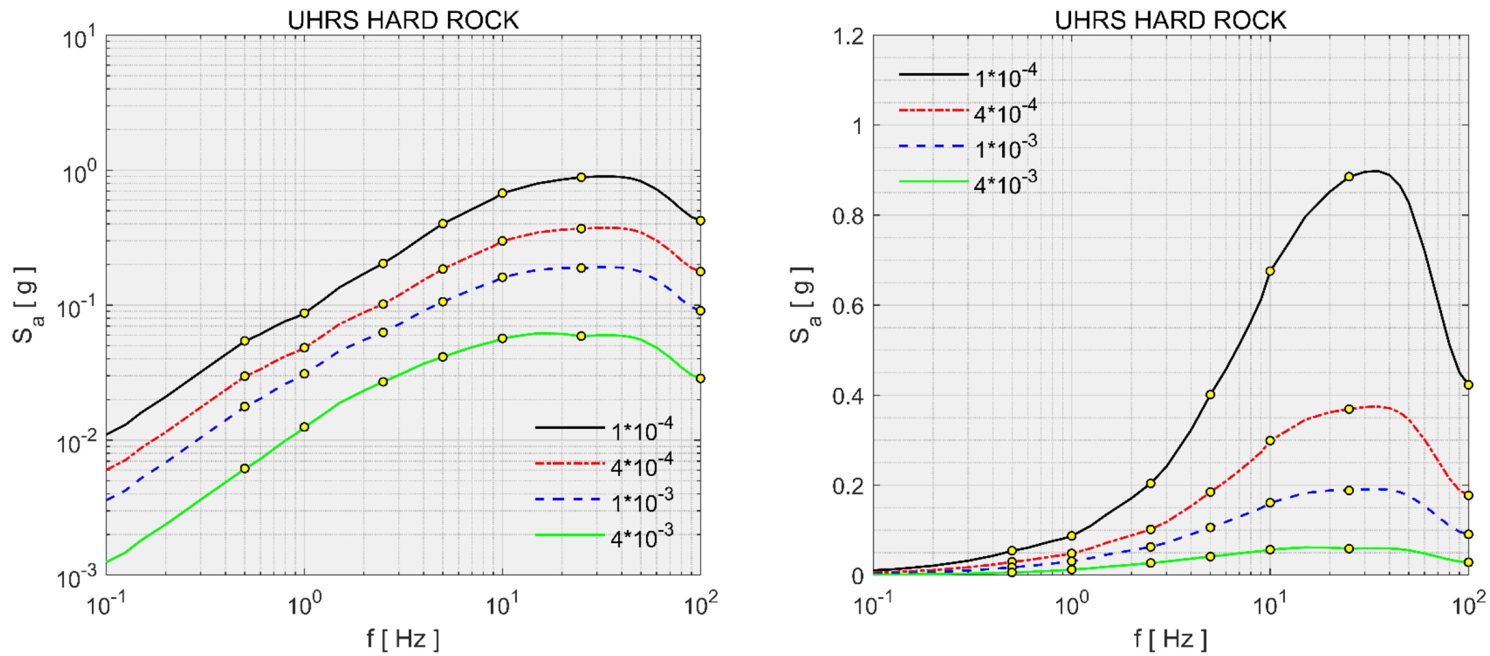


Figure 2.5-11: Location of the Facility at K-33

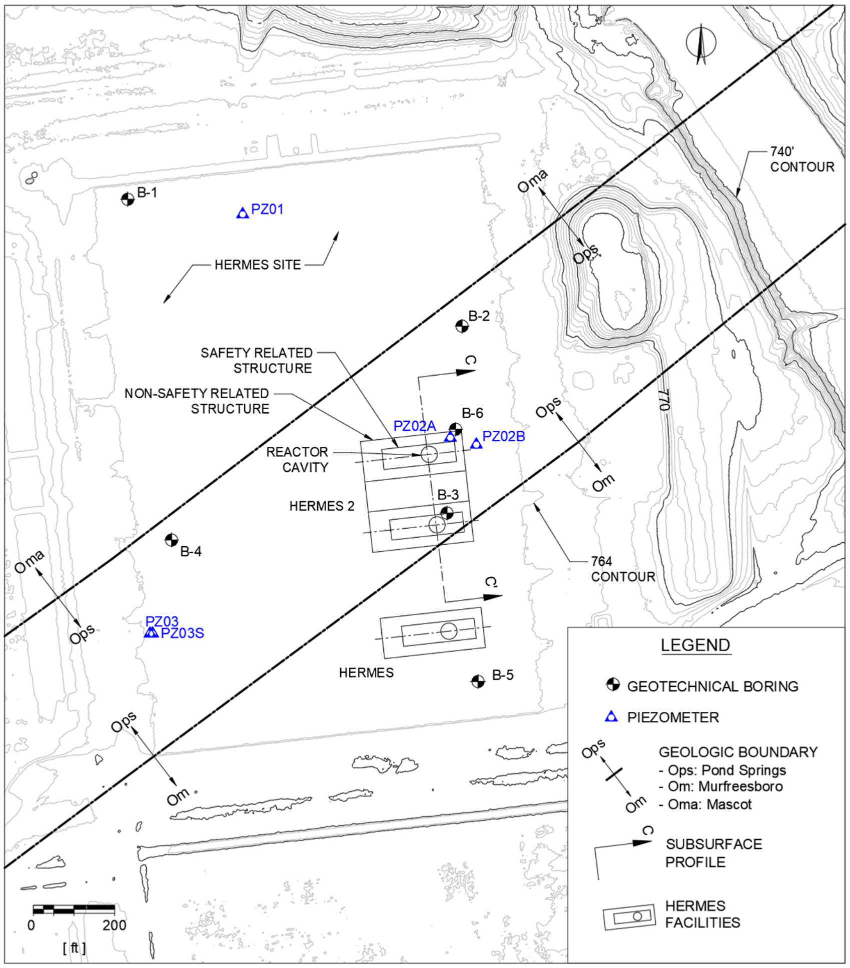
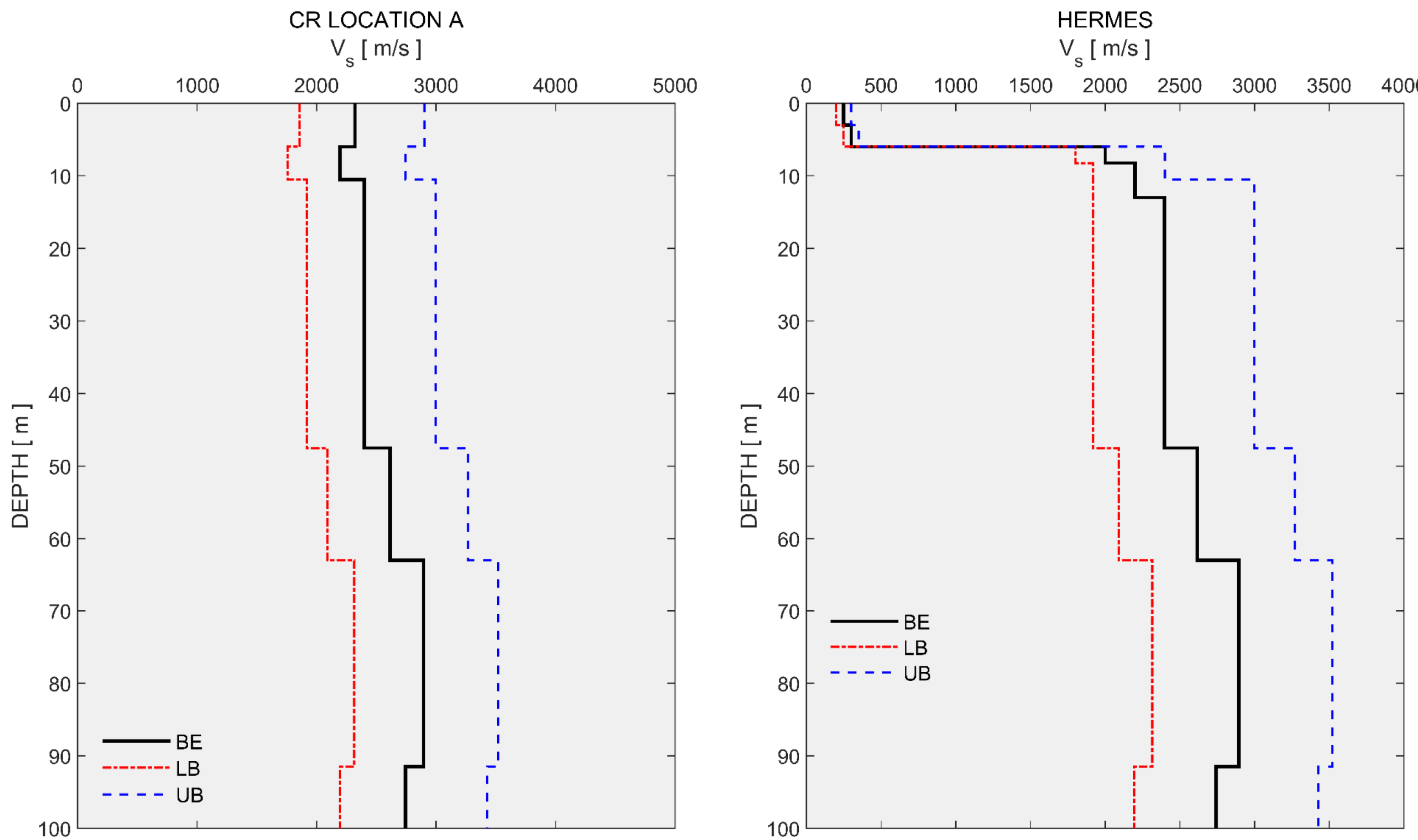


Figure 2.5-12: Hermes and CRN Location A Shear Wave Velocity Profiles



Where BE = Best Estimate, LB = Lower Bound, UB = Upper Bound

Figure 2.5-13: Amplification Ratio between Hard Rock and Location A

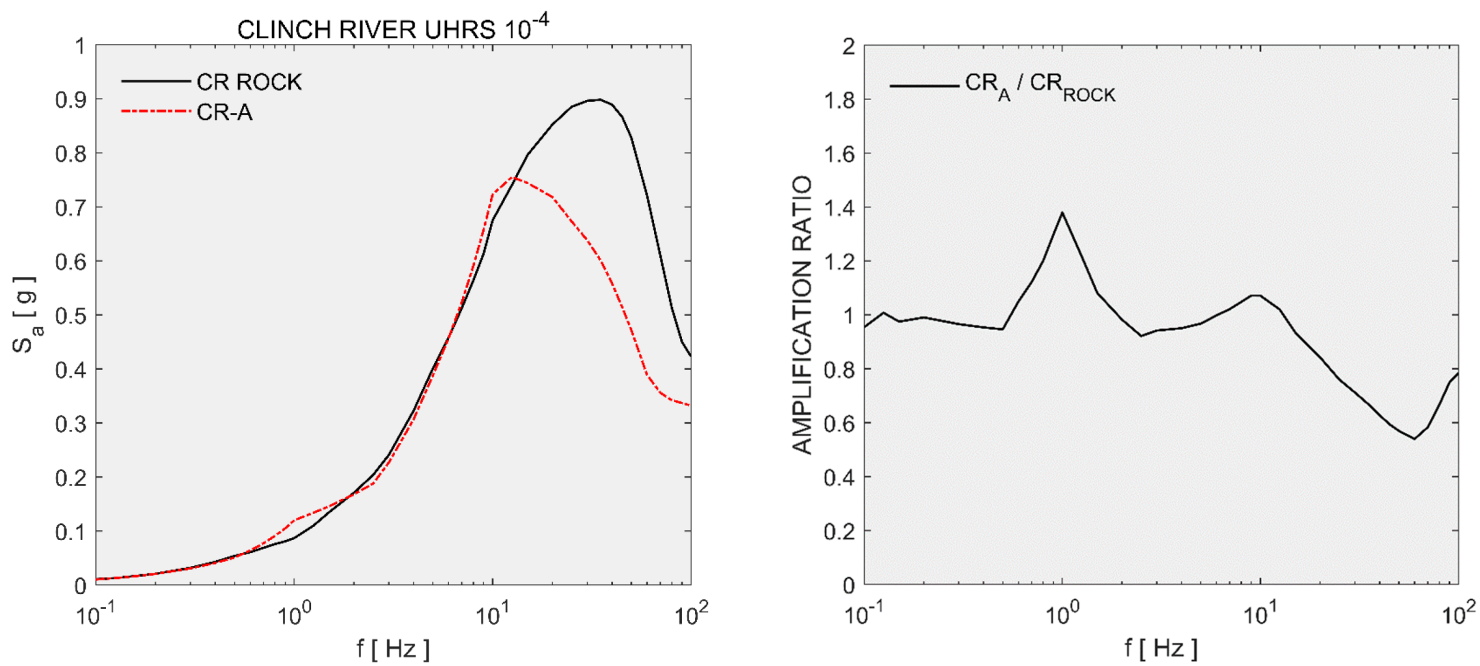


Figure 2.5-14: UHRS at Hermes

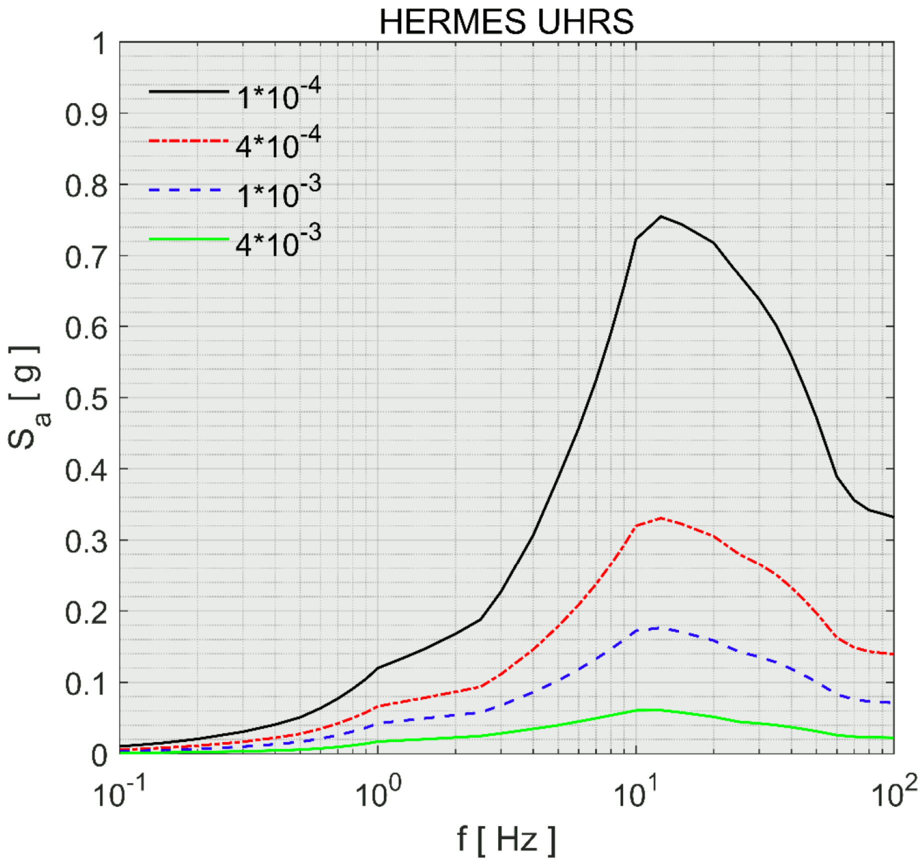


Figure 2.5-15: Comparison of Hermes UHRS to USGS NSHMP

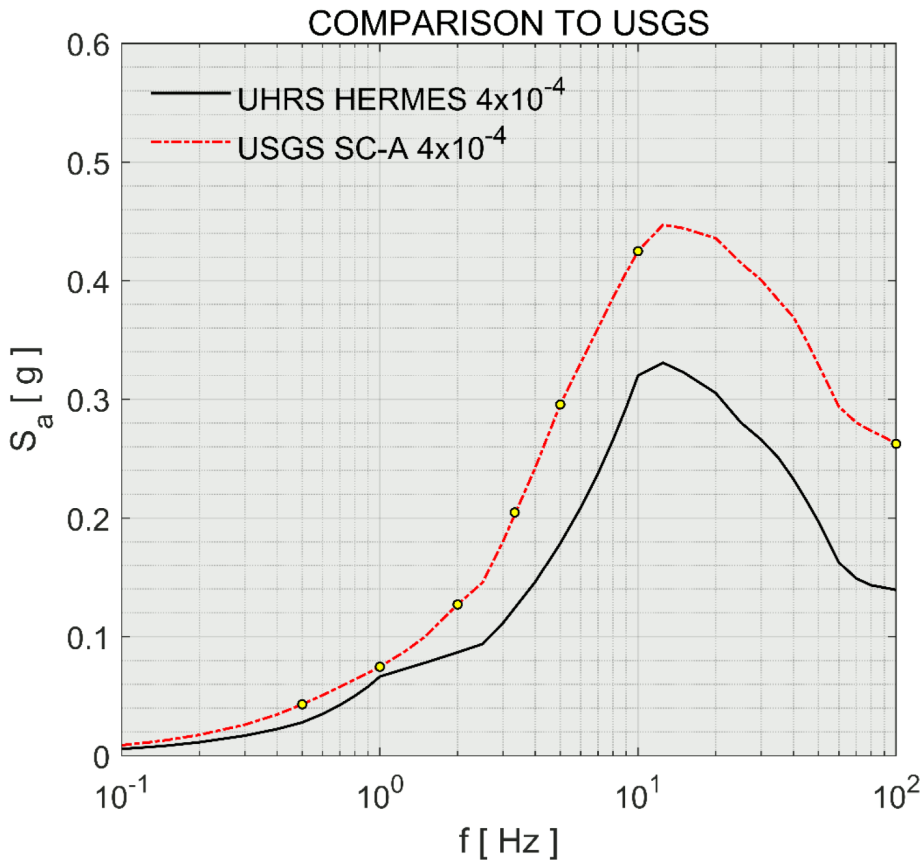


Figure 2.5-16: Hermes Seismic Design Response Spectra

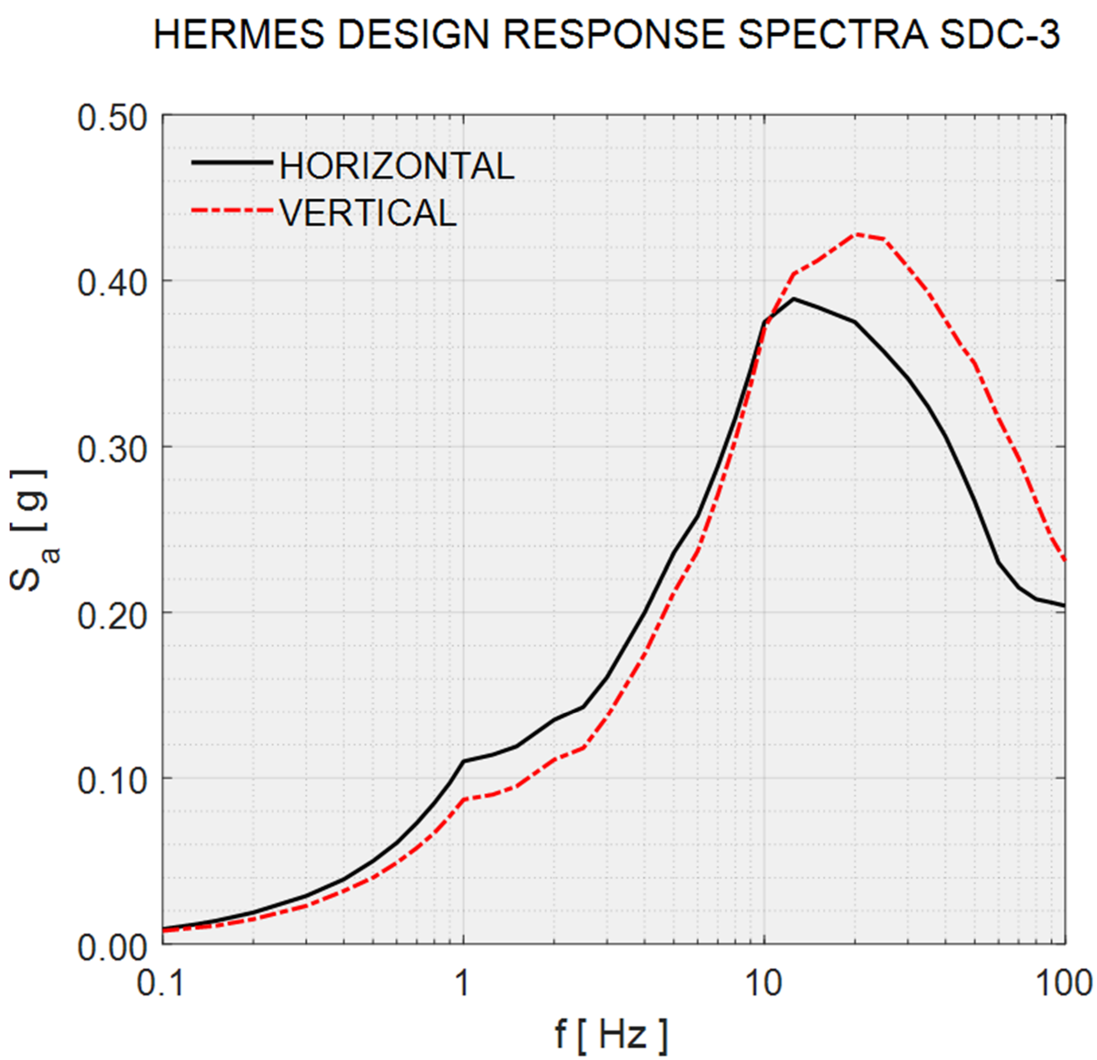


Figure 2.5-17: Original Site Topography

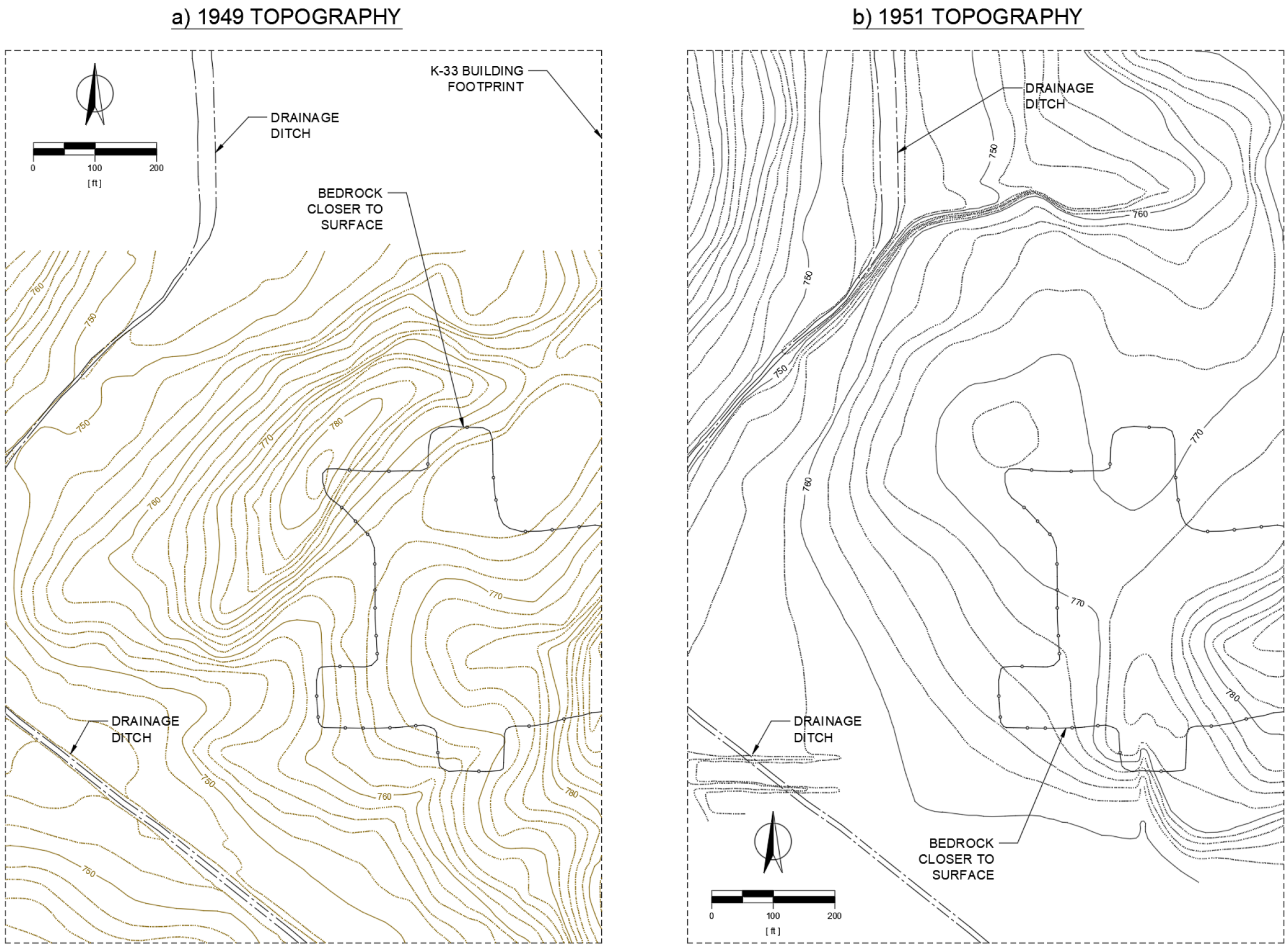


Figure 2.5-18: Original K-33 Building

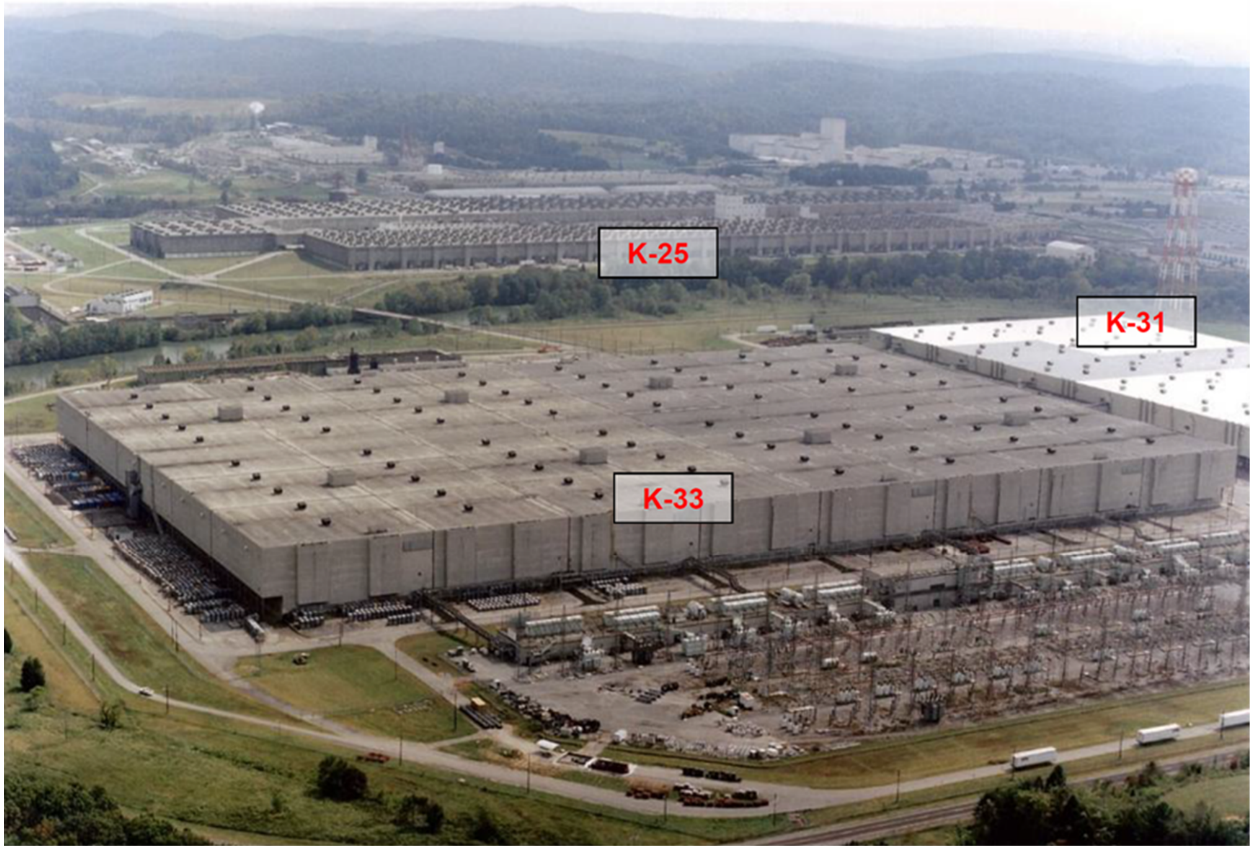


Figure 2.5-19: North to South View of Site (Present Day)

Figure 2.5-20: K-33 Foundation Plan (North)

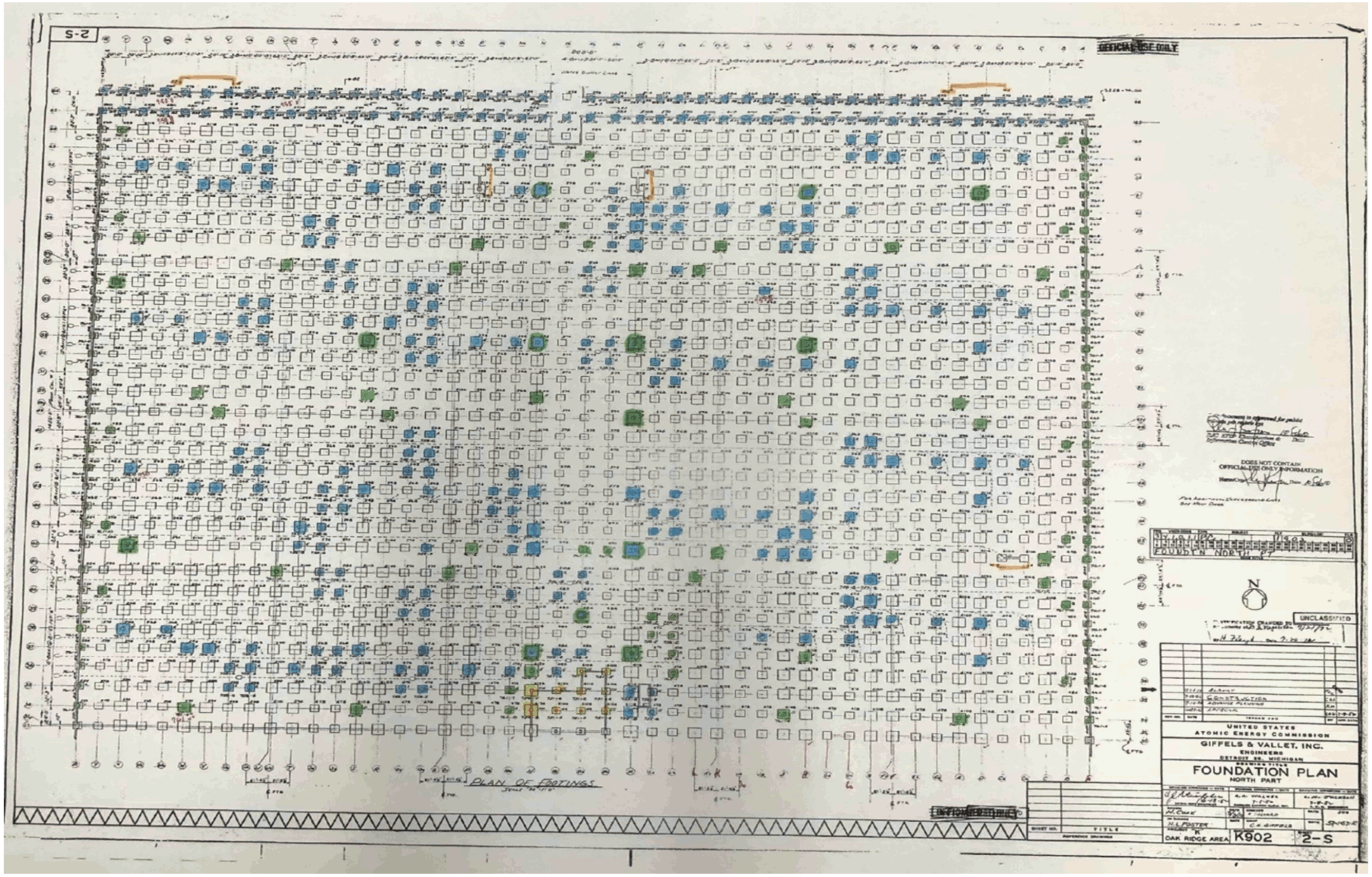
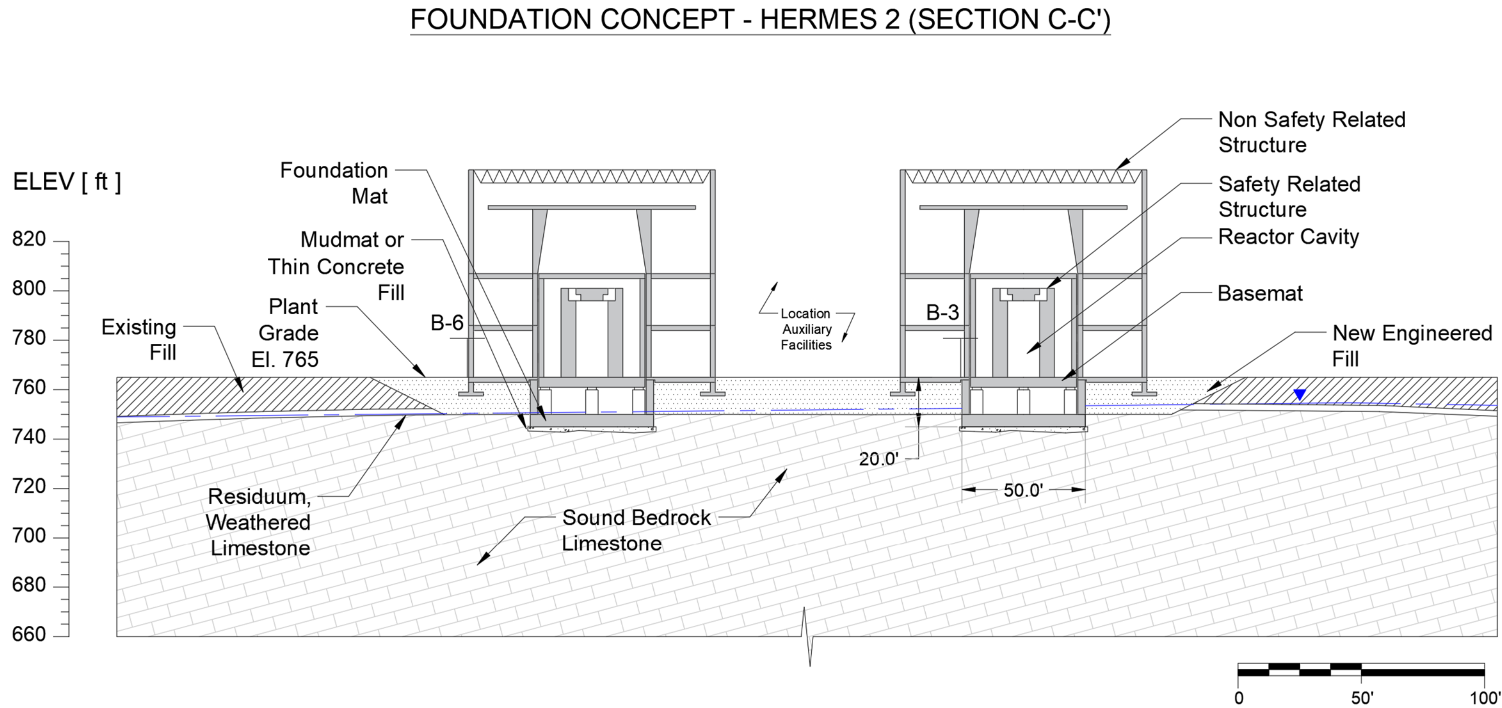


Figure 2.5-21: Abandoned K-33 Footings

OT-2, DEPTH < 4 ft

OT-6, DEPTH 8 ft

Figure 2.5-22: Foundation Interface



NOTES:
- Building and foundation interface representation is schematic and shown to illustrate foundation concept; the building design accounts for applicable environmental and foundation loads including lateral earth pressure.
- Groundwater level shown is observed from borings at time of drilling.

Figure 2.5-23: Profile A-A' (Boring Data Summary)

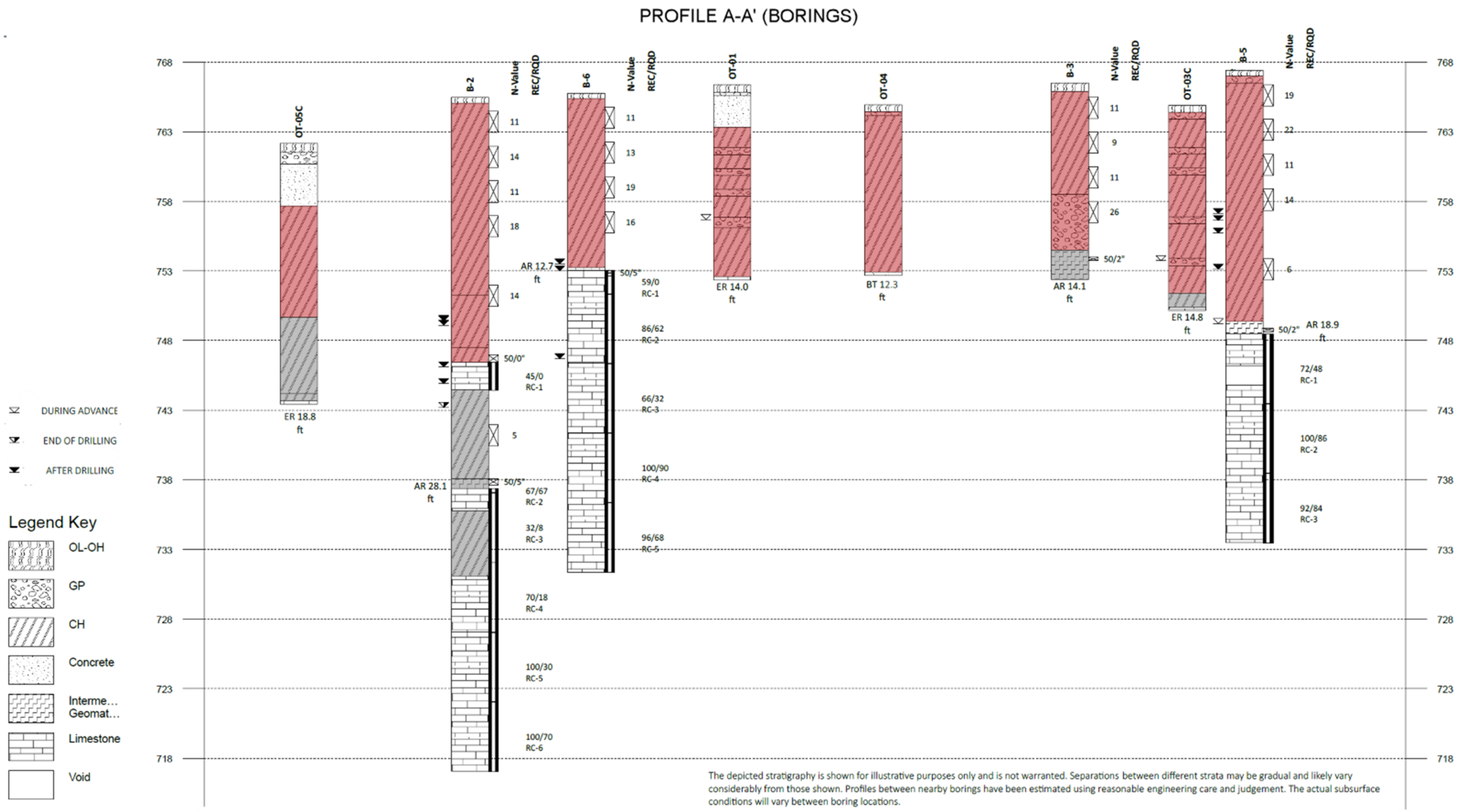
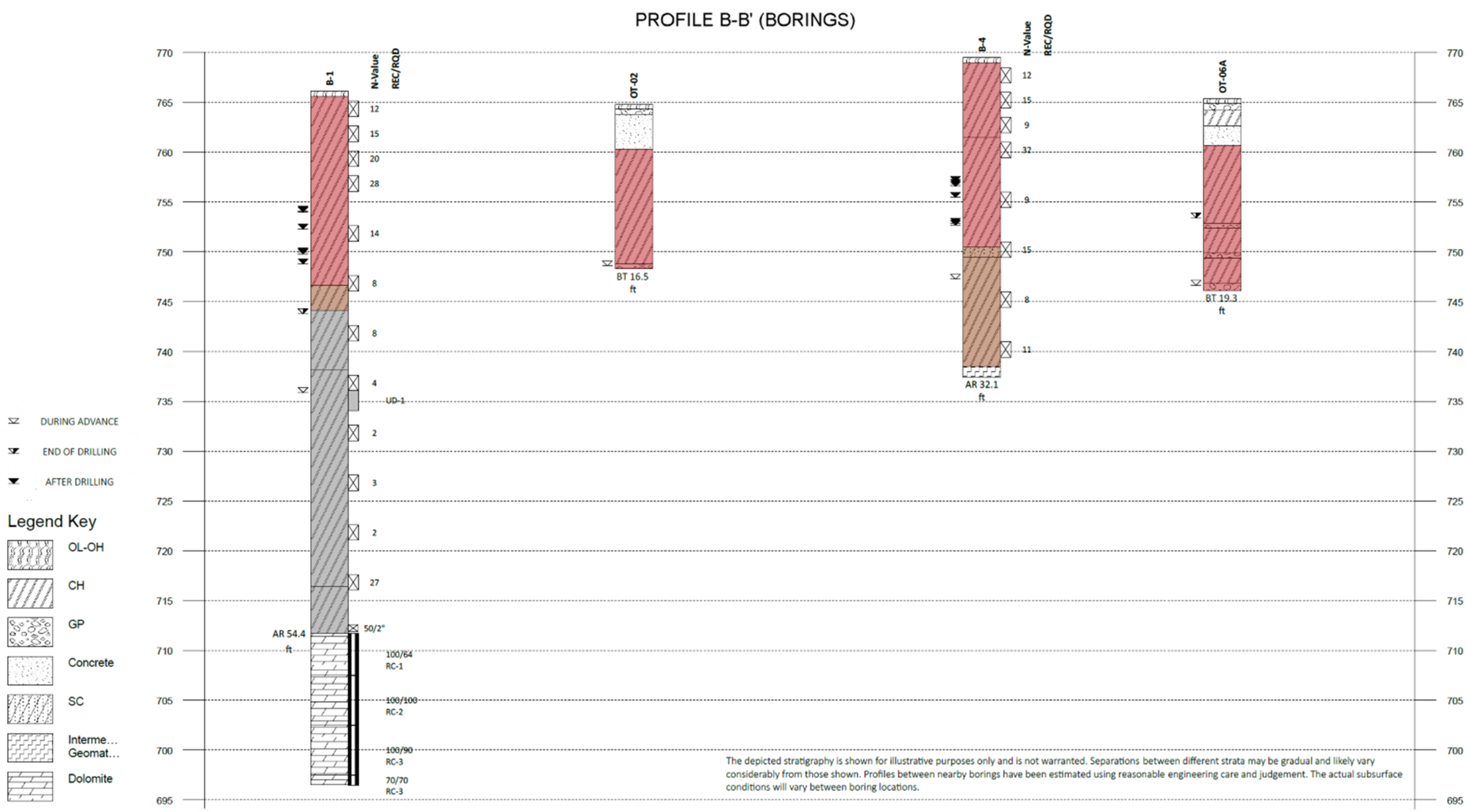


Figure 2.5-24: Profile B-B' (Boring Data Summary)



Kairos Power

Chapter 3

Design of Structures, Systems, and Components

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 INTRODUCTION

This chapter identifies and describes the principal architectural and engineering design criteria for the structures, systems, and components (SSC) that are required to ensure reactor facility safety and protection of the public. The primary safety feature of the design is the unique combination of TRISO fuel and Flibe reactor coolant. Other safety-related systems support maintaining the fuel and coolant configuration within acceptable limits. These SSCs include the safety-related portion of the Reactor Building structure, the reactor vessel and internals, the reactor control and shutdown system, and the decay heat removal system.

3.1.1 Design Criteria

Kairos Power is pursuing a construction permit and subsequent operating license under 10 CFR 50. The NRC regulations in Title 10 to the CFR have been evaluated for applicability to this facility and the results are contained in the “Regulatory Analysis for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor” topical report (Reference 1). The design related regulations that are addressed by this preliminary safety evaluation report (PSAR) are summarized in Table 3.1-1 and addressed throughout this safety analysis report.

Kairos Power has also developed a set of principal design criteria (PDC) applicable for the KP-FHR technology which has been reviewed and approved by the NRC in “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor” (Reference 2). The application of these criteria to the SSCs of the test reactor are shown in Table 3.1-2. Note that while the facility contains two reactor units, no safety-related SSCs are shared between the reactor units, which satisfies PDC 5. Therefore, PDC 5 is not further discussed within this safety analysis report. Specific details regarding how the PDCs are met by the design are described in the individual sections throughout this safety analysis report as summarized in Table 3.1-2.

Note that several of the PDCs in KP-TR-003 contain the terms “safety significant,” “anticipated operational occurrences,” and “accidents.” These terms are not applicable to the Hermes 2 reactors and are not used in this safety analysis report, which represents a departure from the approved topical report. These terms are relevant to power reactors, which use frequency to bin postulated events. In the non-power reactor licensing framework, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors” (NUREG-1537), the postulated events in the design basis are treated the same, regardless of frequency. Consistent with 10 CFR 50.2 (as modified – See Section 1.2.3), SSCs that are relied upon to mitigate the postulated events are classified as safety-related and a significance determination is not made in this framework. There are only two SSC classifications used in this safety analysis report : safety-related and non-safety related. PDCs 1, 2, 3, 4, 5, 13, 14, 15, 16, 17, 18, 20, 28, 30, 31, 32, 33, 34, 44, 61, 71, 73, 75, and 76 use the term “safety significant.” For these PDCs, the term “safety significant” is replaced in this safety analysis report with “safety-related.” Additionally, PDCs 10, 13, 15, 17, 20, 26, 29, 34, 60, 64, and 73 use the term “Anticipated Operational Occurrences.” Since there is no distinction between AOOs and accidents in the non-power reactor licensing framework (NUREG-1537), the AOO terminology (including language that differentiates between AOOs and accidents) is replaced by “postulated events” in this safety analysis report . PDCs 2, 4, 5, 13, 16, 17, 19, 20, 22, 26, 28, 31, 35, 37, 44, 46, 61, 64, 73, and 75 use the term “accidents,” and in these instances “accident” is replaced with “postulated events” in this safety analysis report.

Note that a departure from the 10 CFR 50.2 definition of safety-related is discussed in Section 1.2.3 with respect to the replacement of the words: “integrity of the reactor coolant pressure boundary” with “integrity of the portions of the reactor coolant boundary relied upon to maintain coolant level above the active core.” However, as discussed above, the term “safety-significant” does not apply to Hermes 2. The safety-related portions of the reactor coolant boundary for the reactor are limited to portions of the reactor vessel (see Section 4.3). Failures of other SSCs containing reactor coolant (e.g., pipe breaks within the reactor coolant boundary) do not result in unacceptable consequences as described in Section 13.1.3. A failure of the reactor vessel is a beyond the design basis event as the vessel is designed against such failure consistent with PDC 14. Thus, the makeup inventory of reactor coolant to the reactor vessel is not relied on to mitigate the consequences of a postulated event and the requirements of PDC 33 have been addressed.

3.1.2 NRC Guidance Documents

The NRC guidance documents considered in the design of the reactor are identified within this safety analysis report and are listed in Table 3.1-3. The sections cited in this table describe the extent of usage of these guidance documents. Note that Division 1 regulatory guides are not applicable to non-power test reactors and are not included in this table. In some cases, portions of the Division 1 regulatory guides were utilized and are identified in sections throughout this safety analysis report. Codes and standards used in the design of the reactor structures, systems, and components that contain radioactivity are provided in Section 3.6. Other codes and standards are also identified throughout the report.

3.1.3 References

1. Kairos Power, LLC, “Regulatory Analysis for the Kairos Power Salt-Cooled, High Temperature Reactor,” KP-TR-004-NP-A. June 2022.

2. Kairos Power, LLC, “Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor,” KP-TR-003-NP-A. June 2020.

Table 3.1-1: Design Related 10 CFR Regulations Applicable to the Design

10 CFR Regulation	Title (or subject of regulation)	SAR Section
20.1406	Minimization of Contamination	5.1, 5.2, 9.1.1, 9.1.2, 9.1.3, 9.1.4, 9.1.5, 9.2, 9.3, 9.7, 9.8, 9.9,11.2
20.1601	Control of Access to High Radiation Areas	11.1.5
20.1602	Control of Access to Very High Radiation Areas	11.1.5
20.1701	Use of process or other engineering controls (containment, decontamination, or ventilation)	4.4, 9.1, 9.3
50.34(a)(5)	Contents of applications; technical information - technical specifications	6.2, 14.1, Table 14.1-1
50.34(a)(8)	Contents of applications; technical information - SSCs requiring further Research & Development	1.3
50.34(g)[1]	Contents of applications; technical information - combustible gas control	Not technically relevant as discussed 3.1.1.
10 CFR 50.36(c)(1)(i)(A)	Technical specifications. Limits	14.1
10 CFR 50.36(c)(1)(ii)(A)	Technical specifications. Limiting safety system settings	14.1
10 CFR 50.36(c)(2)(i)	Technical specifications. LCOs	14.1
10 CFR 50.36(c)(2)(ii)(B,C,D)	Technical specifications. LCOs	14.1
10 CFR 50.36(c)(2)(iii)	Technical specifications. LCOs	14.1
10 CFR 50.36(c)(3-8)	Technical specifications. LCOs	7.3, 7.5
10 CFR 50.44(d)[1]	Combustible gas control for nuclear power reactors - requirements for non-water cooled reactor applicants	Not technically relevant as discussed 3.1.1.
10 CFR 50.64	Limitations on the use of highly enriched uranium (HEU) in domestic non-power reactors	Chapter 18
10 CFR 50 Appendix E II	Emergency planning in the PSAR	Chapter 12 Appendix A
10 CFR 70.24	Criticality accident requirements	9.3
10 CFR 73.67	Licensee fixed site and in-transit requirements for the physical protection	Addressed with application for Part 70 license

10 CFR Regulation	Title (or subject of regulation)	SAR Section
10 CFR 75	Safeguards on Nuclear Material	Dependent on written request from NRC
10 CFR 100.10	Factors to be considered when evaluating sites	2.1
10 CFR 100.11	Determination of exclusion area, low population zone, and population center distance	2.1

Notes:

1. Regulations 10 CFR 50.34(g) and 10 CFR 50.44(d) address the potential for the accumulation of combustible gases within a containment structure following a design basis accident which could ignite and damage a principal fission product barrier (the containment). Combustible gas events are not technically relevant to the reactor design.

Table 3.1-2: Principal Design Criteria

Principal Design Criteria	SAR Section
PDC 1, Quality Standards and Records	3.5, 4.3, 6.3, 7.3, 7.4, 7.5
PDC 2, Design bases for protection against natural phenomena	3.5, 4.2.2, 4.3, 4.7, 5.1, 6.3, 7.3, 7.4, 7.5, 8.2, 8.3, 9.1.1, 9.1.2, 9.1.3, 9.1.4, 9.1.5, 9.2, 9.3, 9.4, 9.7, 9.8.2, 9.8.4, 9.8.5, 11.2
PDC 3, Fire Protection	6.3, 7.3, 7.5, 9.3, 9.4
PDC 4, Environmental and dynamic effects design bases	4.2.2, 4.3, 4.7, 6.3, 7.3, 9.1.1, 9.1.2, 9.1.4, 9.3, 9.7, 9.8.2, 9.8.4, 9.9
PDC 5, Sharing of structures, systems, and components	3.1.1
PDC 10, Reactor Design	4.2.1, 4.3, 4.5, 4.6, 5.1, 6.3, 7.3
PDC 11, Reactor Inherent Protection	4.5
PDC 12, Suppression of reactor power oscillations	4.5, 4.6, 5.1
PDC 13, Instrumentation and Control	7.2, 7.3, 7.5, 9.1.3
PDC 14, Reactor Coolant Boundary	4.3
PDC 15, Reactor coolant system design	7.3
PDC 16, Containment design	4.2.1, 5.1
PDC 17, Electric Power systems	8.2, 8.3
PDC 18, Inspection and testing of electric power systems	8.2, 8.3
PDC 19, Control room	7.4
PDC 20, Protection system functions	7.3
PDC 21, Protection system reliability and testability	7.3, 7.5
PDC 22, Protection System Independence	7.3, 7.5
PDC 23, Protection system failure modes	4.2.2, 7.3
PDC 24, Separation of protection and control systems	7.3, 7.5
PDC 25, Protection system requirements for reactivity control malfunctions	7.3

Principal Design Criteria	SAR Section
PDC 26, Reactivity control systems	4.2.2, 4.5
PDC 28, Reactivity limits	4.2.2, 7.3
PDC 29, Protection against anticipated operation occurrences	4.2.2, 7.3, 7.5
PDC 30, Quality of reactor coolant boundary	4.3
PDC 31, Fracture prevention of reactor coolant boundary	4.3
PDC 32, Inspection of reactor coolant boundary	4.3
PDC 33, Reactor coolant inventory maintenance	4.3, 5.1, 9.1.4, 9.3
PDC 34, Residual heat removal	4.3, 4.6, 6.3
PDC 35, Passive residual heat removal	4.3, 4.6, 6.3
PDC 36, Inspection of passive residual heat removal system	4.3, 6.3
PDC 37, Testing of passive residual heat removal system	4.3, 6.3
PDC 44, Structural and equipment cooling	9.1.5, 9.7
PDC 45, Inspection of structural and equipment cooling systems	9.1.5, 9.7
PDC 46, Testing of structural and equipment cooling systems	9.1.5, 9.7
PDC 60, Control of releases of radioactive materials to the environment	5.1, 5.2, 9.1.3, 9.2, 9.9.1, 9.9.3, 11.2
PDC 61, Fuel storage and handling and radioactivity control	9.3
PDC 62, Prevention of criticality in fuel storage and handling	9.3
PDC 63, Monitoring fuel and waste storage	9.3, 11.2
PDC 64, Monitoring radioactivity releases	9.1.2, 9.1.3, 9.2, 9.9.1, 9.9.3
PDC 70, Reactor coolant purity control	5.1, 9.1.1, 9.1.4
PDC 71, Reactor coolant heating systems	9.1.5
PDC 73, Reactor coolant system interfaces	5.2

Principal Design Criteria	SAR Section
PDC 74, Reactor vessel and reactor system structural design basis	4.3, 4.7
PDC 75, Reactor building design basis	3.5
PDC 76, Provisions for periodic Reactor Building inspection	3.5

Table 3.1-3: NRC Guidance Considered in the Design

NRC Guidance	Title	SAR Section
Regulatory Guide 2.2	Development of Technical Specifications for Experiments in Research Reactors	12
Regulatory Guide 2.5	Quality Assurance Program Requirements for Research and Test Reactors	3.5, 3.6, 12.9
Regulatory Guide 2.6	Emergency Planning for Research and Test Reactors	12.7
Regulatory Guide 4.1	Radiological Environmental Monitoring for Nuclear Power Plants	11.1
Regulatory Guide 4.7	General Site Suitability Criteria for Nuclear Power Stations	2.2
Regulatory Guide 4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other than Power Reactors	11.1
Regulatory Guide 4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning	11.1
Regulatory Guide 5.59	Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Safety Significance	12.8
Regulatory Guide 8.2	Administrative Practices in Radiation Surveys and Monitoring	11.1
Regulatory Guide 8.4	Personnel Monitoring Device-Direct-Reading Pocket Dosimeters	11.1
Regulatory Guide 8.7	Instructions for Recording and Reporting Occupational Radiation Exposure Data	11.1
Regulatory Guide 8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	11.1
Regulatory Guide 8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable	11.1
Regulatory Guide 8.13	Instruction Concerning Prenatal Radiation Exposure	11.1
Regulatory Guide 8.25	Air Sampling in the Workplace	11.1

NRC Guidance	Title	SAR Section
Regulatory Guide 8.29	Instruction Concerning Risks from Occupational Radiation Exposure	11.1
Regulatory Guide 8.34	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses	11.1

3.2 METEOROLOGICAL DAMAGE

This section describes the approach used to translate design basis meteorological parameters into loads used in the design of safety-related SSCs. The design basis meteorological parameters are consistent with the findings of the site characterization analysis in described in Section 2.3. The design basis meteorological parameters applicable to the design include normal wind loads, high wind loads from tornados and hurricanes, and precipitation loads. The treatment of these loads is discussed in the following subsections.

3.2.1 Normal Wind Loads

The meteorological characterization of the facility site defined the normal and high wind characteristics for the facility site (See Section 2.3). This section describes the approach to translating the normal winds for the site into loads on the safety-related portion of the Reactor Building.

The safety-related SSCs for the reactor are located within the safety-related portion of the Reactor Building, which is discussed further in Section 3.5. The design of the safety-related portion of the Reactor Building provides protection for safety-related SSCs against adverse effects from winds. The design basis normal wind loading conditions are discussed in the following subsections.

Wind loads affect both the main wind-force resisting system (MWFRS) and components and cladding (C&C). The MWFRS is assembled of the structural elements that provide support and stability for the overall structure. The C&C are elements of the building envelope that do not qualify as the MWFRS.

3.2.1.1 Applicable Design Parameters

Local building code for the facility references ASCE/SEI 7-10, “Minimum Design Loads for Buildings and Other Structures” (Reference 1). ASCE/SEI 7-10 defines risk categories for structures and provides design basis normal wind velocities for each risk category. Risk Category IV is the most stringent Risk Category in ASCE/SEI 7-10 and is selected as the design basis for the safety-related portions of the Reactor Building because it is consistent with the standard to categorize the materials in the facility as “hazardous substances.” Using Figure 26.5-1B from ASCE/SEI 7-10, the safety-related portion of the Reactor Building structure is designed to withstand a basic wind velocity of 120 miles per hour (mph) for Risk Category IV structures. These wind velocities bound the expected velocities for the facility site in Oak Ridge, Tennessee.

For the design of the MWFRS, the wind speed is transformed to equivalent pressure consistent with ASCE/SEI 7-10, Section 27.3. For the design of C&C, the wind speed is transformed to equivalent pressure consistent with ASCE/SEI 7-10, Section 29.3 and Section 30.3, respectively.

The mean recurrence interval of the basic wind speed for Risk Category IV buildings is 1,700 years. Wind loads determined in accordance with the mean recurrence interval from ASCE/SEI 7-10, Chapters 26 to 30, for a Risk Category IV building are more stringent than the 100-year return period wind speed (see Section 2.3). As shown in Figure 2.5-18, the site is open terrain with scattered obstructions having heights generally less than 30 ft. Based on those site characteristics, it is consistent with ASCE/SEI 7-10 to use exposure category “C.” The exposure category is used to determine inputs for the computation of applied forces on structures as discussed in 3.2.1.2 and 3.2.1.3.

3.2.1.2 Determination of Applied Forces

In accordance with Equation 27.3-1 of ASCE/SEI 7-10 the velocity pressure is provided in Equation 3.2-1.

$$Q_z = 0.00256 K_z K_{zt} K_d V^2 (\text{lb/ft}^2) \qquad \text{(Equation 3.2-1)}$$

Where,

q_z = velocity pressure at height (z)

K_z = velocity pressure exposure coefficient at height (z) as determined by Table 27.3-1 of ASCE/SEI 7-10 that corresponds to the height of the safety-related structure

K_{zt} = topographic factor as determined by Section 26.8-2 of ASCE/SEI 7-10 equal to 1.0

K_d = wind directionality factor as determined by Figure 26.6-1 of ASCE/SEI 7-10 equal to 0.85 for the MWFRS and C&C

V = basic wind speed (3 second gust) as determined by Figure 26.5-1B of ASCE/SEI 7-10 for Category IV Buildings and Other Structures equal to 120 mph

3.2.1.3 Application of Normal Wind Load to Design of Structures

The calculated velocity pressure as determined in Section 3.2.1.2 is applied in accordance with ASCE/SEI 7-10 to design the safety-related portions of the Reactor Building to provide protection of safety-related SSCs against the effects of normal wind loads. See Section 3.5 for further discussion of design features that address loads on the safety-related portions of the Reactor Building from natural phenomena.

3.2.2 Tornado Loading

The meteorological characterization of the facility site defined the normal and high wind characteristics for the facility site (See Section 2.3). This section describes the approach to translating the characteristics of design basis tornados for the site into loads on the safety-related portion of the Reactor Building. Tornado characteristics include high wind speed, atmospheric pressure change, and tornado generated missile impacts. The design basis tornado loading conditions are discussed in the following subsections.

3.2.2.1 Applicable Design Parameters

Guidance from Regulatory Guide (RG) 1.76, Revision 1, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” was used to determine characteristics of the design-basis tornado to be applied to the safety-related portions of the facility design. Those design basis characteristics are listed in Table 1 of RG 1.76. Based on the facility location, the parameters for Region I are applicable. The design basis tornado missile spectrum and maximum horizontal speeds are also provided in Table 2 of RG 1.76 for Region I.

RG 1.76 provides wind speeds for the facility location but does not provide a method to determine applied forces from tornadoes. NUREG-1537 also does not provide a method. Although not applicable to non-power reactor facilities, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Section 3.3.2, Revision 3, “Tornado Loadings,” references ASCE/SEI 7, which does provide a method to determine applied forces from tornadoes. Since ASCE/SEI 7-10 is the code of record for the facility location’s local building code, the method from ASCE/SEI 7-10 is used to determine the applied forces on the safety-related portions of the Reactor Building from tornadoes, using the wind speeds from RG 1.76.

3.2.2.2 Determination of Applied Forces

In accordance with Equation 27.3-1 of ASCE/SEI 7-10 the velocity pressure, or design basis high wind speed, is determined using in Equation 3.2-1

Where,

K_d = wind directionality factor equal to 1.0

V = maximum tornado wind speed as determined by RG 1.76, Revision 1, equal to 230 mph

The design basis atmospheric pressure change, or tornado differential pressure, is 1.2 pounds per square inch (psi) as determined by Table 1 of RG 1.76.

Finally, the procedure used for transforming the tornado-generated missile impact into an effective or equivalent static load on the safety-related portions of the structure is consistent with NUREG-0800, Section 3.5.3, Subsection II. Tornado-generated missile impact effects are based on the design missile spectrum from RG 1.76.

3.2.3 Hurricane Loading

The meteorological characterization of the facility site defined the normal and high wind characteristics for the facility site (See Section 2.3). This section describes the approach to translating the characteristics of design basis hurricanes for the site into loads on the safety-related portion of the Reactor Building. Hurricane characteristics include high wind speed and hurricane-generated missile impacts. The design basis hurricane loading conditions are discussed in the following subsections.

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 to determine applicable design parameters for hurricane loads on safety-related portion of the Reactor Building. RG 1.221 provides wind speeds for the facility location that are consistent with the definitions used in ASCE/SEI 7-10. Since ASCE/SEI 7-10 is the code of record for the facility location's local building code, the method from ASCE/SEI 7-10 is used to determine the applied forces from hurricanes, using the wind speeds from RG 1.221.

3.2.3.2 Determination of Applied Forces

The maximum hurricane wind speed, V, is 130 mph, consistent with the guidance in RG 1.221 for the site location. Velocity pressure is determined using the maximum hurricane wind speed and the guidance of RG 1.221 for peak gust wind speed in Equation 3.2-1 (see Section 3.2.1) from ASCE/SEI 7-10. The procedure used for transforming the hurricane-generated missile impact into an effective or equivalent static load on the safety-related portions of the structure is consistent with NUREG-0800, Section 3.5.3, Subsection II. Hurricane-generated missile impact effects are based on the design missile spectrum from RG 1.221.

3.2.4 Precipitation Loads

The meteorological characterization of the facility site defined the precipitation characteristics for the facility site (See Section 2.3). This section describes the approach to translating the characteristics of design basis precipitation for the site into loads on the safety-related portion of the Reactor Building. Precipitation categories include rain, snow, and ice. Grading and drainage on the site preclude loads from precipitation accumulation on the ground affecting the safety-related portion of the Reactor Building. Design features of the site to address precipitation accumulation are discussed in Section 3.5. The non-safety related exterior shell of the Reactor Building has a sloped roof, therefore, loads due to rain accumulation are not considered as a structural load in the structural design. Similarly, as a result of the lack of rain accumulation, load due to ice is anticipated to be minimal and is therefore enveloped by the snow load. The design basis precipitation loading conditions are discussed in the following subsections.

3.2.4.1 Applicable Design Parameters

Based on Risk Category IV characterization (See Section 3.2.1.1) and site location, Chapters 1 and 7 of ASCE/SEI 7-10 provide snow load design parameters to be applied to the safety-related portions of the Reactor Building.

3.2.4.2 Determination of Applied Forces

The sloped roof (balanced) snow load is calculated by Equation 3.2-3 as derived from ASCE/SEI 7-10, Section 7.3 and Section 7.4 using the ground snow load specified in Section 2.3.1.11.

$$p_s = 0.7 C_s C_e C_t I_s p_g \quad \text{(Equation 3.2-2)}$$

Where,

C_s = roof slope factor as determined by Sections 7.4.1 through Section 7.4.4 of ASCE/SEI 7-10 corresponding to the geometry of the roof

C_e = exposure factor as determined by Table 7-2 of ASCE/SEI 7-10 equal to 1.0

C_t = thermal factor as determined by Table 7-3 of ASCE/SEI 7-10 equal to 1.0

I_s = importance factor as determined by Table 1.5-1 of ASCE/SEI 7-10 and 1.5-2 of ASCE/SEI 7-10 equal to 1.2

p_g = ground snow load consistent with Section 2.3.1.11 equal to 21.9 psf

Unbalanced snow loads on the ceiling of the safety-related portion of the Reactor Building are determined in accordance with Section 7.6 of ASCE/SEI 7-10. The design snow drift loads are determined in accordance with Section 7.7 of ASCE/SEI 7-10. If applicable to the roof design, rain-on-snow surcharge loads are determined in accordance with Section 7.10 of ASCE/SEI 7-10.

3.2.5 References

1. American Society of Civil Engineers, Seismic Engineering Institute, ASCE/SEI 7-10, “Minimum Design Loads for Buildings and Other Structures.” 2010.

3.3 WATER DAMAGE

This section describes the approach to establishing loads on the safety-related portion of the Reactor Building from internal and external flooding postulated events.

3.3.1 Internal Flooding

Internal flooding postulated events consider the flow rates and quantities of water from sources inside the safety-related portions of the Reactor Building. Section 3.5.3.2 describes design features that prevent internal flooding from affecting a safety-related SSC's ability to perform its safety function.

3.3.2 External Flooding Events

The hydrologic evaluation of the site described in Section 2.4 found that the flood elevation for the site does not exceed grade elevation at an annual frequency of 4E-05. Therefore, grade elevation is used as the design basis flood elevation and external floods do not result in loads on the safety-related portion of the Reactor Building above grade. In the design basis flood event, the portion of the safety-related structure that is below grade could be subjected to hydrological loads. Section 3.5.3.2 discusses how hydrological loads are evaluated in the design.

3.3.3 References

None

3.4 SEISMIC DAMAGE

This section discusses the design and design bases of SSCs that are required to maintain function in the event of an earthquake at the facility. The facility is designed such that there is reasonable assurance that a potential design basis earthquake will not preclude the reactor from shutting down and being maintained in a safe shutdown condition. The consequences of a potential design basis earthquake would be within the dose limits defined in Chapter 13 and are therefore bounded by the maximum hypothetical accident analysis presented in Chapter 13. As discussed in Chapter 13, the requirements in 10 CFR 100 are used to define the dose limit commitments for safe performance of the facility in a design basis earthquake.

A graded performance approach outlined in ASCE 43-19, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities" (Reference 1), is used to design the safety-related SSCs in the facility to protect against seismic damage from the design basis earthquake. As stated in the introduction of ASCE 43-19, "The intent [of this Standard] is to control the design process such that the performance of the SSC related to safety and environmental protection is acceptable." Safety-related SSCs designed to this standard provide reasonable assurance that the reactor can be shut down and maintained in a safe condition. The performance gradations in ASCE 43-19 are based on the radiological hazards of the facility and the specific safety functions of the SSC.

SSCs are designated based on their safety classification. The safety-related SSCs are designed to Seismic Design Category (SDC) 3 consistent with ASCE 43-19, because they are required to maintain their safety function in the event of a design basis earthquake. SSCs that are non-safety related are designed to local building code, the 2012 International Building Code (IBC, Reference 2), which is consistent with NUREG-1537.

Use of a performance-based approach for graded classification of SSCs is consistent with the guidance from NUREG-1537, including IAEA-TECDOC-403 (Reference 4) and IAEA-TECDOC-348 (Reference 7, now effectively superseded by IAEA-TECDOC-1347, Reference 8) referenced therein. That guidance permits the selection of design basis earthquakes and corresponding SSC seismic design criteria based on their relative safety significance. For the facility, the return period associated with design basis ground motion corresponding to ASCE 43-19 SDC-3 is similar to the maximum considered earthquake specified in building codes with 2% probability of exceedance in 50 years, as were considered and approved by NRC for design of other non-power reactor nuclear facilities. Additionally, due to its relatively shorter operating lifetime, the probability of exceeding the design ground motion level over its operating life is less for than other facilities with design basis ground motions with similar return periods.

3.4.1 Seismic Design for Safety-Related SSCs

3.4.1.1 Seismic Design Criteria

The facility is designed to be capable of shutting down and of being maintained in a safe condition or a condition within acceptable limits (see Chapter 13) in the event of a design basis earthquake. Acceptable seismic performance of safety-related SSCs is defined based on the selected ASCE 43-19 limit state, which is informed by the performance limits or functional safety requirements of the SSC. That is, in the event of a design basis earthquake, SSCs are designed to perform their required safety functions that are credited in the postulated event analyses of Chapter 13. Acceptance criteria are a function of the seismic hazard (ground motion intensity), a design factor, and control of SSC capacity. This design approach defines seismic criteria for credited SSCs using gradation based on limiting dose below specified thresholds.

SSCs used for this purpose are designed to the SDC-3 Design Response Spectra (DRS) to ensure:

- Integrity of the reactor vessel to support the functional containment provided by the pebbles and the Flibe in the core
- Capability to shut down the reactor and maintain it in a safe shutdown condition
- Capability to prevent or mitigate the consequences of postulated events to potential offsite exposures

Acceptable seismic performance criteria to meet this intent are described in American National Standards Institute (ANSI) and American Nuclear Society (ANS) Standard 15.7 (Reference 3). Section 3.2(2) of ANSI/ANS 15.7 states, "reactor safety related structures and systems shall be seismically designed such that any seismic event cannot cause an accident which will lead to dose commitments in excess of those specified in 3.1."

The phrase "any seismic event" from ANSI/ANS 15.7 is defined as the maximum historical intensity earthquake in accordance with the guidance on the design-basis earthquake in Section 3.1.2.1 of International Atomic Energy Agency document IAEA-TECDOC 403. The historical seismicity, as well as probabilistic seismic hazard considerations, are captured in the hazard analysis summarized in Section 2.5. The design basis earthquake ground motion development in Section 2 of ASCE 43-19 is used to develop a DRS appropriate for SDC-3 SSCs based on site seismic hazard.

For SDC-3 SSCs, the DRS determined by Section 2 of ASCE 43-19 is based on a mean annual hazard exceedance frequency, H_P, of 1E-4 reduced by a scale factor informed by the slope of the site seismic hazard. Design provisions of ASCE 43-19 are calibrated to achieve dual criteria: (1) less than about 1% probability of unacceptable performance for a design basis ground motion, and (2) less than about 10% probability of unacceptable performance for 150% of the design basis ground motion. Per Table 1-1 of ASCE 43-19, when the SDC-3 DRS is used with the structural design provisions in ASCE 43-19, SSCs achieve a target performance goal, P_F, of approximately 1E-4.

3.4.1.2 Design Response Motion

Site hazard analysis detailed in Section 2.5 is used to develop the DRS, as described below.

3.4.1.3 Design Response Spectra

Using the developed horizontal and vertical uniform hazard response spectra (UHRS), the 5% damped horizontal and vertical DRS for SDC-3 are determined following Section 2.2 of ASCE 43-19. SSCs designed to this DRS achieve the target seismic performance goals outlined in Section 3.4.1.1. The horizontal and vertical DRS are illustrated in Figure 3.4-1.

3.4.1.4 Seismic Response

Seismic response of the safety-related portion of the Reactor Building subjected to the design ground motion described in section 3.4.1.2 to characterize seismic demands for design of SDC-3 SSCs is determined as summarized in the subsections below.

3.4.1.5 Structural Model

The safety-related portion of the Reactor Building is represented by a three-dimensional finite-element model developed in accordance with Chapter 3 of ASCE 4-16 (Reference 5). The model captures the primary elements of the lateral load resisting system as well as secondary elements that may influence the seismic response (e.g., gravity members for vertical response). The results of the finite element model will be summarized in the Operating License application.

Structural mass is assigned to the models to capture the self-weight of the structural elements and the weight of permanently attached heavy equipment (e.g., reactor). The mass also accounts for a portion of the design live loads (25% of the live load for loads less than 200 psf, 50 psf otherwise) and 25% of the design uniform snow load. Assignment of the structural mass in the models will be described in the Operating License application.

A cracking analysis is performed using the 5% damped DRS to determine if cracking occurs in the structural elements at the design level. Elements judged to be cracked at the design level have stiffnesses modified per Table 3-2 of ASCE 4-16. Structural damping is assigned per Table 3-1 of ASCE 4-16 consistent with the response level determined from the level and extent of cracking anticipated at the design level.

3.4.1.6 Response Analysis

The structural models are subjected to a three-component seismic input, discussed in Section 2.5 and Section 3.4.1.2, to develop structural forces and in-structure response spectra (ISRS) used for SDC-3 structural and equipment qualification, respectively. Response analysis is performed at the seismic levels necessary to demonstrate the SDC-3 SSCs achieve their target performance goal.

Seismic response analysis is performed following Chapter 4 of ASCE 4-16 using deterministic, linear analysis. The relative importance of soil-structure interaction effects, using the characterization of the subsurface materials supporting the SDC-3 structures, defined compatible with those described in Section 2.5, are considered based on the guidance in Chapter 5 of ASCE 4-16. Additional details about the soil-structure interaction analysis results and modeling methods and assumptions will be summarized in the Operating License application. Modeling methods and assumptions as well as results of the seismic response analysis, including structural forces and ISRS, will also be summarized in the Operating License application.

3.4.1.7 Seismic Qualification

Limit states for SDC-3 SSCs are assigned based on the target seismic performance goals of ASCE 43-19 (see Section 3.4). Specific criteria for the qualification of structures and systems and components are outlined in Section 3.6.

3.4.2 Non-Safety Related SSCs and Seismic Design

With respect to seismic design, non-safety related SSCs are designed according to the local building code, the 2012 IBC. For the seismic input, the design basis ground motion is defined in accordance with the deterministic processes of local building code, the 2012 IBC, which refers to ASCE/SEI 7-10 (Reference 6).

Site-specific ground motion parameters are determined per Chapter 21 of ASCE/SEI 7-10. The site response analysis used to inform the SDC-3 (Section 2.5) input will be used to determine the risk-targeted maximum considered earthquake (MCE_R) for the site.

Seismic analysis and qualification of non-safety related SSCs is performed in accordance with the 2012 IBC. Seismic design requirements for non-safety related structures follow Chapter 12 of ASCE/SEI 7-10. Seismic design for non-safety related SDC-2 systems and components follow Chapter 13 of ASCE/SEI 7-10. Exceptions to ASCE/SEI 7-10 for non-safety related structures, as required by the Tennessee building code, are applied as needed.

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 an earthquake. 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 located in the free-field and in the safety-related portion of the Reactor Building. The free-field instrument is mounted on rock or competent ground generally representative of the dynamic site characteristics. The instrumentation records time-history data at time increments suitable to capture the range of vibration frequencies in the design basis earthquake spectra. Seismic instrumentation is designed such that if there is a loss of power, recording still occurs. Instrumentation is housed in appropriate weather and creature-proofed enclosures.

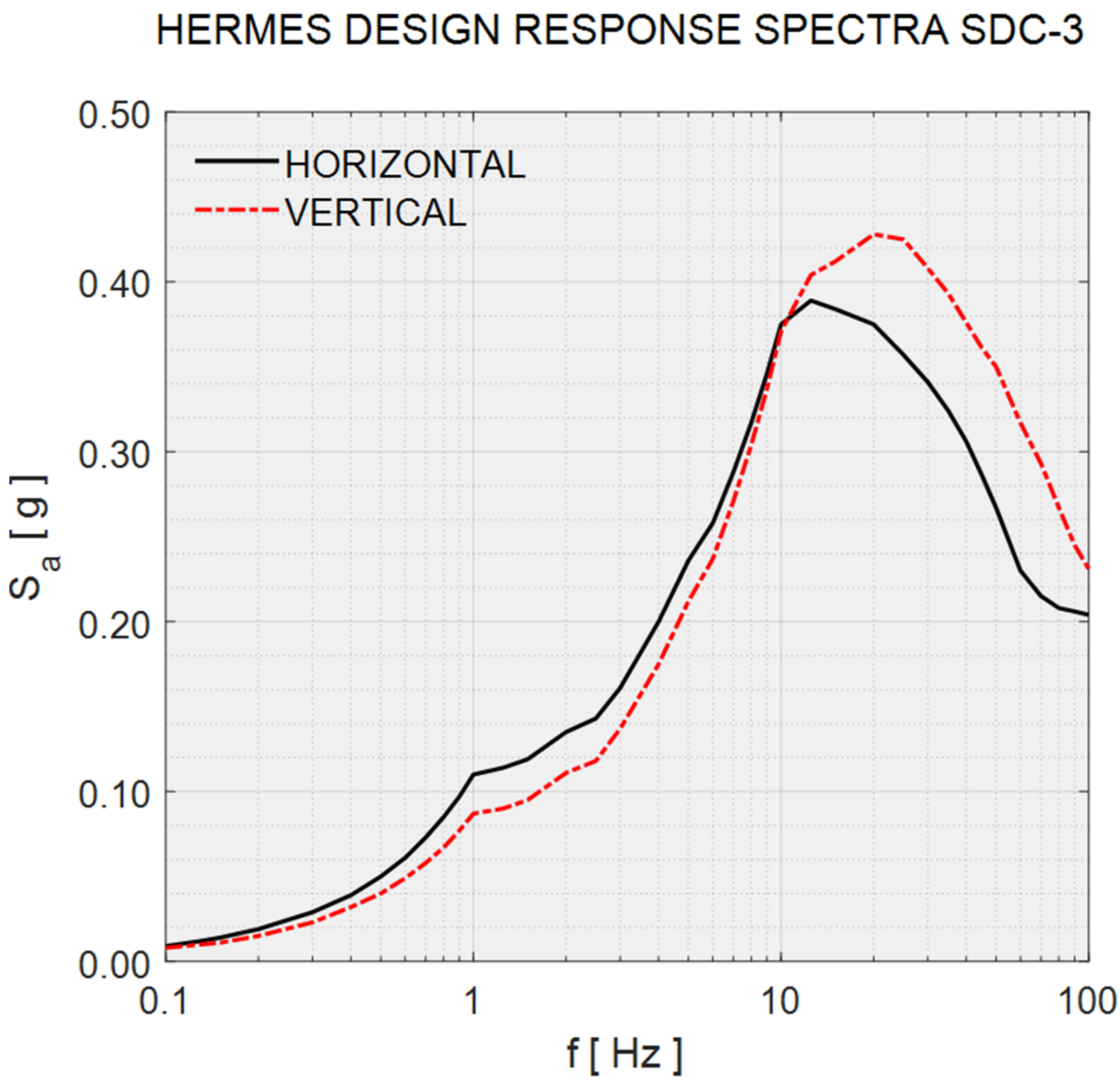
3.4.3.2 Seismic Instrumentation Operability and Characteristics

The 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.

3.4.4 References

1. American Society of Civil Engineers, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," ASCE 43-19. 2019.
2. International Code Committee, "International Building Code." 2012.
3. American National Standards Institute, American Nuclear Society, "Research Reactor Site Evaluation." ANSI/ANS 15.7. 1977
4. International Atomic Energy Agency, "Siting of Research Reactors," IAEA-TECDOC 403. 1987.
5. American Society of Civil Engineers, "Seismic Analysis of Safety-Related Nuclear Structures," ASCE 4-16. 2017.
6. American Society of Civil Engineers, Seismic Engineering Institute, "Minimum Design Loads for Buildings and Other Structures," ASCE/SEI 7-10. 2011.
7. International Atomic Energy Agency, "Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory," IAEA-TECDOC 348. 1985.
8. International Atomic Energy Agency, "Consideration of External Events in the Design of Nuclear Facilities Other Than Nuclear Power Plants, with Emphasis on Earthquakes,"IAEA-TECDOC-1347. 2003.

Figure 3.4-1: Horizontal and Vertical Design Response Spectra



3.5 PLANT STRUCTURES

3.5.1 Description of Plant Structures

Figure 2.1-3 shows the location and orientation of the Reactor Building on the site. The building is approximately 250 ft long and 100 ft wide. A portion of the Reactor Building provides protection to safety-related SSCs from the effects of natural phenomena and external event hazards discussed in Sections 3.2, 3.3, and 3.4. Figure 3.5-1 shows the principal structural elements of the Reactor Building. The figure also shows the portion of the safety-related Reactor Building structure, which uses base isolation, and the non-safety related balance of the Reactor Building surrounding the isolated superstructure.

The foundation for the safety-related portion of the building is a below-grade mat slab. The base isolation system is supported by the foundation and is located in an accessible basement beneath the isolated superstructure. The isolators are supported concrete pedestals and interlinking shear walls in the basement. Isolators will be spring-dashpot elements (e.g., GERB base control system (BCS) isolators). The foundation, isolation system, and associated structural elements form the substructure of the safety-related portion of the building.

The isolation substructure supports the basemat of the safety-related superstructure and the superstructure itself. The superstructure is a reinforced concrete structure that is a hybrid of cast-in-place and precast concrete structural elements. A “moat” provides seismic separation between the safety-related portion of the Reactor Building and the non-safety portion and is large enough to accommodate the seismic displacements of the isolators.

The safety-related portion of the Reactor Building is divided into cells. The cells contain all the safety-related SSCs in the facility and some non-safety related SSCs. One cell contains the reactor cavity, the decay heat removal system, the reactivity control and shutdown system, and the heat rejection radiator. Another contains the pebble handling and storage system, and other safety-related support SSCs.

The non-safety related portion of the Reactor Building is highlighted in Figure 3.5-1 and is comprised of a maintenance hall including a high-bay shell, maintenance corridors, truck bay, and auxiliary worker inhabited areas. It is a steel frame construction with an independent foundation system consisting of a mat slab with grade beams. This non-safety related portion of the Reactor Building does not contain any safety-related SSCs. This portion of the building is designed so that its failure does not interfere with safety functions of SSCs located in the safety-related portion of the building or the safety-related portion of the Reactor Building.

The top part of the Reactor Building is a high bay through which a gantry crane moves. The crane is supported by the non-safety related portion of the Reactor Building. As mentioned in Section 3.2.4, the roof of the non-safety related portion of the building is sloped using either an arch or a slant so that accumulation of water and ice does not result in significant loads. The image in Figure 3.5-1 shows an exterior roof that is slanted.

Other buildings on the site do not contain safety-related SSCs and serve no safety-related function. This includes the Main Control Building. The Main Control Building is a stand-alone building on the site that contains the plant control system and reactor protection system human system interface consoles. There are no postulated events in the safety analyses described in Chapter 13 that rely on operator actions credited for implementing a safety function to maintain doses below limits. The Main Control Building does not serve a safety-related function, but does provide the location for operators to perform normal operational duties and to support monitoring capabilities after postulated events.

The safety functions of the safety-related portion of the Reactor Building are:

- Protection of safety-related SSCs from design basis natural phenomena and external hazards
- Structural support for safety-related SSCs located on the safety-related portion of the Reactor Building
- Protection from adverse effects of non-safety related SSCs failures on the ability of safety-related SSCs to perform their safety functions
- Prevent interactions between reactor coolant (Flibe) and water contained in concrete in the safety-related portion of the reactor building.

3.5.2 Design Bases

- Consistent with PDC 1, the safety-related portion of the Reactor Building is designed in accordance with industry codes and standards, and the quality assurance program described in Section 12.9.
- Consistent with PDC 2, the safety-related portion of the Reactor Building is designed to provide protection for safety-related SSCs housed within to perform their safety functions in design basis meteorological, water, and seismic events as described in Sections 3.2, 3.3, and 3.4.
- Consistent with PDC 3, the safety-related portion of the Reactor Building is designed with design features to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.
- Consistent with PDC 75, the Reactor Building is designed to protect the geometry of the decay heat removal system from postulated natural phenomena events.
- Consistent with PDC 76, the Reactor Building is designed to permit appropriate periodic inspection and surveillance of safety-related structural areas.

3.5.3 System Evaluation

Although the non-safety related portion of the Reactor Building surrounds the safety-related portion of the Reactor building, the non-safety related portion is not credited in the safety analysis. Neither the safety-related nor non-safety related portion of the Reactor Building is credited in the safety analysis to perform a safety-related containment function for retention of fission products since the design relies on a functional containment concept (see Chapter 13). Similarly, the non-safety related portion of the Reactor Building is not credited to provide physical protection to safety-related SSCs from the effects of normal or high winds (see Section 3.5.3.1), or from the effects of design basis earthquakes (see Section 3.5.3.3). Finally, the non-safety related portion of the reactor building is not credited to provide protection to safety-related SSCs from the effects of water damage (see Section 3.5.3.2). However, the shape of the exterior roof precludes adverse effects related to accumulation of water and ice. A list of load combinations for the safety-related portion of the Reactor Building is provided in Table 3.5-1.

Consistent with PDC 1, the safety-related portion of the reactor building is under the quality assurance program described in Chapter 12. The safety-related portion of the Reactor Building is designed to the local building code, ASCE/SEI 7-10 (Reference 1), and augmented for specific design basis natural phenomena as described below. The non-safety related portion of the Reactor Building is designed to local building codes which invoke ASCE/SEI 7-10.

Consistent with PDC 3, the safety-related portion of the Reactor Building is designed to perform its safety function in the event of a fire hazard. The safety-related portion of the Reactor Building includes design features which minimize the probability and effect of fires and explosions by the use of low combustible materials and physical separation. These design features, in conjunction with the fire protection program described in Section 9.4, provide assurance that the safety-related portion of the Reactor Building conforms to PDC 3.

The decay heat removal system (DHRS) contains safety-related SSCs, which are located in the safety-related portion of the Reactor Building. The design of the safety-related portion of the Reactor Building protects the safety-related SSCs within it from adverse effects on those safety-related SSCs from design basis natural phenomena described in Sections 3.2, 3.3, and 3.4. This satisfies PDC 75.

The safety-related portion of the Reactor Building is designed to permit appropriate periodic inspection and surveillance. This includes the basement area containing the base isolation system, which is a safety related SSC. This satisfies PDC 76.

Consistent with PDC 2, the safety-related portion of the reactor 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 non-safety related portion of the reactor building is damaged due to the design basis wind, water or earthquake events described in Sections 3.2, 3.3, and 3.4. The system evaluation for PDC 2 is provided in the following subsections.

3.5.3.1 Conformance with PDC 2 for Meteorological Events

Section 3.2.1 describes the normal wind loads used as design parameters for the safety-related portion of the reactor 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 safety-related portion of the reactor 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 safety-related portion of the reactor building.

Consistent with PDC 2, the safety-related SSCs are located in the safety-related portion of the Reactor Building which is designed to protect safety-related SSCs from the effects of design basis normal and high winds and snow. The safety-related portion of the Reactor Building is a reinforced concrete structure designed to meet American Concrete Institute (ACI) 349-2013 (Reference 2) with internal safety-related steel structures designed in accordance with ANSI and American Institute of Steel Construction (AISC) Standard N690-18 (Reference 3). Both ACI 349-2013 and AISC N690-18 are standards specific to the design of safety-related 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 from Section 3.2. By designing the safety-related portion of the Reactor Building in accordance with these two standards, the safety-related portion of the Reactor Building satisfies PDC 2 for design basis loads from normal winds, high winds, and snow, as discussed in Section 3.2.

3.5.3.2 Conformance with PDC 2 for Internal and External Flooding

This section describes how the design basis for the safety-related portion of the Reactor Building, with respect to water damage (internal and external flooding), provides reasonable assurance that potential water damage will not preclude safety-related SSCs from performing their safety-related functions. Section 3.3 characterized the design basis loads related to external and internal flooding postulated events. This section describes how the safety-related portion of the Reactor Building is designed to address those loads.

3.5.3.2.1 External Flood Design Features

Consistent with PDC 2, the safety-related SSCs are located in the safety-related portion of the Reactor Building which is designed to protect safety-related SSCs from the effects of design basis external flooding described in Section 3.3.

The facility is a passively dry site with respect to external flooding hazards. Section 3.3 describes that in the design basis flood event, there are no loads on the safety-related portion of the Reactor Building that is above grade. The basement containing the seismic isolator units is about 20 feet below grade. The safety-related portion of the Reactor Building is designed to withstand buoyant forces and groundwater, including groundwater associated with the design basis flood.

No SSCs located in the basement are credited to mitigate the effects of a postulated external flood event. The basemat of the safety related portion of the Reactor Building, which is supported by the base isolators, as discussed in Section 3.5.1, is at grade level and there are no safety-related SSCs in the safety-related portion of the reactor building located below the basemat elevation that are classified as safety-related for flooding events. Therefore, PDC 2 is met for design basis flood events based on the location above grade level of all safety-related SSCs that are credited to mitigate the effects of a postulated external flood.

Although they do not perform a safety function to mitigate the adverse effects of a postulated external flood event, the seismic isolator units are on elevated pedestals above the foundation slab. The base isolation basement is a reinforced concrete safety-related structure with the following features:

- Water stops are provided in construction joints below flood level.
- External surfaces exposed to flood level have waterproof coating.

Furthermore, the safety-related portion of the Reactor Building is a reinforced concrete structure designed to meet ACI 349-2013. ACI 349-2013 is specific to the design of safety-related nuclear structures and has built-in margin. ACI 349 is used to design a structure that can withstand the postulated external flooding water loads from Section 3.3. With respect to buoyant forces from a postulated external flood event on the basement area of the safety-related portion of the Reactor Building, based on a flood level no higher than grade, the weight of the building offsets the potential buoyant forces on the basement. By designing in accordance with ACI 349-2013, the safety-related portion of the Reactor Building satisfies PDC 2 for design basis loads from external flooding as discussed in Section 3.3.

Finally, consistent with PDC 2, grading and drainage on the site preclude loads from precipitation affecting the safety-related portion of the Reactor Building. Specific grading and drainage features will be described in the application for an Operating License.

3.5.3.2.2 Internal Flood and Spray Design Features

This section describes the design features that satisfy PDC 2 with respect to protection from internal flooding for safety-related SSCs. Safety-related SSCs that are vulnerable to water damage from internal spray or floods are elevated above the floor and shielded, or otherwise protected, from potential spray. Water is directed away from enclosures for safety-related equipment and sloped floors and curbs preclude water entry into these areas. Where there is a potential for pebbles to be on a sloped or curbed floor, features prevent pebbles from rolling so that pebbles on the floor of the safety-related portion of the Reactor Building maintain a geometrically safe configuration for criticality.

Internal flooding or spraying in the safety-related portion of the Reactor Building has three potential sources: water system with SSCs located in the safety-related portion of the reactor building, water system SSCs located in the non-safety related portion of the Reactor Building, and fire protection water.

For water systems with SSCs located in the safety-related portion of the Reactor Building, the amount of water is limited by design. The maximum flow rate and the volume of water available for release from a break in the safety-related portion of the Reactor Building, is used to determine the effect of internal flooding or spraying on safety-related equipment. The quantity and flow rate of water is limited to the gravity-driven pressure head above the break location. A pump trip in a water system is assumed to terminate the flow and a constrained amount of fluid is assumed to spill into the facility.

For water sources external to the safety-related portion of the Reactor Building (e.g., fire water), automatic or a manual termination of flow will be specified in the application for the Operating License. The fire protection system implements NFPA 801, "Standard for Fire Protection for Facilities Handling Radioactive Materials" (Reference 3). The water collection due to the potential failure of the fire protection piping is bounded by the total discharge from the operation of the fire protection system. The water collection system can accommodate the total firefighting water volume. Sloped floors and curbs prevent fire protection water from draining into the radioactive waste handling system drains. Spray shields, or similar, prevent fire protection water from spraying safety-related SSCs that would be sensitive to water spray.

Safety-related SSCs are protected from spilled Flibe and Flibe-bearing components are also protected from water to prevent interaction between water and Flibe. Features include steel liners, catch pans or troughs, or similar design solutions.

Those pipes, vessels, and tanks with the potential to flood or spray safety-related portions of the Reactor Building are seismically qualified in accordance with local building code and consistent with the seismic design category based on the SSC's safety classification. There are no pressurized piping systems in the safety-related portion of the Reactor Building, therefore pipe whip effects are not considered.

Further information on the analysis of the impacts of internal flooding and spraying will be provided with the application for an Operating License.

3.5.3.3 Conformance with PDC 2 for Earthquakes

Section 3.4 discussed the design basis earthquake characteristics that are the input for the design of the safety-related portion of the Reactor Building. The safety-related portion of the reactor building is designed consistent with the graded approach in ASCE 43-19 (Reference 4). See Section 3.4 for more information about the graded approach. By meeting ASCE 43-19, the safety-related portion of the Reactor Building provides protection for safety-related SSCs from design basis earthquakes, consistent with PDC 2.

The safety-related portion of the Reactor Building uses base isolation as described in Section 3.5.1. The seismic isolation system is designed to limit the loads from design basis earthquakes on safety-related SSC, consistent with PDC 2.

3.5.3.3.1 Seismic Design of the Safety-Related Portion of the Reactor Building

Seismic qualification of SDC-3 structures follows the requirements of Section 5 of ASCE 43-19. Structural demands are determined based on the results of the response analysis outlined in Section 3.4.1. 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 43-19, respectively, for the applicable limit states. Strength-based qualification of structural elements utilize, when appropriate, the inelastic energy absorption factors discussed in Section 5.1.3 of ASCE 43-19 and summarized in Table 5-1 of ASCE 43-19. Allowable drift and rotation limits are based on the discussion in Section 5.2.3 of ASCE 43-19 and summarized in Tables 5-2 and 5-3 of ASCE 43-19.

The facility SDC-3 structures' primary lateral force resisting system (LFRS) is not credited as a radiological barrier. Therefore, the SDC-3 structures' seismic performance criteria, and corresponding limit state selection, is limited to providing physical support to other SDC-3 SSCs and for collapse prevention.

No safety-related SSCs cross the moat that surrounds the safety-related portion of the Reactor Building. Non-safety related SSCs that cross the moat to the safety-related portion of the Reactor Building use design features to accommodate differential displacements of the two parts of the Reactor Building. 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 crossing the moat due to differential motion between the parts of the building during a design basis earthquake. This is not a safety-related function, but the features reduce the likelihood that during an earthquake non-safety related SSCs would adversely affect a safety-related SSC's ability to perform its safety function.

3.5.3.3.2 Seismic Isolation System

The safety-related portion of the Reactor Building design uses a seismic isolation system to limit seismic demands on SDC-3 SSCs. This includes both the structure itself and the systems and components housed within. The seismic isolation system is part of the lateral force resisting system of the safety-related portion of the Reactor Building and is subject to design requirements unique to the isolation system.

The base isolation system design implements Chapter 9 of ASCE 43-19.The isolators and their connections to the super- and sub-structures are designed for the forces and displacements computed by the response analysis outlined in Section 3.4.1. Further details of the design of the base isolation system and associated structural analysis will be provided in the application for an Operating License.

Wind design effects for the safety-related portion of the Reactor Building are accounted for in the design as described in Section 3.5.3.1, including high wind events. Under these demands, the lateral displacement of the isolation system due to wind is verified not to exceed the displacement from a design basis earthquake, with margin.

A moat is provided around the seismically isolated superstructure to accommodate displacement of the isolation system during a seismic event and avoid interaction of the superstructure with the adjacent non-isolated portion of the building. The moat is sized to have a displacement capability large enough such that impact with the moat will not impede the seismic isolation system from meeting the SDC-3 target performance goal of 1E-4/year.

Limit states for SDC-3 SSCs are assigned based on the target seismic performance goals of ASCE 43-19. Design criteria for the qualification of specific SSCs are outlined in Section 3.6.

3.5.3.4 Conformance with PDC 2 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 safety-related structures. The safety-related portion of the Reactor Building is constructed of robust reinforced concrete such that credible accidental external explosions do not result in hazards to safety-related SSCs located in that portion of the building. Internal explosions are considered in the fire hazards analysis (see Section 9.4).

Accidental aircraft impact (AAI) from the proposed nearby airport, as discussed in Section 2.2, is also considered in the design of the safety-related portion of the Reactor Building. The design of the safety-related portion of the Reactor Building is evaluated for global and local effects of AAI hazards from light general aviation aircraft.

The global impact response is analyzed using an energy balance method consistent with Department of Energy (DOE) Standard DOE-STD-3014-2006 (Reference 5). The permissible ductility limits for reinforced concrete elements and truss members are consistent with Appendix F of ACI 349 and Chapter NB of AISC N690, respectively. From these references, the available energy absorption capacity of the structure at the critical impact locations is determined. Section 2.2 provides a projection of the type of aircraft that will be used at the proposed nearby airport. The analysis of global impact response uses aircraft models representative of the projected types with respect to mass of the aircraft, speed, and fuel capacity. Attachment E of Lawrence Livermore National Laboratory UCRL-ID-123577 (Reference 6) is used in the analysis to determine the probabilistic distributions of horizontal and vertical impact velocities corresponding with the 99.5 percent of the impact velocity probability distribution. The analysis includes impacts at locations that bound the effect of AAI on the safety-related portion of the Reactor Building with respect to the global impact response.

The local impact response on the safety-related portion of the Reactor Building is analyzed consistent with DOE-STD-3014-2006. The structure is designed to address credible failure modes based on Appendix F of ACI 349. Using the credible failure modes, DOE-STD-3014-2006 is used to calculate wall and ceiling sizing requirements for the safety-related portion of the Reactor Building. The analysis of local impact response uses aircraft models representative of the projected types (see Section 2.2) with respect to mass of the engine, speed, and fuel capacity. The analysis includes impacts at locations that bound the effect of AAI on the safety-related portion of the Reactor Building with respect to the local impact response.

Additional detail about the structural design features for the safety-related portion of the Reactor Building informed by the results of the analysis will be provided in the application for the Operating License.

3.5.4 Testing and Inspections

Testing and inspections of seismic isolator units is conducted consistent with ASCE 43-19. Prior to installation, testing is performed on both prototype and production isolators consistent with the guidance set forth in Section 9.5 of ASCE 43-19. Testing requirements and procedures follow Section 9.5.2 of ASCE 43-19. Prototype testing is used to verify the displacement capacity of the isolators up to that necessary for demonstrating the isolation system meets its target performance goal of 1E-4/year. Production isolators are manufactured in the same manner and with the same materials of the prototype isolators. Each production isolator is tested per the requirements and procedures of Section 9.5.3 of ASCE 43-19 for the SDC-3 DRS. A monitoring and inspection program for the isolators meets Section 9.2.1.6 of ASCE 43-19.

3.5.5 References

1. American Society of Civil Engineers, Seismic Engineering Institute, “Minimum Design Loads for Buildings and Other Structures,” ASCE/SEI 7-10. 2011.
2. American Concrete Institute, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” ACI 349-13. 2013.
3. American National Standards Institute ANSI/ASCI N690-18, Specification for Safety-Related Steel Structures for Nuclear Facilities.” 2018.
4. American Society of Civil Engineers, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” ASCE 43-19. 2019.
5. Department of Energy, “Accident Analysis for Aircraft Crash into Hazardous Facilities, ”DOE-STD-3014-2006.” 2006.

6. Department of Energy, “Structures, Systems and Components Evaluation Technical Support Document for the DOE Standard on Accident Analysis for Aircraft Crash into Hazardous Facilities,” UCRL-ID-123577. 1996.

Table 3.5-1: Load Combinations for the Safety Related Portion of the Reactor Building

Load Category	Load Combination*
Normal	$D + F + T_o + R_o$ $D + F + T_o + R_o + L + H + C_{cr} + L_r$
Severe Environmental**	$D + L + F + T_o + R_o + E_o + H$ $D + L + F + T_o + R_o + H + W$
Extreme Environmental	$D + L + H + F + C_{cr} + T_o + R_o + E_{ss}$ $D + F + H + L + T_o + R_o + W_t$
Abnormal	$D + F + L + H + T_a + R_a + C_{cr}$ $D + F + H + L + T_a + R_a + E_{ss}$

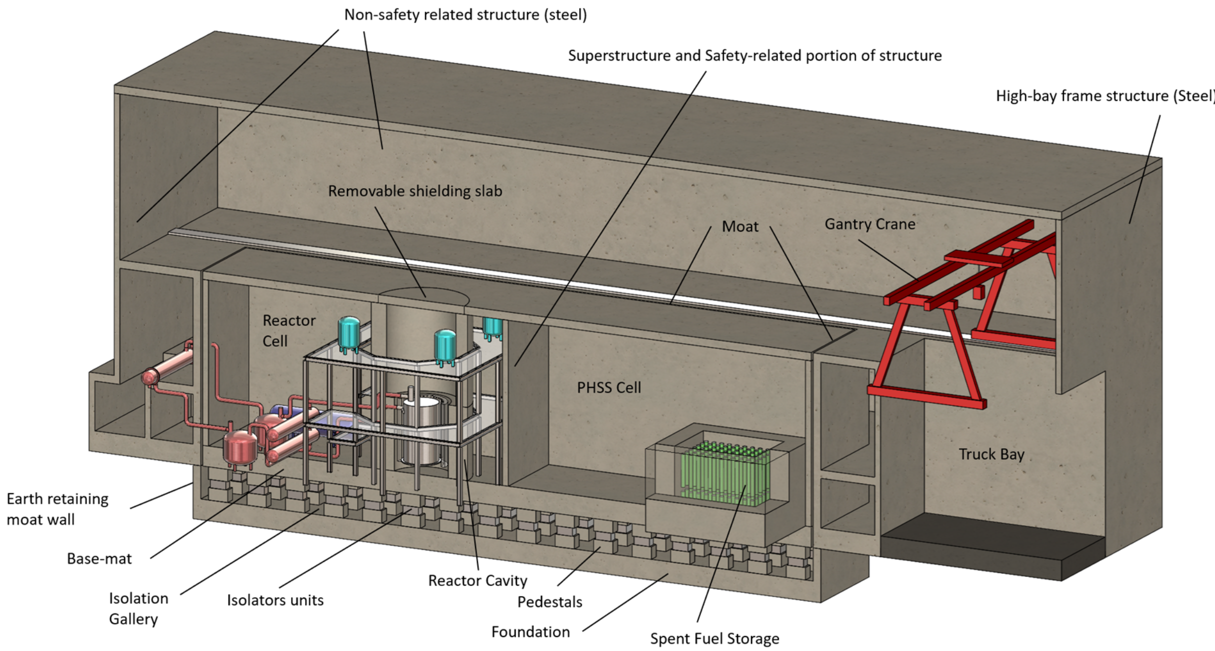
*****Load combination refers to the types of loads considered acting simultaneously. Application of load factors and specific details of load combination effects are per the applicable design standard.

******The T_o terms for Severe Environmental load combinations are not included the ACI code and are not applied to safety-related Reactor Building concrete structures. They are included in the AISC code and are applied to safety-related Reactor Building steel structures.

Load Nomenclature:

D	Dead loads
L	Live loads
L_r	Roof loads, including snow or rain as applicable
F	Fluid Loads
H	Soil Loads
C_{cr}	Crane load-rated capacity
W	Normal wind loads
W_t	High wind loads (tornado and hurricane), including corresponding missiles
T_o	Thermal loads during normal operating and shutdown conditions
T_a	Thermal loads as a result of accident conditions and including T_o
R_o	Pipe and equipment reactions during normal operating and shutdown conditions
R_a	Pipe and equipment reactions as a result of accident conditions and including R_o
E_o	Loads generated by 1/3 of design basis earthquake (DBE)
E_{ss}	Loads generated by DBE

Figure 3.5-1: Schematic of the Reactor Building



3.6 SYSTEMS AND COMPONENTS

This section describes the design bases for the systems and components required to function for safe reactor operation and shutdown. Section 3.6.1 describes the safety functions performed by safety-related SSCs and Section 3.6.2 describes how SSCs are classified.

3.6.1 General Design Basis Information

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.

- Prevent uncontrolled release of radionuclides
- Remove decay heat in the event of a postulated event
- Control reactivity in the reactor core

Section 3.6.2 describes the safety classifications of SSCs based on performance of one of the functions in the fundamental safety functions listed above. Table 3.6-1 identifies the safety classification of SSCs within a system. Note that not all SSC within a system may be safety-related.

3.6.1.1 Prevention of Uncontrolled Release of Radionuclides

The reactor design employs a high-temperature graphite-matrix coated TRISO particle fuel and a chemically stable, low-pressure, molten fluoride salt coolant. These features provide a functional containment (see Section 6.2) which is relied on as a means of retaining fission products and limiting the release of radionuclides to the environment during normal operations and postulated events. The elements of the functional containment for fuel in the reactor core include the TRISO fuel particle’s three layers (IPyC, SiC, and OPyC) surrounding the fuel particles, and the chemical properties of the reactor coolant (Flibe). The design of the TRISO fuel pebbles is discussed in Section 4.2 and the reactor coolant is discussed in Section 5.1.

Other SSCs support the ability of the functional containment strategy for fuel in the reactor core to limit the release of radionuclides during postulated events. These supporting safety-related systems are:

- The reactor protection system (RPS) ensures that pebble extraction and insertion via the pebble handling and storage system (PHSS) is deactivated upon a reactor trip so pebbles are no longer removed from or added to the reactor core. RPS also stops operation of both the primary salt pump (PSP) and the intermediate salt pump (ISP) to ensure the level of Flibe remains constant in the reactor core and that pebbles in the core remain covered in Flibe. See Section 7.3 for a description of the RPS.
- The reactor vessel and internals provide structural support and form the reactor core region which maintains the TRISO pebbles in a coolable geometry within the core where they remain covered with Flibe. This includes the graphite reflector which contributes to providing a coolable geometry within the core. See Section 4.3 for a description of the reactor vessel and internals.
- The safety-related portion of the Reactor Building is designed to provide protection for the functional containment from the effects of natural phenomena on the reactor vessel and associated safety-related SSCs. The safety-related portion of the Reactor Building is also designed to prevent interactions between Flibe and water. No portion of the Reactor Building is credited to perform a fission product containment function in the Chapter 13 safety analysis. See Section 3.5 for a description of the Reactor Building structure.

Fuel outside the reactor core is located in the PHSS. Fuel pebbles in the PHSS are not submerged in reactor coolant. Therefore, the TRISO layers in the fuel particles provide functional containment while pebbles are in the PHSS such that radionuclides are contained within the particles for postulated events.

Other SSCs support the ability of the functional containment strategy for fuel in the PHSS to limit the release of radionuclides during postulated events. These supporting systems are:

- The PHSS storage transporter provides protection of fuel pebbles from postulated events, by preventing pebbles from rolling. See Section 9.3 for a description of the PHSS.

Safety-related and non-safety related fluid systems may contain circulating radiological activity. Table 3.6-2 provides a compilation of the codes and standards to which these fluid system SSCs are designed.

3.6.1.2 Removal of Decay Heat During a Postulated Event

During postulated events, the reactor vessel and internals are designed to support natural circulation so that heat is transported from the fuel pebbles in the core to the exterior surface of the vessel via the reactor coolant. The DHRS passively absorbs heat radiated from the surface of the reactor vessel and transports the heat for rejection directly to the atmosphere. Further information about the design basis for the transportation of heat to the atmosphere is discussed in Section 6.3.

Other SSCs support the ability of the DHRS to remove decay heat during postulated events. These supporting safety-related SSCs include:

- The reactor vessel and the internal graphite reflector in the core provide the structural support to maintain a coolable geometry and provides a coolant flow path for natural circulation. See Section 4.3 for a description of the reactor vessel and graphite reflector.
- Although the DHRS operates continually above a threshold fission product accumulation level, the RPS design provides a priority demand actuation signal and ensures that the plant control system cannot interfere with or override the demand for DHRS removal of decay heat. See Section 7.3.
- The safety-related portion of the Reactor Building provides structural support for the DHRS and reactor vessel and protection from adverse effects of design basis natural phenomena hazards. See Section 3.5 for a description of the Reactor Building structure.
- The reactor vessel support system (RVSS) provides structural support for the reactor vessel and maintains the physical geometry and spatial distance between the DHRS and reactor vessel to facilitate heat transfer. See Section 4.7 for a description of the RVSS.

3.6.1.3 Control of Reactivity in the Core

The reactivity control and shutdown system (RCSS) is designed to ensure that reactivity control and shutdown elements can be inserted into the reactor to provide reactivity control in response to postulated events. The ability of the RCSS to perform its safety function does not rely on the performance of any auxiliary or distribution systems other than the RPS when normal power is available. When normal power is not available, the RCSS releases the control and shutdown elements. See Section 4.2.

Other SSCs support the ability of the RCSS to control reactivity during postulated events. These supporting safety-related SSCs include:

- The reactor vessel and vessel internals maintain the geometry of the core to support control and shutdown element insertion. See Section 4.3 for a description of the reactor vessel and internals.
- The safety-related portion of the Reactor Building provides protection from the effects of natural phenomena on the RCSS. See Section 3.5 for a description of the Reactor Building structure.
- The design of the reactor coolant also supports the control of reactivity via reactivity coefficients. See Section 4.5 for a description of the core design.

3.6.2 Classification of Structures, Systems, and Components

SSCs are assigned safety, seismic, and quality classifications consistent with their safety functions. These classifications are described below. Table 3.6-1 provides a summary of these classifications for all SSCs.

3.6.2.1 Safety Classification

SSCs have two possible safety classifications: safety-related or non-safety related. An SSC is classified as safety-related if it meets the definition of safety-related from 10 CFR 50.2 (with exceptions as described in Section 1.2.3). For the KP-FHR technology, the definition of safety-related is modified from 10 CFR 50.2, to be:

> Safety-related structures, systems, and components means those structures, systems, and components that are relied upon to remain functional during and following design basis events to assure:
>
> (1) The integrity of the portions of the reactor coolant boundary relied upon to maintain coolant level above the active core;
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition; or
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11

Note that for the KP-FHR technology, the definition above reflects a departure from the definitions in 10 CFR 50.2 for light water reactors that include the terminology “integrity of the reactor coolant pressure boundary.” As described in Section 1.2.3 and the Regulatory Analysis for the Kairos Power Salt-Cooled, High Temperature Reactor Topical Report (Reference 1), this departure is necessary because the technology associated with the KP-FHR is based on a near atmospheric pressure design and the reactor coolant boundary does not provide a similar pressure related or fission product retention function as light-water reactors for which these definitions were based.

SSCs that do not meet the definition, as modified above, are classified as non-safety related.

3.6.2.2 Seismic Classification

SSCs are designed according to their safety classification. Safety-related SSCs are classified as SDC-3 consistent with ASCE 43-19 (Reference 2). Section 3.4 discusses the SDC-3 classification and Section 3.5 discusses requirements for SSCs that are required to maintain their function in the event of a design basis earthquake. All safety-related SSCs are located in the safety-related portion of the Reactor Building, which is discussed in Section 3.5.1.

The credited safety systems designed to function in a postulated event are described in Chapter 13. For a design basis earthquake, the SDC-3 SSCs that are relied upon to perform a specific credited safety function are listed in Table 3.6-1.

Safety-related systems and components are qualified to maintain their safety function during a design basis earthquake, after a design basis earthquake, or both, depending on the function performed. For example, the reactor vessel is required to perform its safety function (i.e., maintain structural integrity) both during and after a design basis earthquake, whereas the decay heat removal system is required to perform its safety function only after the event, and not during. The specific safety function, therefore, is used to define the ASCE 43-19 Limit State that is used to qualify the SDC-3 SSCs.

Seismic qualification is accomplished through analysis, testing or a combination of those methods. Acceptance criteria is defined in accordance with ASCE 43-19, Chapter 8, and/or its references.

SSCs that are non-safety related are designed in accordance with local building code (IBC 2012, Reference 13)as discussed in Section 3.4.2. Non-safety related SSCs are subject to the seismic design requirements of the local building code, ASCE/SEI 7-10 (Reference 3).

3.6.2.2.1 Seismic Qualification by Analysis

Seismic qualification by analysis follows Section 8.2 of ASCE 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 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 an earthquake can be established and a margin of loss of functionality during an earthquake 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.).

3.6.2.2.2 Seismic Qualification by Testing

Seismic qualification by testing follows Section 8.3 of ASCE 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.

3.6.2.3 Quality Classification

The quality classification for SSCs conforms with the requirements of Kairos Power's Quality Assurance Program which is discussed in Section 12.9. Safety-related SSCs are classified as Quality-Related, while non-safety related SSCs are classified as Not Quality-Related. These classifications are shown in Table 3.6-1.

3.6.3 References

1. Kairos Power, LLC, "Regulatory Analysis for the Kairos Power Salt-Cooled, High Temperature Reactor," KP-TR-004-NP-A. June 2022.
2. American Society of Civil Engineers, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," ASCE 43-19. 2019.
3. American Society of Civil Engineers, Seismic Engineering Institute, "Minimum Design Loads for Buildings and Other Structures," ASCE/SEI 7-10. 2011.
4. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors." 2017.
5. ASME, Boiler and Pressure Vessel Code, Section VIII, Divisions 1 and 2, "Rules for Construction of Pressure Vessels," New York, NY. July 2017.
6. ASME Standard B31.1, "Power Piping," 1999 Edition, New York, NY. A-9.
7. ASME Standard B31.3, "Process Piping," 2016 Edition, New York, NY.
8. American Petroleum Institute, 610, "Centrifugal Pumps for Petroleum, Heavy Duty Chemical, and Gas Industry Services," 1995.
9. American Petroleum Institute, 674, "Positive Displacement Pumps-Reciprocating." 1995.
10. American Petroleum Institute, 675, "Positive Displacement Pumps-Controlled Volume." 1994.
11. American Petroleum Institute, 650, "Welded Steel Tanks for Oil Storage." 1998.

12. American Petroleum Institute, 620, “Design and Construction of Large, Welded, Low-Pressure Storage Tanks.” 1990.
13. International Code Committee, “International Building Code.” 2012.

Table 3.6-1: Structures, Systems, and Components

SSC Name	Safety Classification	Seismic Design	Quality Program	SAR Section	Plant Area
Reactor System					
Fuel Pebbles	Safety-related	N/A	Quality-Related	4.2.1	SR area[1]
Moderator Pebbles	Non-safety related	N/A	Not Quality-Related	4.2.1	SR area
Reactivity Control and Shutdown System (RCSS)					
Control Elements	Non-safety related	Local Building Code	Not Quality-Related	4.2.2	SR area
Shutdown Elements, including latching/release mechanism	Safety-related	SDC-3	Quality-Related	4.2.2	SR area
RCSS drive systems, except shutdown element latching mechanisms	Non-safety related	Local Building Code	Not Quality-Related	4.2.2	SR area
Neutron Startup Source	Non-safety related	Local Building Code	Not Quality-Related	4.2.3	SR area
Reactor Vessel System	Safety-related	SDC-3	Quality-Related	4.3	SR area
Biological Shield[2]	Safety related	SDC-3	Quality-Related	4.4	SR area
Reactor Vessel Support System	Safety-related	SDC-3	Quality-Related	4.7.3	SR area
Reactor Thermal Management System (RTMS)					
Reactor Auxiliary Heating System	Non-safety related	Local Building Code	Not Quality-Related	9.1.5	SR and NSR areas
Equipment and Structure Cooling System	Non-safety related	Local Building Code	Not Quality-Related	9.1.5	SR and NSR areas
Decay Heat Removal System (DHRS)					
DHRS components[4] except for steam vent discharge and makeup-water components	Safety-related	SDC-3	Quality-Related	6.3	SR area

SSC Name	Safety Classification	Seismic Design	Quality Program	SAR Section	Plant Area
DHRS steam vent discharge outside the safety-related portion of the Reactor Building	Non-safety related	Local Building Code	Not Quality-Related	6.3	NSR area[3]
DHRS Make-up Water SSCs	Non-safety related	Local Building Code	Not Quality-Related	6.3	SR and NSR areas
Pebble Handling and Storage System (PHSS)					
New Pebble Insertion SSCs	Non-safety related	Local Building Code	Not Quality-Related	9.3	SR and NSR areas
Pebble Extraction Machine	Non-safety related	Local Building Code	Not Quality-Related	9.3	SR area
Pebble Processing SSCs	Non-safety related	Local Building Code	Not Quality-Related	9.3	SR area
Pebble Inspection SSCs	Non-safety related	Local Building Code	Not Quality-Related	9.3	SR area
Debris Removal SSCs	Non-safety related	Local Building Code	Not Quality-Related	9.3	SR and NSR areas
Pebble Insertion Machine	Non-safety related	Local Building Code	Not Quality-Related	9.3	SR area
Full Core Offload and Spent Fuel Storage Rack	Safety-related	SDC-3	Quality-Related	9.3	SR area
Canister Transporter	Non-safety related	Local Building Code	Not Quality-Related	9.3	SR area
Spent Fuel Air Cooled Storage Rack	Safety-related	SDC-3	Quality-Related	9.3	SR area
Spent Fuel Storage Canisters	Non-safety related	Local Building Code	Not Quality-Related	9.3	SR area
Primary Heat Transport System (PHTS)					
Primary Salt Pump	Non-safety related	Local Building Code	Not Quality-Related	5.1.1	SR area

SSC Name	Safety Classification	Seismic Design	Quality Program	SAR Section	Plant Area
Intermediate Heat Exchanger	Non-safety related	Local Building Code	Not Quality-Related	5.1.1	SR area
Primary Loop Piping System	Non-safety related	Local Building Code	Not Quality-Related	5.1.1	SR area
Primary Loop Thermal Management	Non-safety related	Local Building Code	Not Quality-Related	5.1.1	SR area
Reactor Coolant	Safety-related	N/A	Quality-Related	5.1.1	SR area
Anti-Siphon Feature	Safety-related	SDC-3	Quality-Related	5.1.1	SR area
Intermediate Heat Transport System					
Intermediate Salt Pumps	Non-safety related	Local Building Code	Not Quality Related	5.2	NSR area
Intermediate Piping System	Non-safety related	Local Building Code	Not Quality Related	5.2	SR and NSR area
Superheater	Non-safety related	Local Building Code	Not Quality Related	5.2	NSR area
Intermediate Loop Auxiliary Heating System	Non-safety related	Local Building Code	Not Quality Related	5.2	SR and NSR area
Intermediate Inert Gas System	Non-safety related	Local Building Code	Not Quality Related	5.2	SR and NSR area
Intermediate Coolant Inventory System	Non-safety related	Local Building Code	Not Quality Related	5.2	SR and NSR area
Intermediate Coolant, Chemistry Control System	Non-safety related	Local Building Code	Not Quality Related	5.2	SR and NSR area
Intermediate Coolant	Non-safety related	N/A	Not Quality Related	5.2	SR and NSR area

SSC Name	Safety Classification	Seismic Design	Quality Program	SAR Section	Plant Area
Reactor Auxiliary Systems					
Chemistry Control System	Non-safety related	Local Building Code	Not Quality-Related	9.1.1	SR area
Inert Gas System	Non-safety related	Local Building Code	Not Quality-Related	9.1.2	SR and NSR areas
Tritium Management System	Non-safety related	Local Building Code	Not Quality-Related	9.1.3	SR and NSR areas
Inventory Management System	Non-safety related	Local Building Code	Not Quality-Related	9.1.4	SR area
Instrumentation and Control Systems					
Reactor Protection System, including field sensors, cabinets and associated wiring except for Cabling to the RPS devices and manual reactor trip switches	Safety-related	SDC-3	Quality-Related	7.1 7.5	SR area
Cabling to the RPS trip devices and manual reactor trip switches	Non-safety related	Local Building Code	Not Quality-Related	7.3	SR and NSR areas
Plant Control System, including field sensors, cabinets and associated wiring	Non-safety related	Local Building Code	Not Quality-Related	7.2 7.5	SR and NSR areas
Main Control Room	Non-safety related	Local Building Code	Not Quality-Related	7.4	Auxiliary Building
Remote Onsite Shutdown Panel	Non-safety related	Local Building Code	Not Quality-Related	7.4	SR area

SSC Name	Safety Classification	Seismic Design	Quality Program	SAR Section	Plant Area
Power Generation Systems					
Turbine Generator System	Non-safety related	Local Building Code	Not Quality-Related	9.9.1	Turbine Building
Steam System	Non-safety related	Local Building Code	Not Quality-Related	9.9.2	NSR areas and Turbine Building
Condensate and Feedwater System	Non-safety related	Local Building Code	Not Quality-Related	9.9.3	NSR areas and Turbine Building
Plant Auxiliary Systems					
Remote Maintenance System	Non-safety related	Local Building Code	Not Quality-Related	9.8	SR and NSR areas
Fire Protection System	Non-safety related	Local Building Code	Not Quality-Related	9.4	SR and NSR areas
Radioactive Waste Handling Systems	Non-safety related	Local Building Code	Not Quality-Related	11.2.2	SR and NSR areas
Physical Security System	Non-safety related	Local Building Code	Not Quality-Related	12.8	SR and NSR areas
Spent Fuel Cooling System	Non-safety related	Local Building Code	Not Quality-Related	9.8	SR and NSR areas
Plant Water Systems	Non-safety related	Local Building Code	Not Quality-Related	9.7	SR and NSR areas
Compressed Air System	Non-safety related	Local Building Code	Not Quality-Related	9.8	SR and NSR areas
Radiation Monitoring System	Non-safety related	Local Building Code	Not Quality-Related	11.1	SR and NSR areas
Reactor Building HVAC System	Non-safety related	Local Building Code	Not Quality-Related	9.2.3	SR and NSR areas
Reactor Building Crane and Rigging	Non-safety related	Local Building Code	Not Quality-Related	9.8	NSR area

SSC Name	Safety Classification	Seismic Design	Quality Program	SAR Section	Plant Area
Auxiliary Site Services	Non-safety related	Local Building Code	Not Quality-Related	9.8	NSR area
Plant Communications System	Non-safety related	Local Building Code	Not Quality-Related	9.5	SR and NSR areas
Electrical Systems					
Electrical Systems	Non-safety related	Local Building Code	Not Quality-Related	8.2	SR and NSR areas
Backup Power Systems	Non-safety related	Local Building Code	Not Quality-Related	8.3	SR and NSR areas
Civil Structures					
Safety-Related Portion of the Reactor Building	Safety-related	SDC-3	Quality-Related	3.5	SR area
Non-Safety Related Portion of the Reactor Building	Non-safety related	Local Building Code	Not Quality-Related	3.5	NSR area
Plant Site, including Auxiliary Buildings and the Access Building	Non-safety related	Local Building Code	Not Quality-Related	3.5.1	Site

Notes:

1. "SR area" for the purposes of this table means the safety-related portion of the Reactor Building.
2. The shielding function of the primary and secondary biological shield is not safety-related, however, the structure itself is a safety-related element for Reactor Building structural support and external event protection reasons.
3. "NSR area" for the purposes of this table means the non-safety related portion of the Reactor Building.
4. Includes the water storage tank.
5. As stated in Section 3.4.2, local building code for the Hermes 2 site is the 2012 IBC which refers to ASCE/SEI 7-10.

Table 3.6-2: Design and Construction Codes and Standards for Fluid Systems

Components	**Safety-Related** (Note 1)	**Non-Safety Related, Containing Radioactive Materials** (Note 2)	**Non-Safety Related, Not Containing Radioactive Materials** (Note 3)
Pressure Vessels	ASME Code, Section III, Division 5, Class A or B (Reference 4) (Note 6)	ASME Code, Section VIII, Division 1 or ASME Code, Section VIII, Division 2 (Reference 5)	Local Building Code
Piping and Valves	ASME Code, Section III, Division 5, Class A or B (Reference 4) (Note 6)	ANSI/ASME B31.1/B31.3 (References 6 and 7) (Note 4 and Note 5)	
Pumps	N/A	Manufacturers' standards or API-610, API-674, API-675 (References 8, 9, and 10)	
Atmospheric Storage Tanks	N/A	API-650 (Reference 11)	
Storage Tanks	ASME Code, Section III, Division 5, Class A or B (Reference 4) (Note 6)	API-620 (Reference 12)	
Core Support Structures	ASME Code, Section III, Division 5, Subsection HG/HH (Reference 4) (Note 6)	N/A	N/A

Notes:

1. The only safety-related fluid containing components in the KP-FHR are the reactor vessel, including the upper and lower heads, nozzles and primary salt pump well, and the Decay Heat Removal System components, including the storage tanks, thermosyphon thimbles, and thimble feedwater lines.
2. Only applicable to SSCs whose failure has the potential to exceed 100 mrem TEDE at the site boundary.
3. This column includes non-safety related systems that contain no radioactive material or non-safety related systems that do not contain enough radioactive material to have a potential to exceed 100 mrem TEDE at the site boundary.
4. Piping Systems are to be designed as category "M" systems if the system processes radioactive material in excess of the A2 quantities given in Appendix A to 10 CFR Part 71.
5. ASME BPVC Section II applied only to pressure retaining components.

6. Components will be designed and fabricated using the technical guidance in ASME Code, Section III, Division 5, with departures. Specifically, an ANSI/ANS 15.8 Quality Assurance Program will be implemented, as described in Section 12.9 rather than the NQA-1 standard specified in the ASME code. Therefore, the components will not meet ASME Code, Section III, Division 5 requirements that are dependent on or tied specifically to an NQA-1 program. Appropriate departures will be taken to the quality assurance related guidance of the ASME Code requirements for Hermes 2 components, including stamping and certification requirements in the Code that are dependent on implementation of an NQA-1 program. Departures from other ASME Code requirements, if any, will be identified and justified with the Operating License Application.

Kairos Power

Chapter 4

Reactor Description

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 4 REACTOR DESCRIPTION

4.1 SUMMARY DESCRIPTION

The reactor is designed with a functional capability to achieve a rated thermal power of up to 35 MWth at a reactor outlet temperature of 650°C. The normal reactor inlet temperature is 550°C. The reactor design employs a high-temperature graphite-matrix coated tri-structural isotropic (TRISO) particle fuel and a chemically stable, low-pressure molten fluoride salt coolant (Flibe). TRISO fuel and Flibe constitute the functional containment, which is relied on as a means of retaining fission products and preventing radionuclide release to the environment during normal operations and postulated events.

This chapter provides a description of the reactor, which includes:

- Reactor Core (Section 4.2)
    - Reactor Fuel (Section 4.2.1)
    - Reactivity Control and Shutdown System (Section 4.2.2)
    - Neutron Startup Source (Section 4.2.3)
- Reactor Vessel and the Reactor Vessel Internals (Section 4.3)
- Biological Shield (Section 4.4)
- Nuclear Design (Section 4.5)
- Thermal Hydraulic Design (Section 4.6)
- Reactor Vessel Support System (Section 4.7)

The reactor generates heat by the controlled fission of material contained within the TRISO fuel. The reactor transfers heat to the reactor coolant and provides for circulation of reactor coolant through the reactor core. Control elements are provided to control the reactivity of the core. A separate and independent set of shutdown elements provides for safe shutdown of the reactor during off-normal conditions. A neutron source is provided during initial pre-critical operations to assist with initial startup of the reactor core. The online refueling capability of the reactor compensates for changes in reactivity due to depletion of fuel and accumulation of fission products. The design of the reactor vessel and internals ensures that a coolable geometry is maintained for the reactor core 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 for in-service inspection, maintenance, and replacement activities. Shielding is included to limit radiation doses to workers and equipment.

Table 4.1-1 provides a summary of key parameters for the reactor.

Table 4.1-1: Reactor Parameters

Parameter	Value
Thermal Power (MWth)	35
Reactor Outlet Temperature (°C)	650
Reactor Inlet Temperature (°C)	550
Reactor Vessel Operating Pressure (bar)	< 2
Reactor Coolant Type	Flibe
Fuel Type	TRISO particle; UCO kernel
Fuel Matrix	Pebble
Equilibrium Fuel Enrichment (wt%)	≤ 19.75
Reflector Type	ET-10 Graphite
Control Material	B_4C
Neutron Spectrum	Thermal

4.2 REACTOR CORE

This section provides a description of the reactor core, including the reactor fuel, reactivity control and shutdown system, and neutron startup sources.

4.2.1 Reactor Fuel

This section describes the fuel design, the qualification of the fuel, and the design bases that the fuel must meet. In addition, an overview of fuel manufacturing is provided along with a testing and inspection plan. The fuel particle is a key component of the functional containment and the fuel, along with the reactor coolant, provide the credited barriers to the release of radioactivity to the environment.

4.2.1.1 Description

The KP-FHR fuel consists of tri-structural isotropic (TRISO) fuel particles embedded in a carbon matrix pebble. Extensive testing and operating experience over many decades as well as the more recent U.S. Department of Energy (DOE) Advanced Gas Reactor (AGR) testing program have demonstrated the robust nature and low failure rate of the TRISO fuel particle that is used in the fuel design (see "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance" topical report, Reference 1).

One of the key safety-related functions of the fuel is to provide the primary barriers that establish the functional containment.

The functional requirements for the fuel are to:

- Contain and confine actinides and fission products
- Maintain the physical form and geometry of the pebble without damage to the TRISO particles during operation, storage, shipping, and handling in the PHSS (see Section 9.3)
- Maintain net positive buoyancy in the coolant for normal operation and postulated events
- Prevent chemical interaction from reactor coolant

The fuel design relies primarily on the multiple barriers within the TRISO fuel particles to ensure that the radiological dose as a consequence of postulated events meets regulatory limits. The TRISO fuel particle design provides four of the five credited safety-related fission product barriers to the release of radioactivity from the reactor, which constitute the functional containment (see Section 6.2). These four barriers are the fuel kernel itself, the inner pyrolytic carbon (IPyC) layer, the silicon carbide (SiC) layer, and the outer pyrolytic carbon (OPyC) layer. The fuel kernel and the SiC layer are the most important of these fuel barriers to the release of radioactivity. In addition to the barriers, the TRISO particles contain a porous carbon buffer layer located between the kernel and IPyC layer, which provides a void volume to accommodate fission gases and limit pressure buildup. The secondary barrier credited in the KP-FHR functional containment for fuel in the reactor core is the reactor coolant, which is discussed in Section 5.1.

The fuel design consists of TRISO-coated particles embedded in an annular shell within a spherical pebble to form a fuel element. The particle is shown in Figure 4.2-1 and the pebble and particle are shown in Figure 4.2-2. The design of the annular pebble is similar to the traditional German pebble design used in pebble bed gas-cooled reactors that was developed in the 1960s and improved in the 1970s and 1980s.

The fuel pebble is 40mm in diameter and has three regions with specific functions that complement the design. The innermost portion of the fuel pebble is a low-density carbon matrix core. The function of the matrix core is to make the pebble buoyant in the reactor coolant during normal operation and postulated events. A fuel annulus is placed around the surface of the sub-dense inner carbon matrix core. A fuel-free carbon matrix shell is located on the surface of the fuel region to protect the fuel primarily from mechanical damage.

The fuel annulus is composed of a carbon matrix embedded with TRISO fuel particles with a packing fraction of approximately 37%. The fuel particles are located near the pebble surface, which reduces particle temperatures relative to non-annular designs. The TRISO particles are fabricated in accordance with a fuel specification that is similar to DOE's AGR program fuel particles matching critical parameters related to fuel performance. The kernels are composed of UCO, a mixture of UO_2, UC, and UC_2 phases, which differs from the traditional TRISO fuel particle kernels containing only UO_2. The addition of carbon to the kernel mitigates the generation of CO gas thus reducing the risk of kernel migration, over-pressurization of the particle with CO gas, and CO gas reactions with the SiC layer. The fuel pebbles are safety-related.

The reactor also contains moderator pebbles. These pebbles have the same diameter as the fuel pebbles, contain no uranium, and are made entirely of the same graphite matrix material that is used in the fuel pebbles and there is no inner low density core. The moderator pebbles have the same buoyancy characteristics as the fuel pebbles. As described in Section 4.5, these pebbles provide neutron moderation. The moderator pebbles are non-safety related. The moderator pebbles will be tested using the methodology in the Fuel Qualification topical report (Reference 2) for buoyancy, wear, impact, and salt infiltration. In addition, the moderator pebbles will be subject to the inspection for physical damage as described in Section 4.2.1.7.

Typical fuel properties are provided in Table 4.2-1 (particle) and Table 4.2-2 (pebble). The primary safety-related functions performed by each of the fuel components are described in Table 4.2-3.

4.2.1.2 Fuel Qualification

The qualification of the initial reactor fuel is based on U.S. and international historical experience with TRISO fuel elements and the advancement in fuel technology through the DOE AGR program. This historical experience provides confidence that the reactor will operate with large thermal margins and therefore the integrity of the fuel is not expected to be challenged. The DOE initiated the AGR project in the early 2000s to design and develop a High Temperature Gas Reactor to support the U.S. domestic electricity and process heat market. A critical part of this effort was evaluating past issues with U.S. manufactured particle fuel in comparison to the successful German experience. The result was a TRISO fuel particle design that was fabricated at laboratory and engineering scales and irradiated in a series of tests in the Advanced Test Reactor at the Idaho National Laboratory (INL). These irradiation tests serve as a foundation for the qualification of a TRISO fuel particle design for application in the KP-FHR test reactor.

The fuel qualification program is described in the "Fuel Qualification Methodology for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR)" topical report (Reference 2). The main elements of this qualification program are:

- DOE AGR and Legacy Data
- Fuel Specification, Manufacturing, and Quality Control through Inspection
- Fuel Element Phenomena Identification and Ranking Table
- Development of Operating Envelope
- Fuel Element Laboratory Testing
- Fuel Irradiation Test Program
- Fuel Performance Modelling

- Fuel Surveillance Program

The AGR test experience provides confidence that TRISO particle failure fractions in particles manufactured to similar specifications in a quality-controlled program will result in similar, very low failure fractions.

The fuel element specification describes the pebble design requirements that the fuel manufacturer must meet to ensure the level of fuel performance consistent with AGR program results. This fuel specification is based upon the fuel specification developed as part of the AGR program. The fuel element is demonstrated to meet specifications through a quality assurance program.

A fuel element phenomena identification and ranking table (PIRT) exercise was performed by Kairos Power with industry subject matter experts. The purpose was to evaluate fuel element phenomena against the figures of merit, which are fuel failure and fission product transport and release from the fuel to the coolant.

The fuel operating envelope is the range of conditions that the fuel will be expected to experience during normal operation and postulated events. The fuel operating envelope is bounded by the fuel qualification envelope. The limits of the fuel qualification are based on the AGR program and are provided in Table 4.2-5.

The reactor will operate within the TRISO particle burnup, and upper temperature bounds of the AGR-1 and AGR-2 irradiation testing. The AGR program data, combined with laboratory testing of the annular fuel pebble and a fuel surveillance program to monitor the fuel during startup and operations, completes the fuel qualification program. No specific additional irradiation qualification test is required for the test reactor for startup.

A fuel element laboratory test program is conducted to improve knowledge in areas identified in the PIRT related to the annular fuel pebble. The areas primarily relate to pebble integrity, buoyancy, and material compatibility. These areas are investigated through mechanical testing, molten-salt and inert gas tribology, buoyancy tests, and material compatibility testing of the pebble in salt and air environments.

Fuel performance is analyzed with the Kairos Power KP-BISON code to understand the response of the fuel and to determine the fuel failure fraction and fission product release. The methodology for KP-BISON fuel performance analysis is provided in the “KP-FHR Fuel Performance Methodology” topical report (Reference 3). This topical report provides limitations on the enrichment, particle power, burnup, fast neutron fluence, IPyC and SiC temperature, fuel temperature, and PyC density. Use of KP-BISON for the test reactor meets these limitations.

A fuel surveillance program is conducted during reactor operation to further confirm that fuel performance behavior remains within expectations during service life. This program monitors the fuel during startup and initial operations by assessing pebbles for burnup and physical condition of the annular pebble form. The monitoring will continue during full power operation.

The limitations in the fuel qualification topical report (Reference 2) relating to the fuel design and the operating envelope are met. The demonstration that the fuel meets the conditions and limitations of the NRC safety evaluation for the “Electric Power Research Institute – Safety Evaluation for Topical Report, Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP)” (Reference 4) will be provided as part of the application for an Operating License.

4.2.1.3 Fuel Manufacturing

This section provides a high-level overview of the particle and pebble manufacturing process.

4.2.1.3.1 TRISO Particle Manufacturing

The manufacturing process for the fuel particles is similar to the process used by the DOE AGR program. The manufacturing process begins with the fuel kernel, which is fabricated using a sol-gel process starting from the source material. The sol-gel process involves creating an aqueous broth which is dropped from a vibrating nozzle to form droplets that solidify into microspheres. The kernel microspheres are aged, washed, and dried. The kernels are then calcined and reduced with a final sintering step to obtain a high density (> 95% of theoretical).

The kernels are coated in a fluidized bed using a chemical vapor deposition process to apply the buffer, IPyC, SiC, and OPyC coating layers. There are four coating process steps, one for each of the coating layers. The buffer layer is formed using acetylene gas in an argon carrier gas. The IPyC and OPyC coatings are formed using a mixture of propylene and acetylene gas in an argon carrier gas. In these process steps, the organic compound decomposes through heating and coats the particle with solid pyrolitic carbon. The SiC layer is formed by the chemical vapor deposition process using methyltrichlorosilane gas in a hydrogen carrier gas. All layers are applied in an uninterrupted continuous process in the same coater.

4.2.1.3.2 Fuel Pebble Manufacturing

The fuel pebbles are fabricated in a three-step pressing and molding process. The inner, porous, lower-density, carbon core is formed first. The second step is the forming of the fuel region shell from the overcoated TRISO particles around the inner core. The TRISO particles are overcoated with carbon material and then pressed into the mold for the fuel region. The overcoated particles are pressed such that there is a minimum inter-spacing between the particles. The third step is the forming of the fuel free carbon outer shell from carbon matrix material.

4.2.1.3.3 Quality Control and Inspection

A quality control program for fuel manufacturing is implemented in the fuel manufacturing process with the quality of TRISO fuel particles and pebbles being maintained through inspection demonstrating that the fuel specification is met.

4.2.1.4 Fuel Design Bases

The fuel design bases are as follows:

Consistent with principal design criteria (PDC) 10, the fuel is designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits (SARRDLs) are not exceeded.

Consistent with PDC 16, the fuel is designed with multiple barriers to constitute the primary portion of the functional containment which controls the release of radioactivity to the environment.

4.2.1.5 Fuel Performance

This section provides an overview of how the fuel responds to irradiation, what the primary failure modes are, an evaluation of how the fuel meets its design bases, and a description of the fuel performance code. The fuel performance methodology is described in Reference 3.

4.2.1.5.1 Behavior of Fuel During Irradiation

Neutron irradiation of a TRISO-coated particle causes its kernel to expand outward and its buffer to shrink inward. In this process, the buffer stays bonded to the kernel but tends to delaminate from the

IPyC layer. The PyC layers shrink early during irradiation and revert to swelling at longer irradiation times. As the buffer pulls away from the IPyC, a gap can be created between the two layers. Simultaneously, the buffer is pushed outward by the swelling kernel, reducing the size of the gap. The void volume of this buffer-IPyC gap adds to the increasing porosity of the kernel and decreasing porosity of the buffer to form a free volume that accommodates fission gases.

The PyC has different irradiation-induced strain rates in the radial and tangential directions because of its slightly anisotropic nature. At low fast neutron fluence, the PyC shrinks in both directions. The strain first reverses from shrinkage to swelling in the radial direction (at fast neutron fluences typically around 2×10^{25} n/m^2, $E_n > 0.18$ MeV) and then in the tangential direction (~5×10^{25} n/m^2, $E_n > 0.18$ MeV), as described in Reference 3. The change in strain behavior depends on intrinsic PyC properties (i.e., density and degree of anisotropy measured by the Bacon Anisotropy Factor or BAF) and on the irradiation temperature.

During the early phases of irradiation, the shrinkage of the PyC puts the IPyC and OPyC layers into tension and creates compressive forces on the more rigid SiC layer, as long as the PyC layers remain intact. Cracking of the PyC can occur if the tensile stress in the layer reaches its fracture strength, resulting in high local stresses on the SiC layer which can lead to SiC failure. In addition to shrinkage, the irradiation induced creep of the PyC layers offsets their shrinkage at longer irradiation times. As a consequence, some of the tensile stress in the PyC layers and some of the compressive stress in the SiC layers are relieved. Concurrently, fission gas pressure builds up in the free volume of the particle, putting the coating layers in tension as this pressure counteracts the effect of the shrinkage of the PyC layers, causing them to push or pull inward on the SiC. The IPyC, SiC, and OPyC act as structural layers to retain this pressure and also function as barriers to the migration of fission products. However, if the internal gas pressure increases enough, the tangential stress in the SiC layer can eventually become tensile. Failure is expected to occur if this stress reaches a value that exceeds the SiC fracture strength for the particle.

The dimensional changes of the SiC layer during irradiation are insignificant compared with the shrinkage, swelling, and creep of the IPyC and OPyC layers (Reference 3). Consequently, the SiC response is mostly elastic. Although some swelling of the SiC layer is anticipated during irradiation, its magnitude is small compared to the dimensional changes of the PyC layers, and it has not been observed to impact the mechanical integrity of the TRISO particle.

Failures of the fission product barriers are categorized as either a “TRISO failure” or a “SiC Failure”. TRISO failure, also referred to as an “exposed kernel”, corresponds to the loss of integrity of all three outer coating layers. Conversely, SiC failure corresponds to the loss of integrity of the SiC layer, the primary barrier to the release of fission products, with at least one remaining intact PyC layer such that fission gases are retained in the TRISO particle.

The AGR-1 and AGR-2 irradiation tests have shown that, within their operating ranges, failure of these particle layers while possible are very rare. The 95% upper confidence bound for the aggregate measured TRISO failure fraction (failure of all three layers) during AGR-1 and AGR-2 irradiations is $\leq 2.3 \times 10^{-5}$. Additionally, the aggregate measured SiC failure fraction, defined as the loss of integrity of the SiC layer with at least one remaining intact PyC layer is $\leq 3.6 \times 10^{-5}$ for AGR-1 and AGR-2, i.e., at the same low level as the TRISO failure fraction. These conclusions are documented in Reference 1.

4.2.1.5.2 Potential Fuel Failure Modes

The potential failure mechanisms of TRISO-coated fuel particles are:

- Pressure vessel failure of spherical or aspherical particles resulting in the failure of all three coating layers
- Cracking of the IPyC layer potentially leading to SiC failure
- Partial debonding of the IPyC from the SiC leading to SiC failure
- Kernel migration towards the SiC layer and its subsequent failure
- Chemical attack of the SiC layer by noble metals
- Thermal decomposition of the SiC layer at high temperatures
- Buffer fracture leading to cracking of undebonded IPyC

The failure modes relevant to UCO fuel and modeled using KP-BISON for the KP-FHR are described in Reference 3.

4.2.1.5.3 Evaluation of Fuel Performance

Fuel performance is central to the determination that the fuel meets PDC 16. Fuel performance calculations are performed using the KP-BISON code. KP-BISON is based on the BISON code which is an engineering-scale multi-dimensional finite-element based nuclear fuel performance code developed and maintained by INL. KP-BISON can model fuel in 1D-spherical, 2D-axisymmetric, or 3D geometries for both steady-state and postulated events.

KP-BISON analyzes the temperature and stress conditions within the particle to determine the state of IPyC, SiC, and OPyC layers and the failure fraction of fuel particles in the reactor core during normal operation and postulated events. In addition, KP-BISON analyzes the radiological release fraction based on expected manufacturing defects and in service failures to determine the radiological release fraction.

4.2.1.6 Evaluation of Fuel Design Bases

In compliance with PDC 16, there are four fuel barriers to the release of radioactivity that are credited in the postulated event analysis (see Chapter 13). These barriers are the TRISO fuel kernel, the IPyC, the SiC, and the OPyC layers. Extensive testing of the TRISO particles, as discussed in Reference 2, has conclusively demonstrated that these barriers are effective in retaining radionuclides and constitute an effective functional containment (which is augmented further in the KP-FHR with the reactor coolant).

A laboratory testing program provides confirmation that the fuel's physical form is maintained during operation, the pebble remains buoyant, and there is no significant salt infiltration into the pebble. These laboratory tests include mechanical testing, molten-salt and inert gas tribology, and material compatibility testing of the pebble in reactor coolant and gas environments. A conservative calculation demonstrates that wear does not exceed the pebble outer layer thickness. This result is confirmed by discrete element modeling analysis. Additional details regarding both the laboratory testing and discrete element modeling can be found in Reference 2. The results of the laboratory testing program will be provided with the application for an Operating License.

The inspection of pebbles in the PHSS (see Section 9.3) provides assurance that damaged pebbles are removed from service to ensure that limits on circulating activity are met. Monitoring of cover gas (see Section 9.1.2) and reactor coolant radioactivity ensures that there is early indication of potential fuel failures. The results and actions provide assurance that PDC 16 is met.

Fuel performance is analyzed using the KP-BISON fuel performance code to determine the temperature profile within the pebble. This temperature distribution is then used to determine the temperature distribution in the particle and the stresses within the fission product barriers. Based on the temperature and stress calculations, KP-BISON then calculates the failure fraction and radiological release, confirming that the fuel performance remains within acceptable limits. Table 4.2-6 presents fuel performance results (SiC temperature and failure probability at a 95% confidence level) for a TRISO fuel particle using the fuel performance methodology in Reference 3.

In compliance with PDC 10, the fuel design locates the fuel particles near the periphery of the fuel pebble, enhancing the ability of the fuel to transfer heat to the coolant. As shown in Table 4.2-6 the peak fuel SiC temperatures are well below the upper temperature bounds based on the AGR program. The thermal hydraulic analysis of the core (see Section 4.6) ensures that adequate coolant flow is obtained to ensure that SARRDLs, which are discussed in Section 6.2, are met.

4.2.1.7 Testing and Inspection

The cover gas and reactor coolant are monitored for circulating activity, which is an indirect measurement of TRISO failures. Circulating activity limits will be provided in the technical specifications.

Fuel pebbles are subject to examination for damage and burnup as they exit the core. Pebbles are inspected to identify damage such as wear, cracking, missing surfaces from chipping, etc. Fuel pebbles are also examined by gamma spectrometry to determine the burnup through the measurement of gamma activity from signature fission products. Pebbles approaching or at the burnup limit are not returned to the core and instead are sent to storage. Similarly, pebbles that show indications of wear, cracking, or missing surfaces are removed from service. These inspections are described in Section 9.3.

4.2.2 Reactivity Control and Shutdown System

The reactivity control and shutdown system (RCSS) provides reactivity control during normal operation and also provides shutdown of the reactor in response to abnormal conditions or postulated events. The RCSS structures, systems and components are not shared by Unit 1 and Unit 2.

4.2.2.1 Description

The RCSS includes two separate system features (means) to control reactivity in the reactor core - control elements and shutdown elements.

The non-safety related control elements are used to control the reactivity for normal operations and for planned, normal startup, and power changes in the reactor. The control elements can be positioned throughout their range of travel to support operational demands.

The shutdown elements are credited for shutting down the reactor during postulated events. The shutdown elements are located to optimize reactivity worth and to meet shutdown margin requirements. These elements have two positions, fully withdrawn or fully inserted. These elements are safety-related.

Both the control and shutdown elements are tripped automatically by the reactor protection system, or manually from the main control room or remote shutdown panel. The plant control system is used to withdraw and insert the control and shutdown elements during normal operation. The portion of the PCS that controls RCSS is not shared between units. Instrumentation and control systems are described in Chapter 7.

The shutdown and control elements have two different designs. Each control element is an assembly of segmented annular cylinders. The annular cylinders are welded to connection plates at various points along the length of the control elements. Stainless steel spines are used to connect the array of control elements together. Each shutdown element is an array of small rods arranged in a cruciform shape. The control element design is shown in Figure 4.2-3 (cross-section) and Figure 4.2-4 (side-view). The shutdown element design is shown in Figures 4.2-5 (cross-section) and Figure 4.2-6 (side-view). There are seven elements in total in the RCSS design, which is comprised of three shutdown elements and four control elements. The control elements insert into guide structures in the upper and side reflector, near the periphery of the core. The shutdown elements insert into guide structures in the upper reflector, then directly into the pebble bed. The locations of the control and shutdown elements are shown in Figure 4.2-7. The control element and shutdown element design parameters are summarized in Table 4.2-4.

The control elements are positioned via a counter-weighted winch system (Figure 4.2-8). The shutdown elements are also positioned by a counter-weighted winch, but they are typically only fully inserted or fully withdrawn. In the counter-weighted winch system, a wire-rope is connected to the element, and travels up around the sheave and down to a counter-weight. The counter-weight allows the wire-rope to wrap around the sheave without having to anchor the wire rope, similar to a capstan. The sheave, commonly known as a winch drum, is rotated by an electric motor. There is an electric clutch between the motor and the sheave. The motor allows small and controlled movements of the element. The maximum withdrawal and insertion time for the shutdown and control elements is 100 seconds over the full range of motion for motor-driven operations.

On a reactor trip, the electric clutch opens, which allows the sheave to rotate freely. With the sheave rotating freely, the shutdown and control elements are released from their drives and drop into the core and reflector, respectively, as a result of gravity. The control and shutdown elements reach full insertion by gravity in no more than 10 seconds. Although both the control elements and the shutdown elements receive a reactor trip signal, the release of the clutch for the shutdown elements provides the primary safety-related reactor trip release mechanism.

Control and shutdown element position is monitored using two independent and diverse methods. The motor position is measured using an absolute encoder allowing the determination of the angle the sheave has swept from a known reference point, which directly correlates to the element position. The second position measurement device is a high-density reed switch array. Similar to existing reed switch position measurement designs, this instrument measures the position of the counterweight over its full range of motion. The reed switch array provides an analog signal, and the encoder provides a digital signal and the two used together provides the ability to determine the element position, while allowing real time functional checks.

The materials used in the RCSS are shown in Table 4.2-4. The primary materials are the B_4C absorber material and the stainless steel 316H cladding. The operating conditions are such that the control and shutdown elements are immersed in reactor coolant and experience temperatures up to 700°C during operation. The upper portions of the control and shutdown elements are exposed to reactor cover gas above the reactor coolant free surface. The control and shutdown drive mechanisms above the vessel are maintained at temperatures below their mechanical limits. The B_4C neutron absorber material is contained in pellets, which are stacked in SS 316H cylindrical tubes (pressurized with inert gas). The control and shutdown drive mechanisms are also made of stainless steel.

4.2.2.2 Design Basis

Consistent with PDC 2, the safety-related portion of the RCSS performs the shutdown function under design basis natural phenomena events.

Consistent with PDC 4, the safety-related portion of the RCSS accommodates the effects of the environmental conditions during normal plant operation as well as during postulated events as a result of equipment failures.

Consistent with PDC 23, the safety-related portion of the RCSS fails into a safe state in the event of adverse conditions or environments.

Consistent with PDC 26, the RCSS provides an independent and diverse means of controlling reactivity to assure that shutdown margin is maintained and that SARRDLs are not exceeded under conditions of normal operation. In addition, the RCSS provides a means of inserting negative reactivity at a sufficient rate to assure with appropriate margin for malfunctions and also provide a means to maintain the reactor shutdown for fuel loading, inspection and repair.

Consistent with PDC 28, the RCSS has appropriate limits on the potential amount and rate of reactivity increase to ensure the effects of postulated reactivity events can neither damage the safety-related elements of the reactor coolant boundary or disturb the core and internals such the ability to cool the core is impaired. The system allows only one element to move at a given time.

Consistent with PDC 29, the RCSS, in conjunction with reactor protection systems, assures an extremely high probability of accomplishing its safety-related functions.

4.2.2.3 System Evaluation

The RCSS meets the design bases as described below:

PDC 2

As noted in Section 4.2.2.1, the shutdown elements are inserted into guide structures in the upper reflector and then directly into the pebble bed. The guide structures and reflector blocks ensure the ability of the shutdown elements to insert under conditions of reflector block misalignment that could potentially occur in a design basis earthquake. The design basis earthquake is described in Section 3.4. This seismic analysis determines the maximum deflection of the insertion path. Insertion capability will be assessed in a one-time, out-of-pile, at scale test prior to initial operation, with and without maximum deflection of the shutdown element guide structures consistent with the maximum misalignment caused by such an event and accounts for the expected changes in pebble bed packing fraction and concurrent insertion of all three shutdown elements into the pebble bed. The three shutdown element insertion times and insertion depths are measured and compared to the insertion time testing performed with no deflection of the upper reflector guide structures. The testing is performed to confirm that the shutdown element insertion time is within the insertion time assumed in the postulated event analysis in Chapter 13 under the condition of maximum expected misalignment of the upper reflector guide structures from a design basis earthquake. The tests will also confirm that the shutdown elements fully insert to the depth assumed in the shutdown margin calculations in Table 4.5-5. Additionally, the reflector blocks maintain the element insertion pathway as described in Section 4.3. These shutdown element design features provide conformance to PDC 2.

PDC 4

The safety-related portions of the RCSS are compatible with the environmental conditions that they will be subjected to during normal operation, maintenance, testing, and postulated events.

The RCSS shutdown elements are made with stainless steel cladding. Wear rates due to flow induced vibration are expected to be low in comparison to those of typical operating reactors with stainless steel cladding given the lower core flow rates (<0.13 meter/second) in the design. The neutron absorbing material is enclosed in two stainless steel barriers to mitigate the loss of neutron absorbing material in the shutdown elements. The shutdown elements are qualification tested out of pile prior to operation and a conservative wear limit is established to ensure that wear during shutdown element movement is acceptable. The shutdown elements can be removed for inspection or replaced if necessary. In addition, the shutdown elements are not adversely affected by neutron and gamma heating.

Analysis is performed on the shutdown elements to determine the internal gas release and swelling of the B_4C during normal operation over their design lifetime. The resulting increase in gas pressure is analyzed to ensure that stresses on the shutdown element tubes are within allowable stress limits for SS 316H. In addition, the effects of irradiation on SS 316H and clad wear are accounted for in the stress analysis.

A finite element model is developed to calculate the forces on the shutdown elements during normal operation and postulated events. This analysis includes thermal stresses from internal heat generation, is performed under maximum heat generation conditions, and demonstrates that shutdown element cladding stresses are within limits and are not subject to bowing or binding due to differential thermal expansion.

There is extensive experience (References 5, 6, and 7) with B_4C under irradiation. In addition, the B_4C melting temperature is more than 1000˚C above the Hermes operating temperatures.

The shutdown elements and drive mechanisms are also analyzed to meet ASME Section III, Division 5 (Reference 8) loads due to operational stepping, reactor trip, stuck element, fatigue, and shipping and handling. All stresses in the components of the reactivity elements are within limits.

Materials utilized in the shutdown elements are qualified for their operating environment. Materials are chosen to ensure reactor coolant induced diffusion bonding does not occur at interfaces where movement or separation is necessary.

These evaluations demonstrate conformance with PDC 4.

PDC 23

The safety-related reactor trip function of the RCSS is initiated by the reactor protection system through the reactor trip system (RTS) and is based on redundant trip determination signals to automatically open the reactor trip breakers. Removal of power from the electromagnetic clutch on the shutdown elements allows them to fall into the core by gravity. Normally open relays are utilized for this system such that during operation they are energized allowing the system to operate. When the RTS actuates, the energy holding the relays closed is removed and this loss of supply power initiates a reactor trip. The shutdown elements accomplish safe shutdown (i.e., reactor trip) via gravity insertion on a reactor trip signal; or on a loss of normal electrical power after a short time delay to mitigate spurious trips. The electrical system design is described in Chapter 8. The reactor control and reactor protection system architecture are described in Chapter 7. These features, in conjunction with Chapter 7, demonstrate conformance to PDC 23.

PDC 26

The control and shutdown elements meet the requirements of PDC 26. The compliance with the requirements in PDC 26 is discussed in Section 4.5.

PDC 28

The control elements traverse their full range of movement in 100 seconds. This maximum design speed is analyzed in Chapter 13 to ensure that the rate of reactivity addition does not impact the safety-related portions of the reactor coolant boundary and also does not disturb the core and internals and impair cooling of the core.

PDC 29

The RCSS supports a high probability of accomplishing its design function, because the trip function is safety-related and the elements are inserted via gravity. There are two means of inserting negative reactivity and these two means contain sufficient negative reactivity such that the highest worth reactivity element can fail to insert, and the function can still be achieved. The first means of inserting negative reactivity would be to use the motor to lower the element into the core region. The second means is upon a reactor trip which releases the elements, allowing them to drop into the core by gravity.

The shutdown element position and reactivity insertion versus time will be provided in the application for an Operating License. A conservative shutdown element drop time and reactivity insertion value is used in Chapter 13. These features demonstrate conformance to PDC 29 for the RCSS.

4.2.2.4 Testing and Inspection

The shutdown elements are periodically inspected to ensure that there is no unacceptable wear or other damage to the cladding that encapsulates the B_4C absorber material. In addition, the reactor coolant is periodically examined for an increase in boron from B_4C absorber material, which provides an indication of shutdown element cladding failure.

RCSS shutdown element insertion times and shutdown margin are periodically confirmed to be within safety analysis limits by surveillance requirements provided in the technical specifications (see Chapter 14).

4.2.3 Neutron Startup Source

A neutron startup source is used to provide an adequate neutron flux to the source range excore detectors during initial and subsequent plant startups. The startup neutron source allows monitoring of the change in neutron multiplication during the addition of fuel and the approach to criticality. The neutron startup source does not perform any safety-related functions.

The neutron source(s) will be located in the reflector region of the reactor near the outside edge of the core and optimally located relative to an excore source range detector for best detectability of criticality. The source will have sufficient strength to provide a detectable count rate.

The source material is encased in a metal sheath. The neutron startup source is compatible with the chemical, thermal, and irradiation conditions expected in the reflector. The neutron startup source can be removed and replaced during the life of the plant, if needed and is not shared by Unit 1 and Unit 2.

4.2.4 References

1. Electric Power Research Institute, “Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance,” Topical Report EPRI-AR(NP)-A, 3002019978, November 2020.
2. Kairos Power, LLC, “Fuel Qualification Methodology for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR),”KP-TR-011-P-A, March 2023.
3. Kairos Power, LLC, “KP-FHR Fuel Performance Methodology,” KP-TR-010-P-A, May 2022.
4. Nuclear Regulatory Commission, “Electric Power Research Institute – Safety Evaluation for Topical Report, Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP),” August 11, 2020.
5. Fryger, B., Gosset, D., & Escleine, J.M., “Irradiation Performances of the Superphenix Type Absorber Element, Absorber Materials, Control Rods and Design of Backup Reactivity-Shutdown Systems for Breakeven and Burner Cores for Reducing Plutonium Stockpiles,” 1995.

6. Pitner, A.L., & Russcher, G. E., “Irradiation of Boron Carbide Pellets and Powders in Hanford Thermal Reactors,” 1970.
7. Demars, R.V., Dideon, C.G., Thornton, T.A., Tulenko, J.S., Pavinich, W.A., & Pardue, E. B. S., “Irradiation Behavior of Pressurized Water Reactor Control Materials, Nuclear Technology,” 62(1), 75-80, 1983.
8. American Society of Mechanical Engineers, ASME Boiler & Pressure Vessel Code, Section III, Division 5, “High Temperature Reactors.” 2017.

Table 4.2-1: Fuel Particle Properties

Property	Nominal Value
Kernel diameter (µm)	425
Buffer thickness (µm)	100
PyC thickness (µm)	40
SiC thickness (µm)	35
Kernel density (g/cm^3)	≥10.4
Buffer density (g/cm^3)	1.05
PyC density (g/cm^3)	1.90
SiC density (g/cm^3)	≥3.19
C/U atomic ratio	0.40
O/U atomic ratio	1.50
PyC BAF	≤1.045

Table 4.2-2: Fuel Pebble Dimensions and Properties

Property	Nominal Value
Fuel Pebble	
Outer shell outer radius (cm)	2.0
Average Pebble Density (g/cm^3)	1.74
TRISO particles packing fraction (%)	~37
Pebble Uranium loading (g)	6.0
Number of particles per pebble	~16,000
Moderator Pebble	
Moderator Pebble Radius (cm)	2.0
Average Density (g/cm^3)	1.74

Table 4.2-3: Safety-Related Fuel Component Functions

Layer	Purpose
UCO Kernel $UO_2 + UC + UC_2$	Limits free oxygen release compared to traditional UO_2 Suppresses CO production and subsequent kernel migration, over-pressure, and corrosion of SiC Oxygen used to form less-mobile-than-carbides fission product oxides, which reduces chemical attack of SiC
Porous Carbon Buffer	Provides void volume to accommodate fission gases and limit pressure buildup Mechanically decouples kernel from outer coating layers by accommodating swelling of UCO kernel Protects IPyC from fission products recoil
IPyC	Protects kernel during SiC deposition (chlorine attack) Protects SiC from fission product attack Secondary structural layer that puts SiC in compression and reduces risk of failure Fission gas barrier
SiC	Primary structural layer Primary fission product barrier
OPyC	Secondary structural layer that puts SiC in compression and reduces risk of failure Fission gas barrier
Overcoat	Prevents particle-to-particle contact during pebble manufacturing
Inner Core	Reduces peak fuel temperature by placing TRISO particles close to pebble’s edge Lowers overall density of the pebble and allows buoyancy in reactor coolant
Outer Shell	Protects TRISO fuel particles from potential in-service pebble-to-pebble damage and during fuel handling

Table 4.2-4: Reactivity Control and Shutdown Element Parameters

Parameter	Control Elements	Shutdown Elements
Number of Control Elements	4	3
Location	Reflector near core periphery	In-bed
Drive Mechanism	Counter-weighted Winch	Counter-weighted Winch
Release Mechanism	Electric Clutch Motor Electrical Isolation	Electric Clutch Motor Electrical Isolation
Absorber Material	B_4C	B_4C
Absorber Clad	Stainless Steel 316H	Stainless Steel 316H
Element Geometry	Rectangular	Cruciform
Absorber Material Length	70 inches	96 inches

Table 4.2-5: Fuel Qualification Envelope

Parameter	Existing TRISO Particle Qualification Envelope
Peak Fuel SiC Temperature – Normal Operation (°C)	1360
Peak Fuel SiC Temperature - Transient (°C)	1600
Burnup (%FIMA)	13.2
Peak Particle Power (mW)	155
Peak Fluence ($x10^{25} n/m^2$, E>0.1MeV)	3.8

Table 4.2-6: Fuel Performance Results for Normal Operation

Parameter	**Value**
Peak SiC Temperature (°C)	< 830
Bounding Trajectory (95% CL)[(1)]	
SiC Failure	$< 2.3x10^{-3}$
Contribution from IPyC Cracking	$< 2.3x10^{-3}$
Contribution from Pd Penetration	$< 3.6x10^{-6}$
TRISO Failure (over-pressure)	$< 3.6x10^{-6}$

Notes:

1. Calculated using fuel performance methodology described in Reference 2.

Figure 4.2-1: Fuel Particle

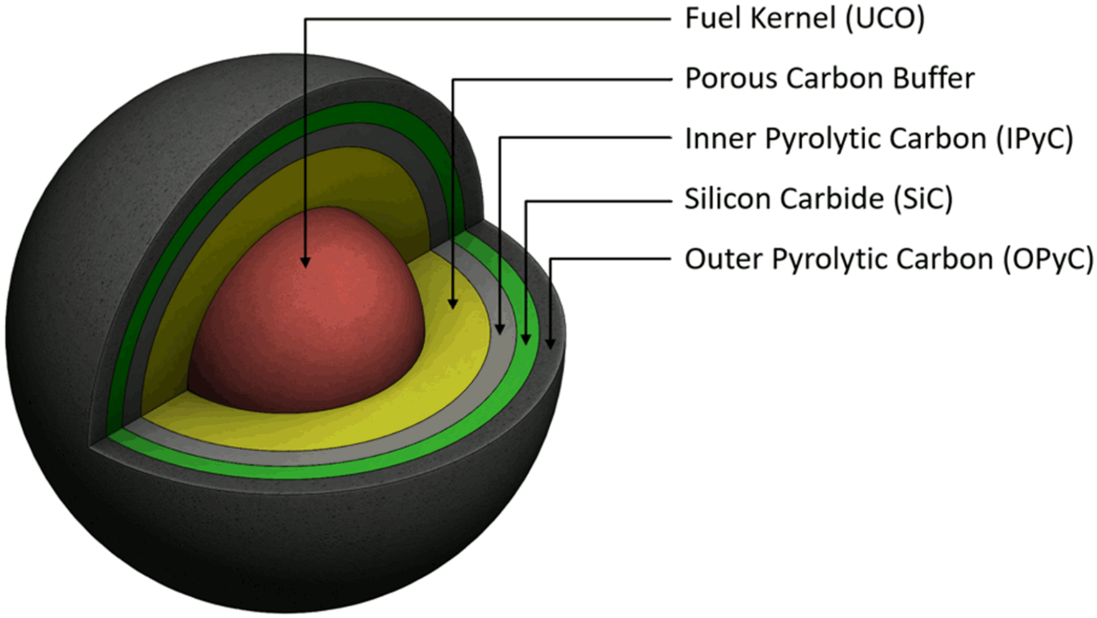


Figure 4.2-2: Fuel Pebble and Particle

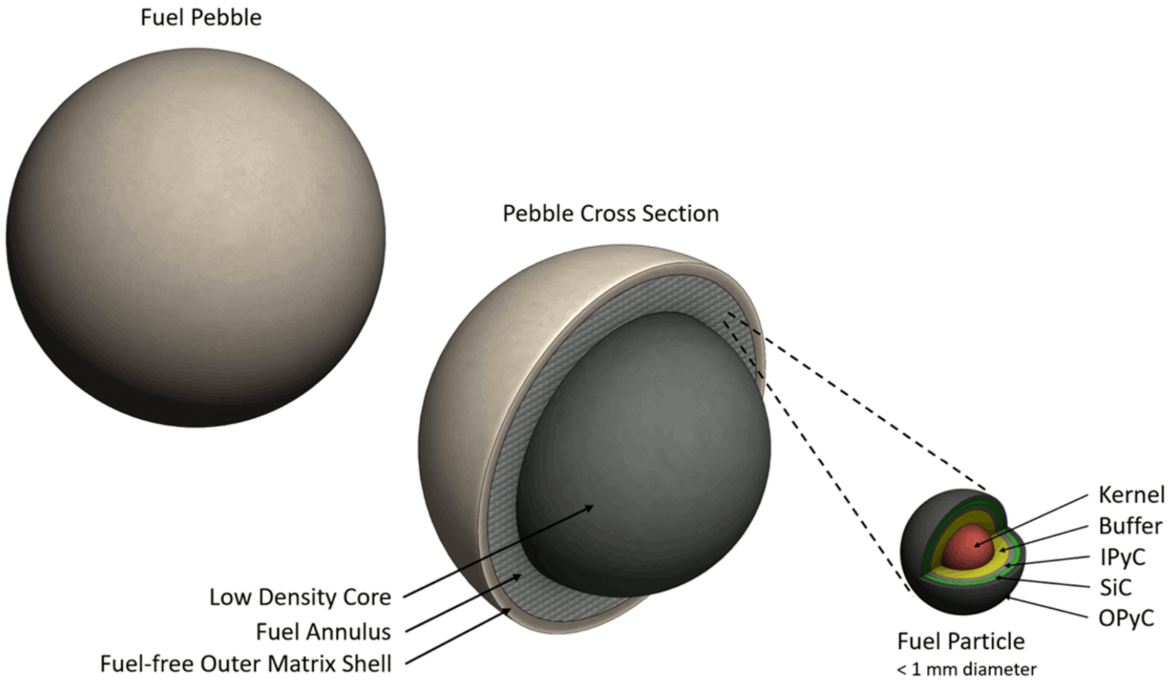


Figure 4.2-3: Control Element Cross-section

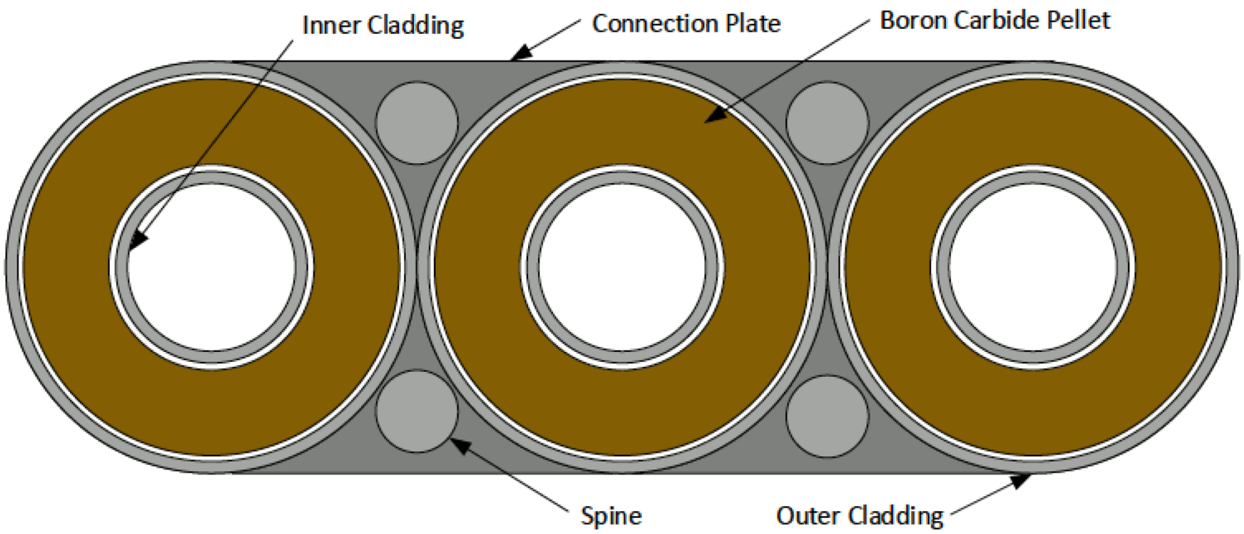


Figure 4.2-4: Control Element Side View

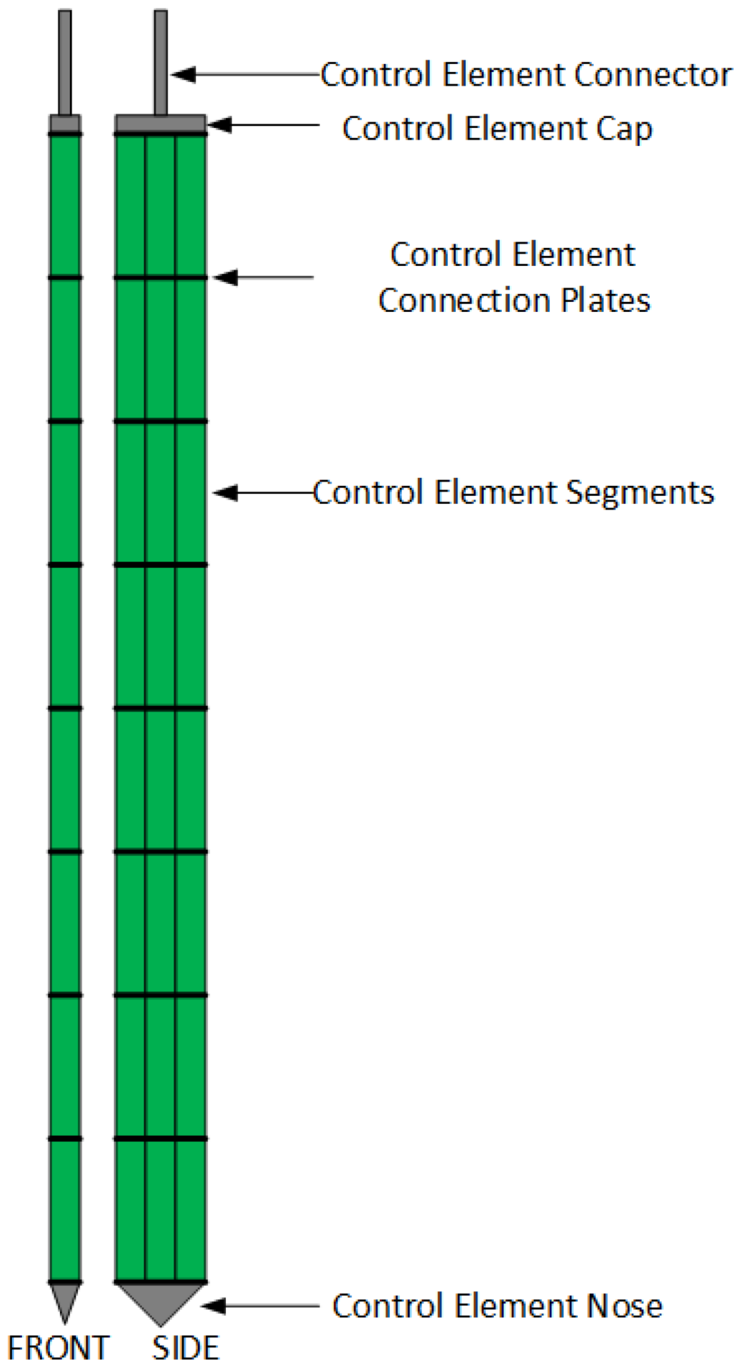


Figure 4.2-5: Shutdown Element Cross-section

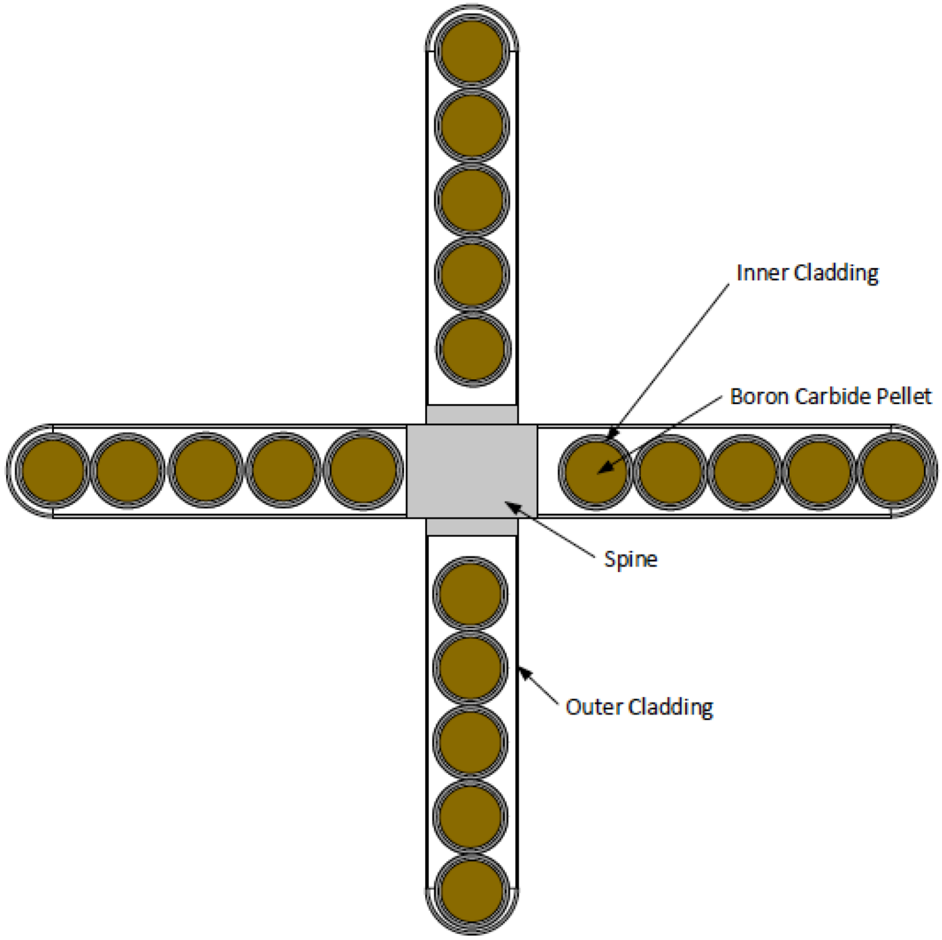


Figure 4.2-6: Shutdown Element Side View

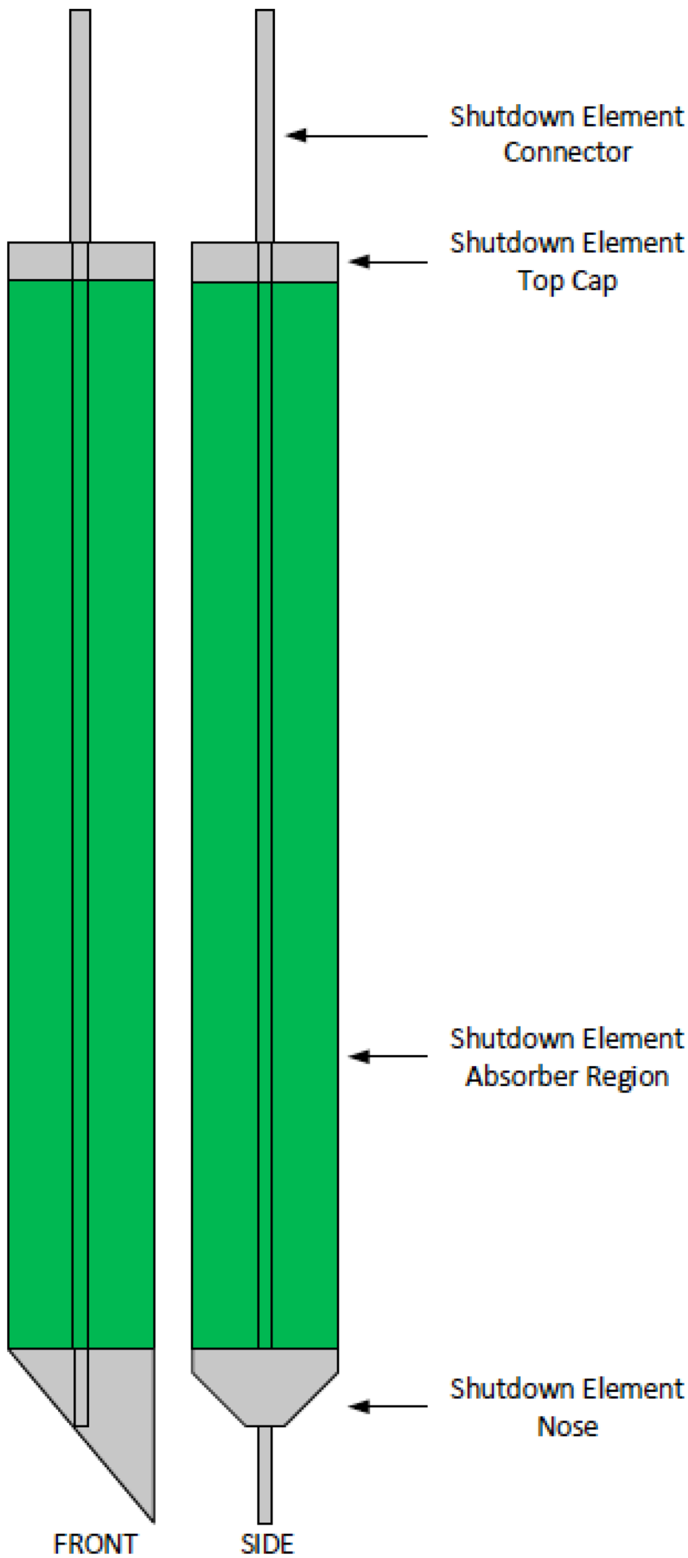


Figure 4.2-7: Control and Shutdown Element Locations

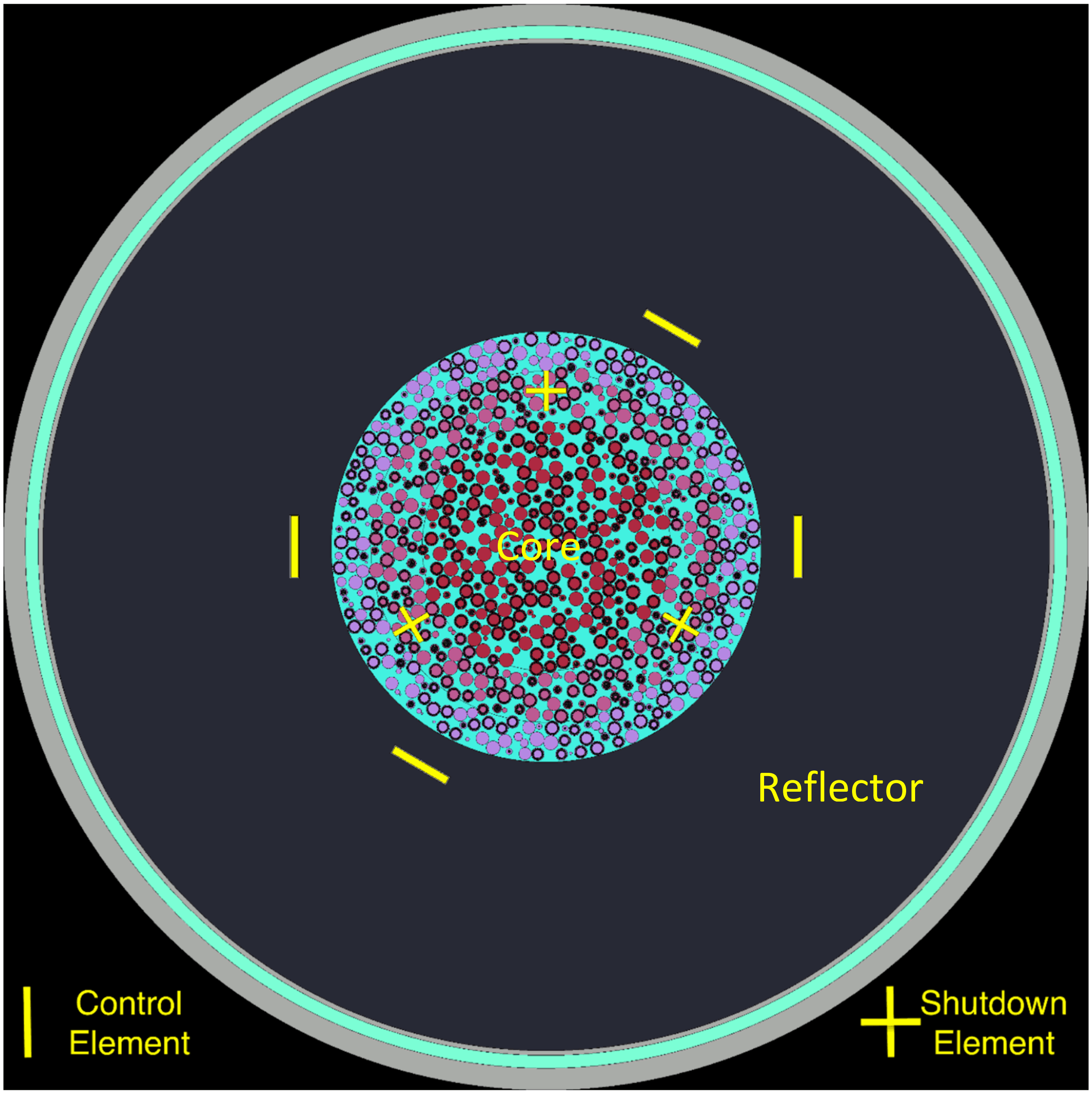
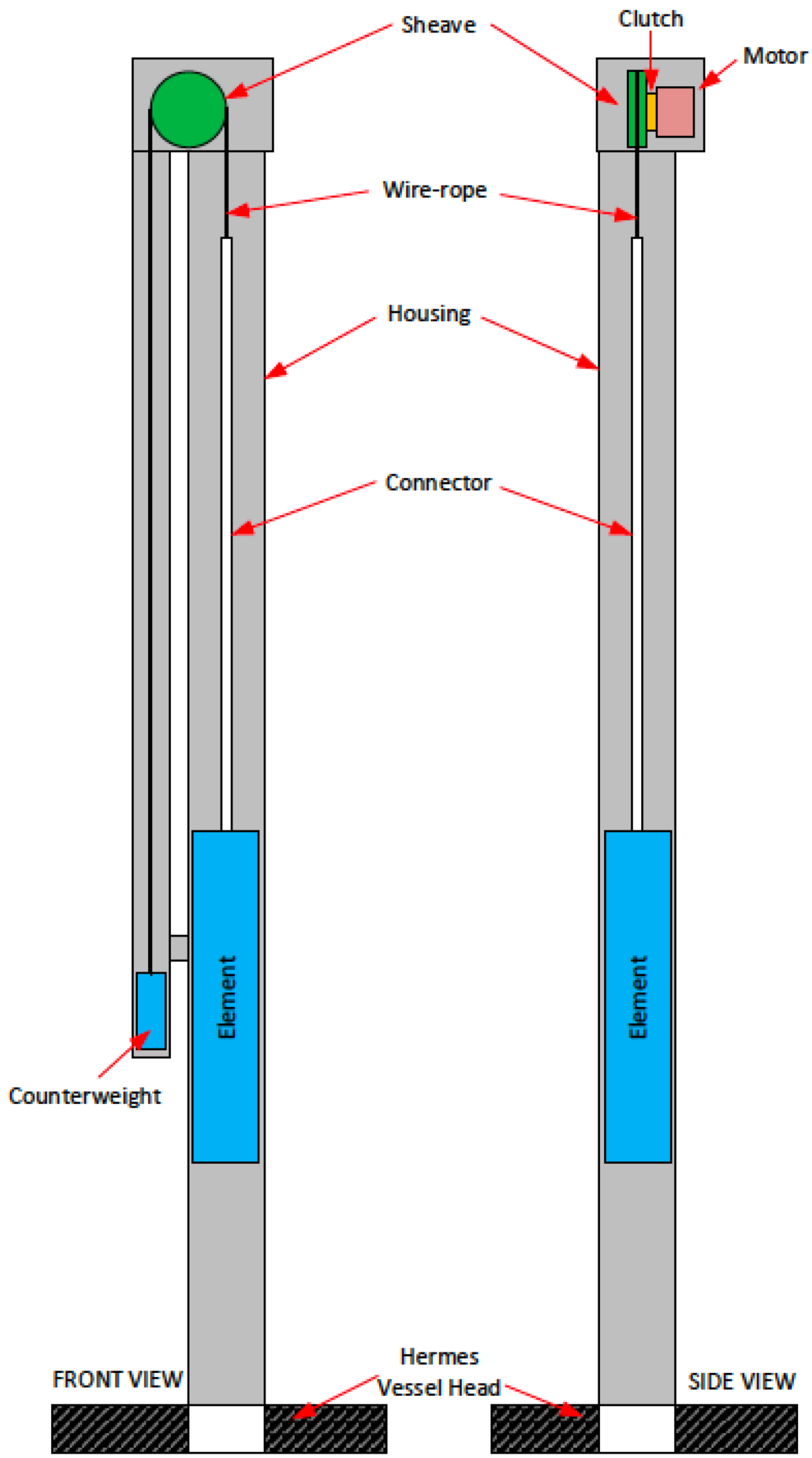


Figure 4.2-8: Counter-weighted Winch Drive Mechanism



4.3 REACTOR VESSEL SYSTEM

4.3.1 Description

This section provides an overview of the reactor vessel system (see Figure 4.3-1), which includes the reactor vessel and the reactor vessel internals. The reactor vessel forms a major element of the reactor coolant boundary and the inert gas boundary. The reactor vessel and vessel internals define the flow path for reactor coolant and fuel into the core. The reactor vessel system contains the reactor core and provides for circulation of reactor coolant and pebbles as well as insertion of the reactivity control and shutdown elements through the reactor core.

The reactor vessel system provides a flow path for reactor coolant to transfer heat from the reactor core to the primary heat transport system (PHTS) during normal operations. The reactor coolant enters the reactor vessel through two side inlet nozzles and flows downward through a downcomer annulus formed between the metallic core barrel and the reactor vessel shell. Coolant flow moves through the vessel bottom plenum formed by the reflector support structure and is distributed into the core by the design of the reflector blocks. Upon exiting the core, the coolant leaves the reactor vessel via the primary salt pump (PSP) (see Section 5.1.1), which draws suction directly from a pool of reactor coolant above the core and inside the vessel. Design features are provided in fluid systems connected to the reactor vessel to limit loss of coolant inventory in the event of a break in those systems as described in Sections 5.1, 9.1.4, and 9.3.

The reactor vessel system also provides a flow path for pebbles to allow online refueling and defueling of the reactor core by the pebble handling and storage system (PHSS) (Section 9.3) during normal operation. The PHSS inserts pebbles into the reactor vessel and delivers them to the fueling chute below the reactor core by the pebble insertion line (Section 9.3.1). The buoyant pebbles float upward, and pebbles inserted via the insertion line will join the packed pebble-bed in the reactor core. Upon circulating through the core, the pebbles accumulate in the de-fueling chute at the top of the reactor core. The pebble extraction machine (PEM) (Section 9.3.1) at the top of the reactor core removes pebbles from the reactor vessel (see Figure 4.3-2.)

During postulated events when the PHTS and the intermediate heat transport system (IHTS) are not available, the reactor vessel provides an alternative flow path as discussed in Section 4.6.1 to allow natural circulation of the reactor coolant to remove heat from the reactor core. The reactor coolant leaving the core flows into the hot well, fluidic diode pathway, fluidic diode, through a core barrel penetration, and back into the downcomer annulus as shown in Figure 4.3-1. The heat from the core is transferred to the reactor vessel shell, which transfers the heat to the decay heat removal system (DHRS) (Section 6.3).

The reactor vessel system interfaces with fuel (Section 4.2.1), primary heat transport system (PHTS) (Section 5.1), reactivity control and shutdown system (RCSS) (Section 4.2.2), reactor vessel support system (RVSS) (Section 4.7), decay heat removal system (DHRS) (Section 6.3), pebble handling and storage system (PHSS) (Section 9.3), reactor thermal management system (RTMS) (Section 9.1.5), inert gas system (IGS) (Section 9.1.2), inventory management system (IMS) (Section 9.1.4), and instrumentation and controls (Chapter 7).

4.3.1.1 Reactor Vessel

The reactor vessel is a vertical cylinder design with flat top and bottom heads. The vessel houses the reactor vessel internals. The reactor vessel shell and bottom head provide a major element of the reactor coolant boundary. The vessel is constructed of 316H stainless steel (SS) with ER16-8-2 weld metal and is designed and fabricated using the technical guidance in ASME BPVC Section III, Division 5

(Reference 1) as shown in Table 3.6-2. It contains the inventory of reactor coolant such that the reactor core is covered by the coolant during normal operation and postulated event. There are no penetrations or attachments to the vessel below the coolant level. The design of the reactor vessel allows for online monitoring, in-service inspection, and maintenance.

4.3.1.1.1 Vessel Top Head

The reactor vessel top head (see Figure 4.3-2) is a flat 316H SS disc bolted and flanged to the vessel shell. This interface is designed for leak-tightness but is not credited as being leak tight in safety analyses. The vessel top head controls the radial and circumferential positions of the reflector blocks to ensure a stable core configuration for all conditions (e.g., reactor trip and core motion). The top head contains penetrations, as shown in Figure 4.3-2 and Table 4.3-1, into and out of the vessel and provides for the attachment of supporting equipment and components (e.g., reactivity control elements, reactivity shutdown elements, pebble handling and storage system components, material sampling port, thermocouples, etc.). The top head supports the vessel material surveillance system (MSS), which provides a remote means to insert and remove material test specimens into and from the reactor to support testing. A hold-down structure sub-assembly is welded underneath the vessel top head. This structure contacts with the top surface of the graphite reflector and provides structural support against upward loads during normal operation and most postulated events. A secondary hold-down structure is installed through the upper graphite layers, extending from the reflector top into submerged graphite layers to transfer upward loads from submerged graphite to the vessel top head during postulated air ingress events. The secondary hold down structure extends to below the minimum reactor vessel coolant level that could result from postulated salt spill events.

4.3.1.1.2 Vessel Shell

The reactor vessel is a 316H SS cylindrical shell that, along with the vessel bottom head, serves to form the safety-related reactor coolant boundary within the reactor vessel. It contains and maintains the inventory of reactor coolant inside the vessel. The shell provides the geometry for coolant inlet and vessel surface for the DHRS, which transfers heat from the reactor vessel during postulated events. The inside of the shell uses 316H SS tabs to maintain the core barrel in a cylindrical geometry and has a welded connection at the top of the core barrel.

4.3.1.1.3 Vessel Bottom Head

The reactor vessel bottom head is a flat 316H SS disc that is welded to the vessel shell. It contains and maintains the inventory of the reactor coolant inside the vessel, supports the vessel internals, maintains the reactor coolant boundary and provides flow geometry for low pressure reactor coolant inlet to the core. Hydrostatic, seismic and gravity loads on the vessel and vessel internals are transferred to the bottom head and are transferred to the RVSS.

4.3.1.2 Reactor Vessel Internals

The reactor vessel internal structures include the graphite reflector blocks, core barrel and reflector support structure. The vessel internal structures define the flow paths of the fuel and reactor coolant, provide a heat sink, a pathway for instrumentation insertion, control and shutdown element insertion, as well as provide neutron shielding and moderation surrounding the core. The reactor vessel internal structures are designed and fabricated using the technical guidance in ASME BPVC Section III, Division 5 (Reference 1) as shown in Table 3.6-2. The design of the structures support inspection and maintenance activities as well as monitoring of the reactor vessel system.

4.3.1.2.1 Reflector Blocks

The reflector blocks are constructed of grade ET-10 graphite. The reflector blocks provide a heat sink for the core and are restrained ensuring alignment of the penetrations to insert and withdraw control elements. The reflector blocks are buoyant in the reactor coolant. The top surface of the reflector blocks contacts the vessel top head hold-down structure sub-assembly, which provides structural support against upward loads during normal operation and most postulated events. A secondary hold-down structure is installed through the upper reflector layers to transfer upward loads from submerged graphite to the vessel top head during postulated air ingress events. The bottom reflector blocks are machined with coolant inlet channels for distribution of coolant inlet flow into the core. The top reflector blocks are machined with coolant outlet channels to direct the coolant exiting from the core into the upper plenum, which includes the hot well, and the PSP pump well, from which the PSP draws suction. The top reflector blocks also form a pebble defueling chute, as shown in Figure 4.3-1, to direct the pebbles from the core to the pebble extraction machine (PEM), allowing online defueling of the reactor (see Section 9.3). The reflector blocks also provide machined channels for insertion and withdrawal of the reactivity control and shutdown elements described in Section 4.2.2.

The reflector blocks form a hot well and pathways to each of four fluidic diodes. The fluidic diodes are stainless-steel passive devices that connect the hot well via the pathway to the top of the downcomer via a penetration in the core barrel as shown in Figure 4.3-1. The diode introduces a higher flow resistance in one direction, while having a lower flow resistance in the other direction. The diode restricts flow from the higher-pressure downcomer into the hot well during normal plant operating conditions with forced (pumped) circulation. During natural circulation, the flow passes in the low-resistance direction of the diode from the hot well to the top of the downcomer. Nozzles on the reactor vessel head and diode inspection channels in the upper reflector block structure are used to perform remote visual inspections of the fluidic diodes.

The graphite reflector blocks reflect neutrons back into the core, increasing the fuel utilization while protecting the reactor vessel from fluence based forms of degradation. Further discussion of the reflector's neutronic characteristics are detailed in Section 4.5.

4.3.1.2.2 Core Barrel

The 316H SS core barrel creates an annular space between itself and the reactor vessel and defines the downcomer flow path for the coolant. The core barrel includes cutout features which limit the siphoning of reactor coolant in the event of a break in the vessel cold leg, and has a flanged top which is welded to the inner wall of the vessel shell. The barrel is kept concentric to the shell by radial tabs, which allow for differential thermal expansion.

4.3.1.2.3 Reflector Support Structure

The 316H SS reflector support structure, as shown in Figure 4.3-1, defines the flow path from the downcomer annulus into the core as well as provides support to the graphite reflector blocks. The reflector support structure ensures a stable core configuration for all conditions (e.g., reactor trip and core motion) by controlling the radial and circumferential positions of the reflector blocks.

4.3.2 Design Basis

Consistent with PDC 1, the safety-related portions of the reactor vessel and reactor vessel internals are fabricated and tested in accordance with generally recognized codes and standards.

Consistent with PDC 2, the reactor vessel and reactor vessel internals perform their safety functions in the event of a design basis earthquake and other natural phenomena hazards.

Consistent with PDC 4, the reactor vessel and reactor vessel internals accommodate the environmental conditions associated with normal operation, maintenance, testing, and postulated events.

Consistent with PDC 10, the reactor vessel and internals maintain a geometry and coolant flow path to ensure that the specified acceptable system radionuclide release design limits (SARRDLs) will not be exceeded during normal operation including postulated events.

Consistent with PDC 14, the reactor vessel is fabricated and tested to have an extremely low probability of abnormal leakage or sudden failure of the reactor coolant boundary by gross rupture.

Consistent with PDC 30, reactor vessel is fabricated, and tested to quality standards, and pre- and in-service inspections, as well as testing where practicable, will be used to detect and identify the location of coolant leakage.

Consistent with PDC 31, the reactor vessel has sufficient margin to withstand stresses under operating, maintenance, testing, and postulated events such that the reactor coolant boundary does not degrade due to the effects of neutron embrittlement, corrosion, material wear, fatigue, stress rupture, thermal loads, or failure due to stress rupture and fracture. The design shall account for residual, steady-state, and transient stresses and consider flaw size.

Consistent with PDC 32, the reactor vessel permits inspection, monitoring, or functional testing of important areas and features to assess structural integrity and leak-tightness of the safety-related portions of the reactor coolant boundary.

Consistent with PDC 33, the core barrel design includes anti-siphon features to limit reactor coolant inventory loss in the event of breaks in the PHTS cold leg.

Consistent with PDC 34, the flow path established by the reactor vessel internals is designed to support the removal of decay heat during normal operation and postulated events, such that SARRDLs and the design conditions of the safety-related elements of the reactor coolant boundary are not exceeded.

Consistent with PDC 35, the reactor vessel internals are designed to maintain structural integrity to assure sufficient core cooling during postulated events and to support removal of decay heat. The safety function of the fluidic diode, reflector blocks, and downcomer is to maintain a flow path that supports natural circulation and to transfer heat from the reactor core during and following postulated events to prevent fuel and reactor internal structure damage that could interfere with continued effective core cooling.

Consistent with PDC 36 and PDC 37 the fluidic diodes are designed to permit periodic monitoring and inspection to provide assurance that the integrity of the natural circulation flow path for decay heat removal is maintained. The design of the decay heat removal natural circulation flow path provided by the downcomer, graphite reflector, hot well, diode pathway and fluidic diode, is also capable of being periodically confirmed to provide assurance that the integrity of the natural circulation flow path for decay heat removal is maintained.

Consistent with PDC 74, the design of the reactor vessel and reflector blocks shall be such that their integrity and geometry are maintained during postulated events to permit sufficient insertion of the control and shutdown elements providing for reactor shutdown.

4.3.3 System Evaluation

The 316H SS structures of the reactor vessel system are fabricated and tested to meet the intent of Reference 1 standards as shown in Table 3.6-2. The 316H SS vessel internals also satisfy the chemistry restrictions of the ASME Section III code in Division 5, Article HGB-2000. Per the ASME standard, ER16-8-

2 weld metal will be used in fabrication of the 316H structures. Commensurate with the safety-related function of the reflector block in ensuring acceptable design limits and maintaining the reactor coolant flow path, quality related controls will be placed on the ET-10 graphite. The graphite reflector will be designed to meet the intent of Reference 1 standards shown in Table 3.6-2. KP-FHR specifications and procurement documents incorporate and reference the applicable guidance and ASME standards. The quality assurance program is described in Section 12.9. These controls demonstrate conformance with PDC 1.

The reactor vessel system makes up a portion of the reactor coolant boundary. The reactor vessel and graphite reflector blocks are therefore designed to maintain geometry during a design basis earthquake to ensure the vessel integrity, insertion of negative reactivity via the RCSS, and to maintain the flow path. The reactor vessel and vessel internals will have dynamic behaviors during a design basis earthquake. These include fluid-structure interaction within the vessel, oscillatory response of components mounted to the reactor top head, i.e., head-mounted oscillators, and relative movement of graphite reflector blocks with respect to one another within the coolant. These dynamic behaviors are accounted for in the design of the reactor and its internals, to ensure continued functionality during and after a design basis earthquake. Models are used to understand fluid migration tendencies considering the pebble bed, reflector blocks, core barrel, and other reactor vessel internal features. The insights gained from the analysis of these models are used to design the reactor to prevent damage to the vessel during a design basis earthquake. The reactor vessel, vessel internals, and vessel attachments such as the RCSS are classified as SDC-3 per ASCE 43-19 “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities” (Reference 2). The reactor vessel will also be protected from the failure of nearby non-safety related SSCs during a design basis earthquake by seismically mounting, physically separating, or using a barrier to preclude adverse interaction, and from failure of attached non-safety related SSCs, such as attached piping (e.g., by design for preferential failure of the non-safety component is a way that does not impact the vessel). These features demonstrate compliance with PDC 2.

The reactor vessel can accommodate internal and external static and dynamic loads. The thermal expansion of the reactor vessel shell and bottom head is supported by the reactor vessel support system (RVSS) (see Section 4.7) during reactor startup, normal operation, and postulated events. Mechanical loadings from static weight, seismic load, and forces from the pebble bed, coolant, and core components are transferred to the vessel shell, to the bottom head, and then to the RVSS. The lateral load path of the vessel support is designed to preclude damage to the decay heat removal system and ensure the vessel maintains its integrity and remains in an upright position. The design of the vessel shell resists hoop stresses from the pressure in the downcomer and supports the transfer of static and dynamic loads between the vessel top head and the vessel bottom head to the RVSS. There are also no pressurized piping systems in or around the reactor vessel, thus precluding pipe whip hazards. Heavy load considerations are addressed in Section 9.8.4, Cranes and Rigging. These features demonstrate compliance with PDC 4.

Core cooling is maintained through the design of the reactor vessel and the reactor vessel internals. As described in Section 4.3.1.2, the vessel and vessel internals define the coolant flow path. To preclude degradation to the vessel due to corrosion of the stainless steel, the reflector blocks and the vessel are “baked” (i.e., heated uniformly) to remove residual moisture prior to coming into contact with coolant. The reflectors, which act as a heat sink in the core, are spaced to prevent the formation of tensile and bending stresses and accommodate thermal expansion and hydraulic forces during normal operation and postulated events. The gaps between the graphite blocks support coolant flow to the reflector thus maintaining a coolable core geometry and precluding reflector degradation by overheating. Maintaining a coolable core geometry and adequate coolant flow through the core ensures the vessel wall temperature is below design limits, which prevent vessel failure. Dynamic behavior of the reactor, its support, and its internals are analyzed and designed to ensure vessel integrity and core geometry are maintained in a design basis earthquake to a degree sufficient to ensure passive heat removal. The vessel, as part of the reactor coolant boundary, ensures the containment of radionuclides by ensuring the coolant is confined and the TRISO particles in the fuel pebbles are protected from damage. These features demonstrate conformance to PDC 10.

To demonstrate compliance with PDC 14, the reactor vessel is fabricated, erected, and tested so as to have an extremely low probability of leakage, rapidly propagating failure, and gross rupture. The reactor vessel materials and weld metal will be qualified for use as described in Kairos Power topical report "Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor," KP-TR-013-P-A (Reference 3). The 316H SS of the reactor vessel as fabricated and tested in accordance with Reference 1 standards has a high fracture toughness at reactor operating conditions, thus reducing the likelihood of crack propagation. The design of the reactor vessel and vessel internals support an 11-year lifetime. This is accomplished by operating the reactor vessel within the as-designed operational and transient condition stresses and by monitoring for changes (e.g., irradiation and thermally induced degradation, corrosion, creep) to the reactor vessel during in-service inspection and testing. . The RVSS-reactor vessel bottom head interface is designed to allow access for weld inspections. The reactor vessel top head supports in-service inspection of attachments and penetrations.

The reactor vessel shell and bottom head maintain a coolant pathway for cooling the reactor core and ensure submergence of fuel pebbles in the core. The reactor vessel is fabricated, erected, and tested in accordance with Reference 1 as a Class A component to account for thermal and physical stresses during normal operation and postulated events. The vessel is fabricated from 316H SS base metal and ER16-8-2 weld metal using a gas tungsten arc welding process. Reference 1 provides for weldment stress rupture factors up to a temperature of 650°C for ER16-8-2 weld metal with 316H base metal. Testing provides stress rupture factors up to 750°C for weld material with 316H base metal (Reference 3). The plant control system will detect leakage from the reactor vessel with catch basins, as described in Section 4.7, that are used to detect leaks in nearby coolant-carrying systems. These features demonstrate compliance with PDC 30.

Reactor vessel stress rupture factors are determined up to 750°C to encompass transient conditions. The stress rupture factors are determined by a creep-rupture test on the vessel base material with weld metal under the gas tungsten arc welding process. The vessel precludes material creep, fatigue, thermal, mechanical, and hydraulic stresses. The leak tight design of the reactor vessel head minimizes air ingress into the cover gas and precludes corrosion of the internals. The high temperature, high carbon grade 316H SS of the core barrel and reflector support structure have high creep strength and are resistant to radiation damage, corrosion mechanisms, thermal aging, yielding, and excessive neutron absorption. Load combinations for the reactor vessel system and the RVSS are provided in Table 4.3-2 and Table 4.7-1. Vessel fluence calculations, as described in Section 4.5, confirm adequate margin relative to the effects of irradiation. The fast neutron fluence received by the reactor vessel from the reactor core and pebble insertion and extraction lines is attenuated by the core barrel, the reflector, and the reactor coolant. Coolant purity design limits are also established in consideration of the effects of chemical attack and fouling of the reactor vessel. These features demonstrate conformance with PDC 31.

The MSS utilizes coupons and component monitoring to confirm that irradiation-affected corrosion is non-existent or manageable. The 316H SS reactor vessel and ER16-8-2 weld material, as a part of the reactor coolant boundary, will be inspected for structural integrity and leak-tightness. As detailed in Reference 3, fracture toughness is sufficiently high in 316H SS under reactor operating conditions that additional tensile or fracture toughness monitoring and testing programs are unnecessary. These features demonstrate conformance to PDC 32.

Anti-siphon cutouts are above the PHTS cold leg with coolant on both sides of the core barrel during normal operation. In the event of a cold leg break, reactor coolant level is expected to decrease and the cover gas moves into the downcomer to break the siphon, thus precluding coolant from being siphoned below the fluidic diode flow pathway elevation. These design features demonstrate conformance to PDC 33.

The reactor vessel internals support decay heat removal during normal operations by establishing the physical geometry for the coolant flow path. During normal operations, the reactor vessel internal structures act in conjunction with forced flow in the PHTS to ensure the transfer and rejection of heat from the core via the coolant flow path. When passive decay heat removal is required in response to postulated events, the physical geometry and structure of the reactor vessel internals provides a pathway for continuous natural circulation of coolant via flow through the fluidic diodes. These features demonstrate conformance to PDC 34.

The downcomer, graphite reflector, hot well, fluidic diode pathway and fluidic diodes are used to establish a flow path for continuous natural circulation of coolant in the core during postulated events to remove decay heat from the reactor core to the vessel wall. During and following a postulated event, the hot coolant from the core flows from the hot well through the diode pathway, the low flow resistance direction of the fluidic diode to the cooler downcomer via natural circulation. The core is thereby cooled passively. Continuous coolant flow through the reactor core prevents potential damage to the vessel internals due to overheating thereby ensuring the coolable geometry of the core is maintained. These features demonstrate compliance with PDC 35. Additional functions performed by the DHRS to support passive decay heat removal are described in Section 6.3.

The downcomer, graphite reflector blocks, and fluidic diodes are passive components designed to maintain structural integrity during postulated events to maintain a natural circulation path and a coolable core geometry for removal of decay heat. The reactor vessel internals are qualified in accordance with Reference 3 and Reference 4 and are designed to perform their function during seismic events as noted above. Based on the design and qualification, there are no credible failure mechanisms within the design basis of the core barrel and the graphite structures that result in a loss of structural integrity. Therefore, degradation of the natural circulation flow path required to support decay heat removal is not expected during normal or postulated events and such failures would be beyond the design basis. However, graphite dust is expected to be present in small quantities in the system and could be postulated to accumulate in portions of the reactor coolant pathway. The functional capability of the normal flow path can be periodically confirmed during operation by monitoring temperature changes to the exit from the reactor vessel. Similarly, the portions of the reactor coolant flow path that are unique to natural circulation (diode pathway and fluidic diode) are capable of being confirmed during normal operations via temperature changes across the diode and across the pathway. Additionally, the fluidic diodes are designed to permit periodic remote inspections via penetrations on the vessel top head to ensure the pathway remains unobstructed. Instrumentation for temperature measurement across the fluidic diode is permitted via the same penetrations used for visual inspection. These features and capabilities demonstrate conformance to PDC 36 and PDC 37. Additional functions performed by the DHRS to support passive decay heat removal are described in Section 6.3.

The reactor vessel reflector blocks permit insertion of the reactivity control and shutdown elements. The ET-10 grade graphite of the reflector blocks is compatible with the reactor coolant chemistry and will not degrade due to mechanical wear, thermal stresses and irradiation impacts during the reflector block lifetime. The graphite reflector material is qualified as described in the Kairos Power topical report

"Graphite Material Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor," KP-TR-014-P-A (Reference 4). To preclude damage to the reflector due to entrained moisture in the graphite, the reflector blocks are "baked" (i.e., heated uniformly) prior to coming into contact with coolant. The reflectors, which act as a heat sink in the core, are spaced to accommodate thermal expansion and hydraulic forces during normal operation and postulated events. The gaps between the graphite blocks also allow for coolant to provide cooling to the reflector blocks. The reactor vessel permits the insertion of the reactivity control and shutdown elements as well. The vessel is classified as SDC-3 per ASCE 43-19 and will maintain its geometry to ensure the RCSS elements can be inserted during postulated events including a design basis earthquake. These features demonstrate compliance with PDC 74.

4.3.4 Testing and Inspection

The reactor vessel and internals will be included in an in-service inspection program, which will be submitted at the time of the Operating License Application.

4.3.5 References

1. American Society of Mechanical Engineers, ASME Boiler & Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors." 2017.
2. ASCE 43-19, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities."
3. Kairos Power, LLC, "Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor," KP-TR-013-P-A, April 2023.
4. Kairos Power, LLC, "Graphite Material Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor," KP-TR-014-P-A, April 2023.

Table 4.3-1: Reactor Vessel Top Head Penetrations

Name of Penetration	Number of Penetrations	System
Pebble Extraction Machine (PEM)	1	PHSS
Pebble Insertion	2	PHSS
Reactivity Shutdown Element	3	RCSS
Reactivity Control Element	4	RCSS
Primary Salt Pump (PSP)	1	PHTS
Coolant Fill/Drain Line	2	IMS
Inert Gas Line	2	IGS
Material Surveillance System	1	MSS
Neutron Source	1	RSS
Reserve Instrumentation	3	I&C
Reactor Coolant Level Sensor	4	I&C
Reactor Coolant Thermocouple	3	I&C
Graphite Thermocouple	2	I&C
Fluidic Diode Inspection Nozzle	4	I&C

Table 4.3-2: Load Combinations for the Reactor Vessel System

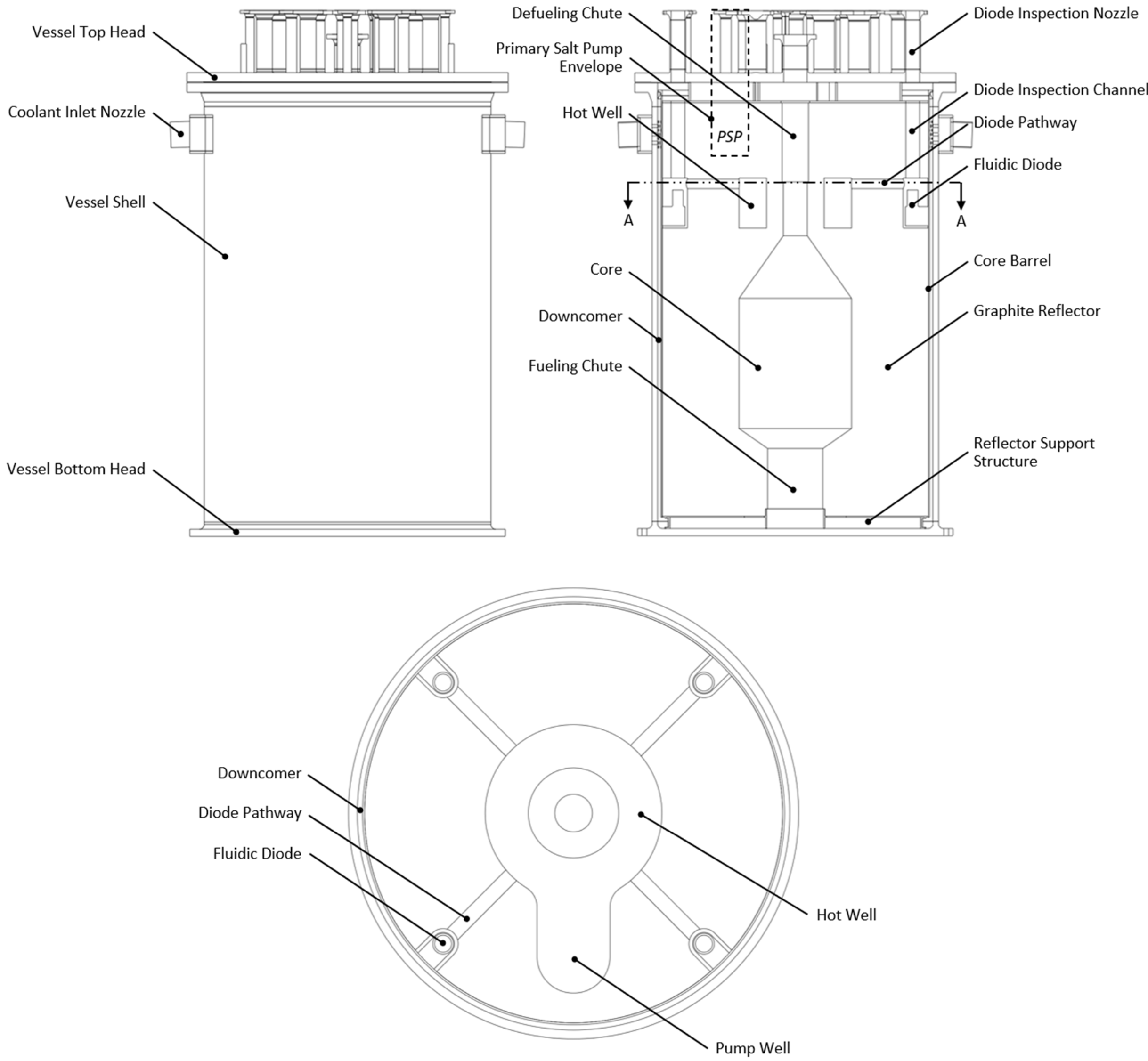
Service Level	Load Combination*
A	$D + L + T_o + P_o + R_o$
B	$D + L + T_o + P_o + R_o + E_o$ $D + L + T_i + P_i + R_i + E_o$
C	$D + L + T_o + P_o + R_o + E_{ss}$ $D + L + T_s + P_s + R_s + E_{ss}$
D	$D + L + T_a + P_a + R_a + W_t$ $D + L + T_a + P_a + R_a + E_{ss}$

*Load combination refers to the types of loads considered acting simultaneously. Application of load factors and specific details of load combination effects are per the applicable design standards.

Load Nomenclature:

D	Dead loads
L	Live loads
T_o	Thermal loads during startup, normal operating, or shutdown conditions
T_i	Thermal loads during Service Level B loadings
T_a	Thermal loads during Service Level D loadings
T_s	Thermal loads during Service Level C loadings
P_o	Pressure loads during startup, normal operating, or shutdown conditions
P_i	Pressure loads during Service Level B loadings
P_s	Pressure loads during Service Level C loadings
P_a	Pressure loads during Service Level D loadings
R_o	Pipe reactions during startup, normal operating, or shutdown conditions
R_i	Pipe reactions during Service Level B loadings
R_a	Pipe reactions during Service Level D loadings
R_s	Pipe reactions during Service Level C loadings
E_o	Loads generated by 1/3 of design basis earthquake (DBE)
E_{ss}	Loads generated by DBE
W_t	Accidental loads due to missile impact effects

Figure 4.3-1: The Reactor Vessel System



Section A-A

Figure 4.3-2: Reactor Vessel Top Head Design

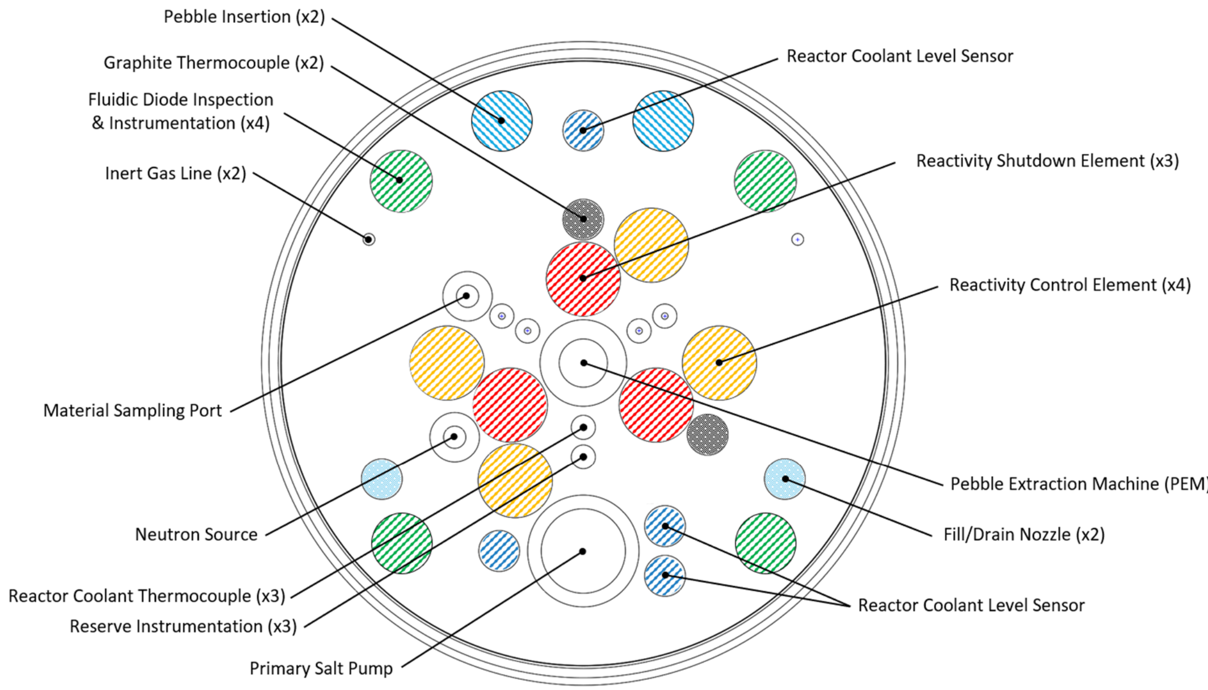
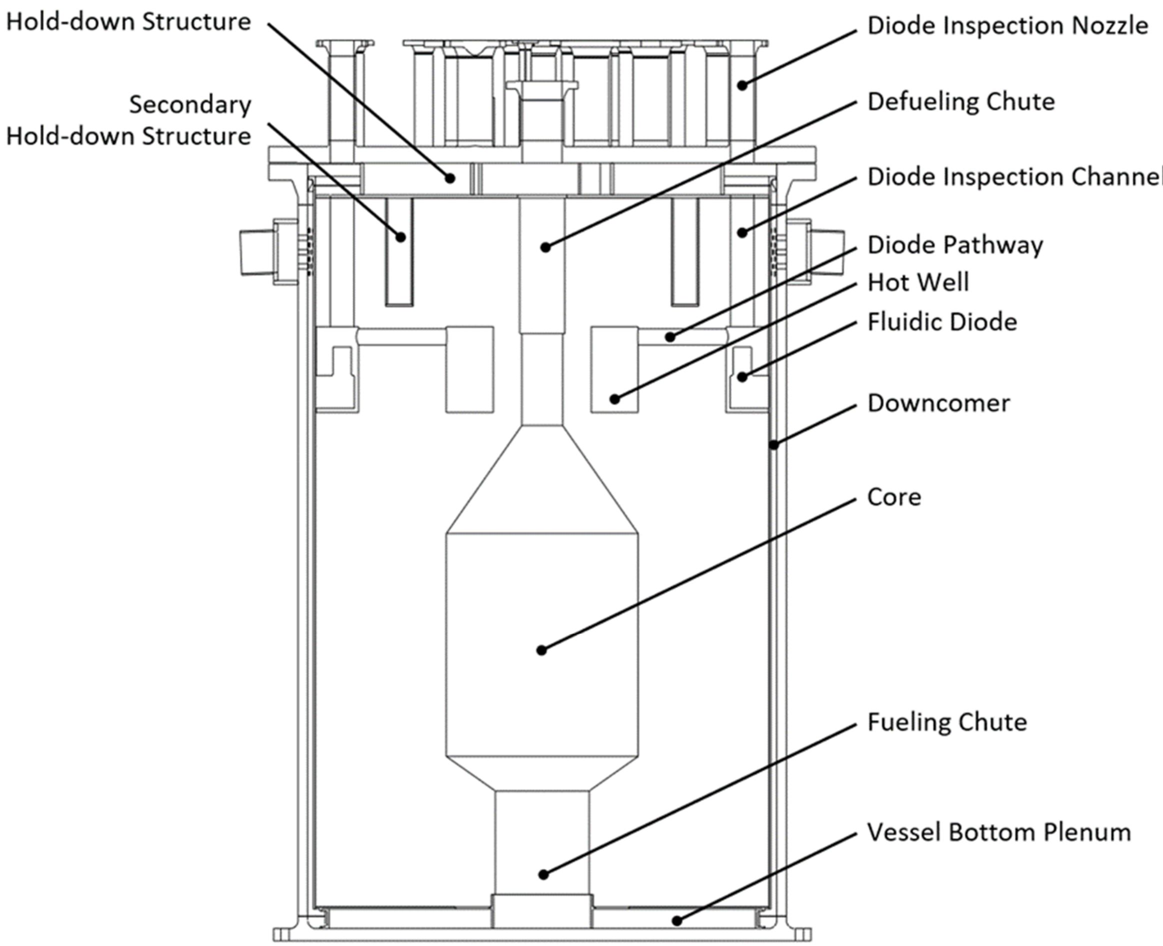


Figure 4.3-3: The Reactor Vessel System Secondary Hold-Down Structure



4.4 BIOLOGICAL SHIELD

4.4.1 Description

The biological shield forms a barrier to protect plant workers and the public from radiological exposure. In addition, the biological shield reduces radiation damage to plant equipment and also reduces the potential for Beryllium exposure to reactor personnel. The shielding provided by the biological shield is sufficient to meet the radiation exposure goals described in Chapter 11. The biological shield accomplishes this shielding primarily using reinforced concrete.

There are two biological shields in the design, a primary biological shield and a secondary biological shield. The primary biological shield is constructed of concrete and is located just outside the reactor vessel. The secondary biological shield is located outside the primary biological shield and contains the inventory management and the primary to intermediate heat transfer systems. A notional representation of the primary and secondary biological shields is shown in Figure 4.4-1. The biological shield is not shared by Unit 1 and Unit 2.

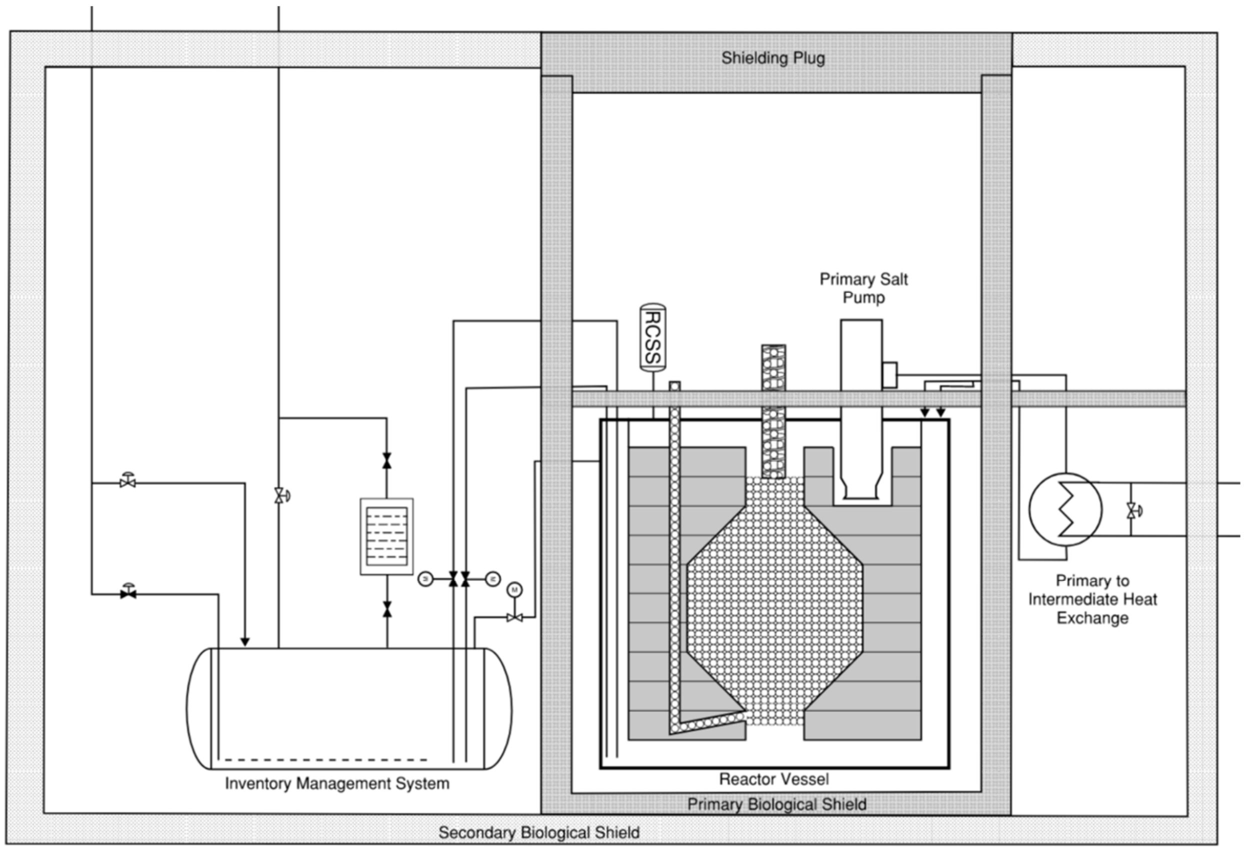
4.4.2 Design Bases

The biological shield is provided for worker protection to meet 10 CFR 20 requirements and is not credited in the prevention or mitigation of postulated events. However, the primary biological shield is a safety-related structure and remains intact during normal operation and postulated events. The structural design bases are described in Chapter 3.

4.4.3 Evaluation

An evaluation of the shielding performance of the biological shield to meet 10 CFR 20 will be provided with the application for an Operating License.

Figure 4.4-1: Primary and Secondary Biological Shield



4.5 NUCLEAR DESIGN

This section describes the nuclear design of the reactor, including the design bases and the analytical methods used to perform the nuclear design. Analytical results for equilibrium core operation are presented.

4.5.1 Nuclear Design Description

4.5.1.1 Overview of Core Nuclear Design

The reactor core is comprised of a packed bed of spherical fuel pebbles (see Section 4.2.1), with each fuel pebble containing approximately 6 grams of uranium. The core is roughly 60% pebbles and 40% reactor coolant by volume. Pebbles are introduced at the bottom of the core and transit slowly to the top of the core in approximately 30 to 50 days, where they exit to the pebble handling and storage system (PHSS). The pebbles make multiple passes through the core during their lifetime, before reaching their design burnup. The total residence time of a fuel pebble in the core at equilibrium is approximately 316 days. Some of the pebbles are made entirely of graphite matrix material as described in Section 4.2.1. These moderator pebbles are used to improve neutron moderation and constitute a fraction of the core at initial operation and during normal operation. The reactor core contains approximately 36,000 pebbles (fueled and moderator). The Flibe reactor coolant (see Section 5.1) also provides neutron moderation to the core.

The core is surrounded by a graphite reflector (see Section 4.3), which increases neutron economy, provides neutron moderation, and shields the reactor structures (core barrel, reactor vessel, and other critical components) from fast neutrons. The reflector also maintains the core geometry during the life of the plant. An overview of the core and surrounding reflector is shown in Figure 4.2-7.

The reactor is continuously refueled. As pebbles exit the core, they are examined for burnup and potential physical damage in the PHSS (see Section 9.3). As pebbles approach their design burnup, they are removed and placed in storage, and fresh pebbles are introduced into the core along with recirculated pebbles, which have not yet reached their design burnup.

Neutron moderation is provided by the graphite in the fueled pebbles, the graphite moderator pebbles, the surrounding graphite reflector, and by the reactor coolant. The core composition is such that the core is slightly under-moderated during all operating conditions. Also, as a result of the continuous addition and removal of fuel from the reactor, the reactor operates with a low excess reactivity.

The reactor may be initially started up with a mixture of pebbles; some with natural uranium, some with fuel enrichments in the range of 10 to 15 wt% U-235, and moderator pebbles. As fission products build up, the pebbles with natural uranium will be replaced with fresh pebbles enriched to just under 20 wt% U-235. Another approach to startup is a critical height approach, which has been demonstrated in the Chinese pebble bed reactor, HTR-10. Initial startup and power ascension will be discussed in the application for an Operating License.

There are four main periods of core operation in the life of the reactor: startup, power ascension, transition to equilibrium, and equilibrium operation. The first period is reactor startup, which is defined as the approach to criticality. Power ascension is the process of increasing to full power and can be characterized in two distinct phases: Low Power (0-10% of full power) and Ascension to Full Power (10% - 100% of full power). During the Transition to Equilibrium, the natural uranium pebbles are gradually replaced with fresh pebbles containing uranium until all fueled pebbles contain particles of just under 20 wt% U-235. Equilibrium Operation is achieved when the radionuclide inventory in the core is no longer changing, the ratio of insertion of fuel and moderator pebbles is stable, the enrichment of the fresh pebbles being inserted into the core is not changing, and control elements are not repositioning (or are very minimally engaged). These periods of core operation are depicted in Figure 4.5-1.

The neutronic results for the equilibrium core are the limiting results for the reactor and fuel for normal power operation. When operating at 100% power, the equilibrium core will have the highest average enrichment, pebble power, fuel temperatures, average core burnup, and fast neutron flux. Therefore, neutronic results for startup and initial operation are bounded by the results for the equilibrium core.

A comparison of the neutronic parameters for the reactor and a small light water reactor is provided in Table 4.5-1. A summary of reactor neutronic parameters is provided in Table 4.5-2.

4.5.1.2 Reactivity Coefficients

The following reactivity coefficients are important for the reactor: fuel temperature (Doppler), moderator temperature (graphite in the fuel pebbles and graphite in the moderator pebbles), coolant temperature, coolant void, and reflector temperature.

The fuel temperature reactivity coefficient is the change in reactivity due to a change in fuel temperature. The moderator temperature reactivity coefficient is the change in reactivity due to the change in fuel pebble graphite and graphite pebble temperature. The coolant temperature reactivity coefficient is the change in reactivity due a change in reactor coolant temperature (including the appropriate density change). The coolant void reactivity coefficient is the change in reactivity due to coolant void fraction. The reflector temperature reactivity coefficient is the change in reactivity due to reflector temperature change.

4.5.1.3 Power Distribution

The parameters are used to characterize the core power distribution in the reactor are:

Axial Peaking Factor (F_Z)

This is the ratio between the average power at a given elevation divided by the average power over all elevations.

Radial Peaking Factor (F_R)

This is the ratio of the average power at a radial location divided by the average power over all radial locations.

Total Peaking Factor (F_Q)

This is the ratio of the maximum power anywhere in the core to the average power for the entire core.

4.5.1.4 Shutdown Margin

Shutdown margin is the instantaneous amount of reactivity by which the reactor is subcritical, or would be subcritical from a given condition, assuming that all shutdown elements are inserted with the exception of the highest worth shutdown element, which is assumed to be fully withdrawn.

The shutdown margin calculation accounts for the following factors:

- Power Defect
- Xenon Decay
- Operating Excess Reactivity
- Margin for Uncertainties

The methodology for determining shutdown margin is described in the “KP-FHR Core Design and Analysis Methodology” technical report (Reference 1).

Hot shutdown is defined as the state where reactor is subcritical at a temperature of 550°C. The shutdown margin is defined for the most limiting core at the reactor coolant freezing temperature. The shutdown margin design criterion is that k-effective must be less than 0.99.

4.5.1.5 Nuclear Transient Parameters

The key kinetic parameters that are used in transient analysis are:

- Prompt neutron lifetime
- Delayed neutron fraction groups and their decay constants

In addition, core power distribution and reactivity coefficients are also provided as initial condition inputs to the transient analysis. The methodology for calculating these coefficients is provided in Reference 1.

4.5.1.6 Analytical Methods

The core design methods are comprised of the Serpent 2, STAR-CCM+, KPACS, and KPATH computer codes. The Serpent 2 code is a multi-purpose, three dimensional continuous-energy Monte Carlo particle (neutrons and gammas) transport code. STAR-CCM+ is a computational fluid dynamics simulation software that uses discrete element modeling and porous media approximation capabilities for thermal-hydraulic characterization of pebble bed flow and temperature. KPACS is a fuel cycle analysis code. KPATH is used for coupling Serpent 2 and STAR-CCM+.

The method for validation and verification of these codes including the method for determining uncertainty factors is described in Reference 1.

4.5.2 Design Bases

The design bases related to nuclear design are as follows:

Consistent with PDC 10, the reactor core has appropriate margin to assure that the specified acceptable system radionuclide release design limits (SARRDLs) are not exceeded. SARRDLs are described in Section 6.2.

Consistent with PDC 11, the reactor core is designed so that in the power operating range the net effect of prompt inherent nuclear feedback tends to compensate for rapid increase in reactivity.

Consistent with PDC 12, the reactor core assures that power oscillations which can result in conditions exceeding SARRDLs are not possible or can be reliably and readily detected and suppressed.

Consistent with PDC 26, the nuclear design analysis is performed to confirm that the reactor control and shutdown system (RCSS) provide a means for (1) inserting negative reactivity such that SARRDLs, are not exceeded and safe shutdown can be achieved during normal operation; (2) reliably controlling reactivity changes during normal operation; (3) inserting negative reactivity of a sufficient amount to cool the core and maintain safe shutdown following an accident; and (4) holding the reactor shutdown during fuel loading, inspection, and repair.

4.5.3 Nuclear Design Evaluation

This section provides an evaluation of the nuclear design and describes how the nuclear design bases in Section 4.5.2 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 Evaluation of Design Bases

Reactivity Coefficients

The range of values for reactivity parameters is shown in Table 4.5-3 for startup and equilibrium operation. These reactivity coefficients are calculated in accordance with the methodology described in Reference 1.

In compliance with PDC 11, the net effect of reactivity is such that the overall reactivity coefficient is negative, which compensates for rapid increase in reactivity. As shown in the Table 4.5-3 the prompt components (doppler, moderator, coolant, and void) are all negative and only the reflector temperature coefficient is positive. The reflector reactivity coefficient is the result of spectrum hardening at the periphery of the core due to increased reflector temperature. This change in spectrum reduces the fission rate next to the reflector and shifts flux more towards the inner part of the core, effectively reducing leakage. This effect combined with local over moderated conditions ultimately leads to a positive feedback coefficient. The mechanisms determining the reflector feedback are different from those that determine the moderator temperature feedback. In the moderator temperature coefficient case, because the core is designed to be under moderated, the spectrum hardening leads to a reduced resonance escape probability greater than the increase in thermal utilization. It should be noted that the methodology applied to determine the reflector reactivity feedback does not assume any thermal expansion of the reflector which would be a negative feedback. Furthermore, the reflector effect is considerably delayed compared to fuel temperature and coolant temperature feedbacks.

Power Distribution

Power distributions are summarized in Table 4.5-4. These results are provided for equilibrium operation. These power distribution results are calculated in accordance with the methodology described in Reference 1.

Neutron flux distributions are verified during startup using excore detectors. These measurements are compared against core design calculations to ensure that the core is operating as designed.

Thermal hydraulic analysis is summarized in Section 4.6 and is performed on a limiting power distribution.

In compliance with PDC 10, the power distribution results in combination with the thermal hydraulic analysis in Section 4.6 ensure that SARRDLs are not exceeded during normal operation and postulated events.

The control element and shutdown element pattern is shown in Figure 4.2-7. The pattern is not one-quarter core symmetric, however, this is of no consequence due to the small core size and long neutron diffusion length.

Shutdown Margin

Shutdown margin values for equilibrium operation is shown in Table 4.5-5. These values are best estimate values determined using the methodology in Reference 1. Uncertainties will be applied to these values in accordance with the uncertainty values in Table 6-1 of Reference 1. This methodology includes the assumption of a single most reactive control or shutdown element being fully withdrawn from the core.

The nuclear design provides confirmation that the RCSS provides two means of controlling reactivity. As described in Section 4.2.2, there are four reactivity control elements that insert in the neutron reflector and three reactivity shutdown elements that insert into the pebble bed core. In compliance with PDC 26

Condition 1, the shutdown elements are solely credited to provide a means to ensure that SARRDLs are not exceeded, and that safe shutdown is achieved and maintained during normal operation and postulated events. Condition 1 is met assuming the highest worth shutdown element is fully withdrawn. In compliance with Condition 2 of PDC 26, the control elements by themselves provide the capability to control reactivity changes during planned normal power changes such that the SARRDLs are not exceeded. The control elements provide a means of reactivity control that is independent and separate from the shutdown elements. The control elements are diverse from the shutdown elements because they have a different geometry, insert into different locations, and have different mechanisms (i.e., the control elements use a motor-driven winch and shutdown elements are gravity driven). In compliance with Condition 3 of PDC 26, the shutdown elements provide a means of inserting reactivity at a sufficient rate and amount, to ensure that the capability to cool the core is maintained and a means for shutting down the reactor and maintaining it at safe shutdown following a postulated event. Condition 3 is met assuming that the most reactive shutdown element is fully withdrawn. In compliance with Condition 4 of PDC 26, the shutdown elements provide a means for maintaining the reactor shutdown to allow for interventions such as fuel loading, inspection, and repair. Although the shutdown elements are also solely credited for meeting conditions 1,3, and 4 of PDC 26, the control elements are also automatically inserted in response to a reactor trip signal and provide an additional line of defense against exceeding reactivity margins. Compliance with PDC 26 is summarized in Table 4.5-6.

Nuclear Stability

The inherent nuclear characteristics of the reactor are such that uncontrolled power oscillations are not possible. The reactor is small in size and is neutronically connected due to the long diffusion length of neutrons in the core. As a result, the reactor is inherently stable with regard to both axial and radial power oscillations. In compliance with PDC 12, the reactor is not susceptible to nuclear instability.

4.5.3.2 Nuclear Design Analysis Inputs to Other Sections

Vessel Irradiation

The fast neutron fluence received by the reactor vessel from the reactor core and pebble insertion and extraction lines is attenuated by the core barrel, the reflector, and by the reactor coolant. Fluence and depletion calculations are performed to confirm that the vessel is not adversely affected by this neutron fluence. The methodology for calculating best estimate vessel fluence and associated transmutation products is described in Reference 1. These calculated values are evaluated using conservative uncertainties.

The calculation of associated dpa on the vessel uses the fluence as input. The preliminary best estimate dpa plus uncertainty is within 30% of the low-level irradiation value discussed in Reference 2.

Nuclear Transient Analysis

Values for neutron generation time and delayed neutron fraction are shown during startup and equilibrium operation in Table 4.5-7. In addition, conservative values for power distribution, reactivity coefficients, and shutdown margin are provided for the initial conditions for each of the postulated reactivity transient events analyzed in Chapter 13.

The most credible inadvertent insertion of excess reactivity does not disturb the core and does not adversely impact the capability to cool the core in accordance with PDC 28 as described in Chapter 13.

4.5.4 Core Design Limits

4.5.4.1 Nuclear Core Design Limits

The reactor core design is performed such that the design parameters during normal operation are within the fuel qualification envelope described in Section 4.2.1 for peak particle power, burnup, peak fluence, and peak fuel temperature.

4.5.4.2 Testing and Monitoring

Neutron flux and burnup are monitored during operation to ensure that the core is performing within design. Neutron flux excore detectors are further described in Section 7.3.1. Burnup measurement sensors are further described in Section 9.3.1.5.

The following core nuclear design parameters are anticipated to be included in the technical specifications:

- Shutdown Margin
- Coolant Outlet Temperature
- Moderator pebble to fuel pebble ratio

There will also be a technical specification controlling the fuel enrichment to less than 20 wt% U-235. The technical specifications are described in Chapter 14.

4.5.5 References

1. Kairos Power LLC, “KP-FHR Core Design and Analysis Methodology,” KP-TR-017-P, Revision 1. September 2022.
2. Kairos Power LLC, “Metallic Materials Qualification of the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor.” KP-TR-013-P-A, April 2023.

Table 4.5-1: Comparison of KP-FHR Test Reactor with Light Water Reactor

Nuclear Parameter	KP-FHR Reactor	Small Light Water Reactor
Power Level (MWth)	35	200
Reactor Inlet/Outlet Temperature (°C)	550/650	258/310
Power Density (MWth/m^3)	17.5	58.9
Core Volume (m^3)	2	3.4
Number of Reactivity Control Elements	7	16
Shutdown Margin at Equilibrium (pcm)	4997	2696
Discharge Burnup (% FIMA)	6	4.3
Enrichment (% U-235)	< 20	<5

Table 4.5-2: Nuclear Design Parameters for the Reactor Core

Parameter	Value
Power (MWth)	35
Core Volume (m^3)	2.0
Power Density (MW/m^3)	17.5
Moderation (%graphite pebbles)	16
Average Power per TRISO Particle (mW)	73
Total Pebbles in Core	36,000
Average Residence Time for Equilibrium Core (effective full power days)	316
Fuel Consumption (# of pebbles per day) during equilibrium	108
Equilibrium Discharge Burnup (% FIMA)	~6
Max Fuel Kernel Surface T (°C)	876
Core Flow (kg/sec)	210

Table 4.5-3: Reactivity Coefficients

Reactivity Coefficient	Startup	Equilibrium
Fuel Doppler (pcm/°C)	-6.2	-4.1
Moderator (pcm/°C)	-1.5	-0.4
Coolant (pcm/°C)	-2.3	-1.6
Void (pcm/%void), @3% void	-34	-53
Reflector (pcm/°C)	+2.6	+2.0

Table 4.5-4: Calculated Power Distribution Peaking Factors for Equilibrium Operation

Power Distribution	Equilibrium
Axial Peak (F_Z)	1.2
Radial Peak (F_R)	1.2
Total Pebble Peaking (F_Q)	1.8

Table 4.5-5: Shutdown Margin for Equilibrium

Parameter	Value at Equilibrium
Required Shutdown Margin	1,000
Actual Shutdown Margin (pcm)	3,654
Required Worth for Shutdown (pcm) [1]	11,578
Worth of Shutdown Elements (pcm)	14,232

Notes:

1. Required worth considers highest worth shutdown element fully withdrawn (which is 6,266 pcm)

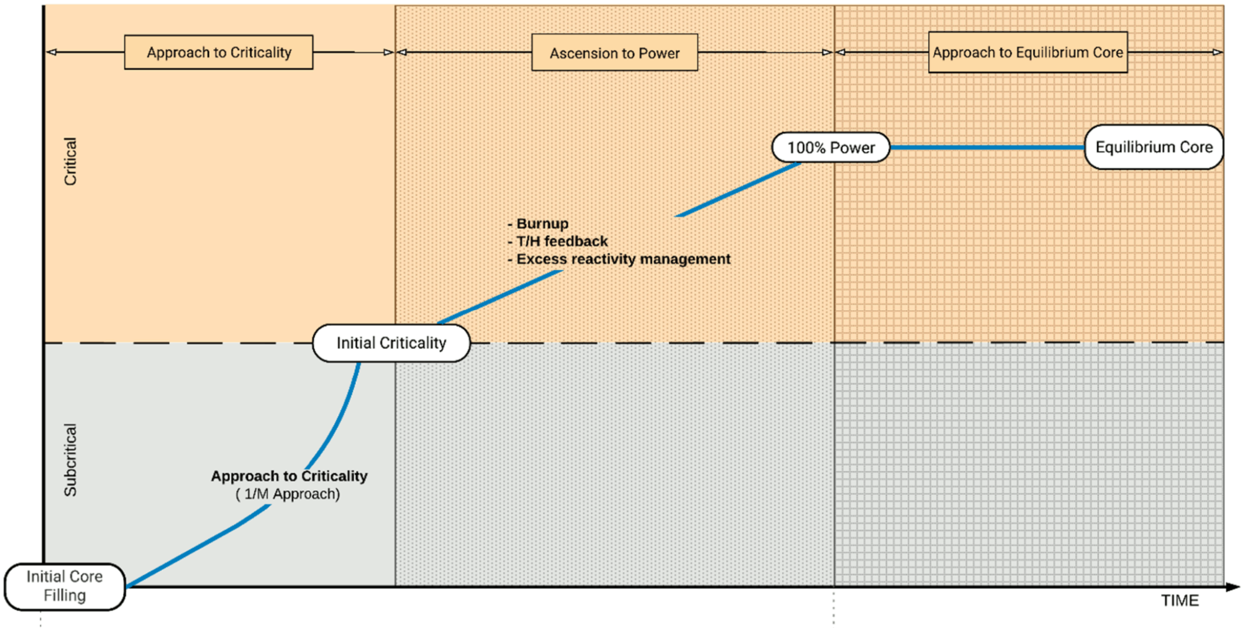
Table 4.5-6: PDC 26 Compliance

PDC 26 Criteria	Credited Means for Compliance
A minimum of two 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 are not exceeded and safe shutdown is achieved and maintained during normal operation, including anticipated operational occurrences.	3 SE With maximum worth element assumed fully withdrawn
(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 specified acceptable system radionuclide release design limits are not exceeded.	4 CE
(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 event.	3 SE With maximum worth element assumed fully withdrawn
(4) A means for holding the reactor shutdown under conditions which allow for interventions such as fuel loading, inspection and repair shall be provided.	3 SE
CE – Control Elements SE – Shutdown Elements	

Table 4.5-7: Values for Kinetics Coefficients

Parameter	Startup	Equilibrium
Neutron Mean Lifetime (seconds)	5.87×10^{-4}	4.60×10^{-4}
Effective Delayed Neutron Fraction (pcm)	668	605
Neutron Mean Generation Time (seconds)	5.84×10^{-4}	4.56×10^{-4}

Figure 4.5-1: Startup and Equilibrium Operation



4.6 THERMAL-HYDRAULIC DESIGN

4.6.1 Description

The thermal hydraulic design of the reactor is a combination of design features that enable effective heat transport from the fuel pebble to the reactor coolant and eventually to the heat rejection system of the reactor, considering the effects of bypass flow and flow non-uniformity. The design features that play a key role in the thermal-hydraulic design of the reactor system include the fuel pebble (see Section 4.2.1), reactor coolant (see Section 5.1), reactor vessel and reactor vessel internal structures (see Section 4.3), the primary heat transport system (PHTS) (see Section 5.1) and the intermediate heat transport system (IHTS) (see Section 5.2).

Thermal hydraulic computer codes and evaluation models are discussed in Section 4 and 5 of Reference 1, and Section 4 of Reference 2.

4.6.1.1 Core Geometry

The core geometry is maintained in part by the reactor vessel internals including the reflector blocks, which keep the pebbles in a general cylindrical core shape. Coolant inlet channels in the graphite reflector blocks are employed to limit the core pressure drop. The use of pebbles in a packed bed configuration also creates local velocity fields that enhance pebble-to-coolant heat transfer. The reactor thermal hydraulic design uses the following heat transfer mechanisms to extract the fission heat.

- Pebble-to-coolant convective heat transfer
- Pebble radiative heat transfer
- Pebble-to-pebble heat transfer by pebble contact conduction
- Pebble-to-pebble heat transfer by conduction through the reactor coolant
- Heat transfer to the graphite reflector by modes of conduction, convection, and radiation.

4.6.1.2 Coolant Flow Path

During normal operation, reactor coolant at approximately 550°C enters the reactor vessel from two PHTS cold leg nozzles and flows through a downcomer formed between the metallic core barrel and the reactor vessel shell as shown in the normal, pumped flow pathway on Figure 4.6-1, part (a). The coolant is distributed along the vessel bottom head through the reflector support structure, up through coolant inlet channels in the reflector blocks and the fueling chute and into the core with a portion of the coolant bypassing the core via gaps between the reflector blocks, the fluidic diode pathway and the fluidic diode. The coolant transfers heat from fuel pebbles, which are buoyant in the coolant and provides cooling to the reflector blocks and the control elements via engineered bypass flow. Coolant travels out of the active core through the upper plenum via the coolant outlet channels and exits the reactor vessel via the PHTS outlet. The nominal core outlet temperature is dependent on the amount of corresponding bypass flow.

During postulated events where the normal heat removal path through the PHTS is no longer available, including when the PHTS is drained, a fluidic diode (see Section 4.3), is used to create an alternate, natural circulation flow path. During such events, forced flow from the primary salt pump (PSP) is also not available. The fluidic diode then directs flow from the hot well and diode pathway through the core barrel and into the downcomer as shown in the natural circulation flow pathway on Figure 4.6-1, part (b). This opens the path for continuous flow via natural circulation. During normal operation, while the PSP is in operation, the fluidic diode minimizes reverse flow. Qualification or functional testing plans for the fluidic diode as well as any test results needed to validate performance assumed in the safety analysis will be available with the application for an operating license.

4.6.2 Design Basis

Consistent with PDC 10, the thermal-hydraulic design provides adequate transfer of heat from the fuel to the coolant to ensure that the specified acceptable system radionuclide release design limits (SARRDLs) will not be exceeded during normal operation and unplanned transients.

Consistent with PDC 12, the thermal hydraulic design of the reactor system ensures that power oscillations that can result in conditions exceeding SARRDLs are not possible or can be reliably and readily detected and suppressed.

Consistent with PDC 34, the thermal hydraulic design removes residual heat during normal operation and anticipated transients, such that SARRDLs and the design conditions of the safety-related elements of the reactor coolant boundary are not exceeded.

Consistent with PDC 35, the reactor transfers heat from the reactor core during anticipated transients such that fuel and reactor internal structure damage that could interfere with continued effective core cooling is prevented.

4.6.3 System Evaluation

The reactor core and heat removal systems associated with the thermal hydraulic design of the reactor system have appropriate margin to ensure that SARRDLs are not exceeded during any condition. The height of the core (e.g., height of the downcomer) and the axial decay heat profile (e.g., the temperature difference between the hot leg and the cold leg) ensure there is sufficient driving force to enable natural circulation in the event of a loss of forced circulation. Pressure losses are also minimized by design to ensure that heat is transferred from the coolant in the downcomer below the fluidic diode to the vessel shell during a loss of forced circulation event. Due to buoyancy forces, hot fluid coming out from the fluidic diode path into the downcomer will flow downward as a plume, which enhances heat removal from the vessel shell above the elevation of the fluidic diode. A summary of pertinent thermal-hydraulic parameters is provided in Table 4.6-1. These features and analyses demonstrate conformance to PDC 10 with respect to thermal hydraulic design.

The thermal hydraulic design of the reactor system inherently prohibits instability phenomena that could exceed SARRDLs. The reactor is kept at atmospheric pressure; the coolant in the core does not experience two phase flow and has a high thermal inertia making the reactor restrictive to core-wide thermal-hydraulic instability events. This demonstrates compliance with PDC 12 with respect to the thermal hydraulic design. The results of analyses supporting the inherent stability of the reactor will be provided with the application for an Operating License.

The thermal hydraulic design of the reactor system provides residual heat removal during normal operations, including startup and shutdown. During normal operations, the thermal hydraulic design of the reactor in conjunction with forced flow in the PHTS and IHTS ensures the transfer and rejection of heat from the core via the coolant flow path as described in Section 4.6.1.2. The relationship between power and flow of the thermal hydraulic system as well as the thermal inertia of the coolant ensures that heat transfer can be achieved at a rate that maintains the design conditions of the core. These features demonstrate conformance to PDC 34 with respect to thermal hydraulic design.

The thermal hydraulic design of the reactor supports passive residual heat removal following postulated events. The design of the reactor downcomer, reflector blocks, and the fluidic diode provide a path for continuous flow to ensure decay heat is transferred via natural circulation from the core to the reactor vessel shell, as described in Section 4.6.1.2. These features, in part, demonstrate compliance with PDC 35. Residual heat is removed from the vessel wall by the DHRS as described in Section 6.3.

4.6.4 Testing and Inspection

Reactor coolant temperatures, flow, and core power will be periodically monitored during operations to be within specified limits. Instrumentation will also be periodically calibrated.

4.6.5 References

1. Kairos Power LLC, "KP-FHR Core Design and Analysis Methodology," KP-TR-017-P, Revision 1. September 2022.
2. Kairos Power LLC, "Hermes 2 Postulated Event Methodology," KP-TR-022-P, Revision 0. June 2023.

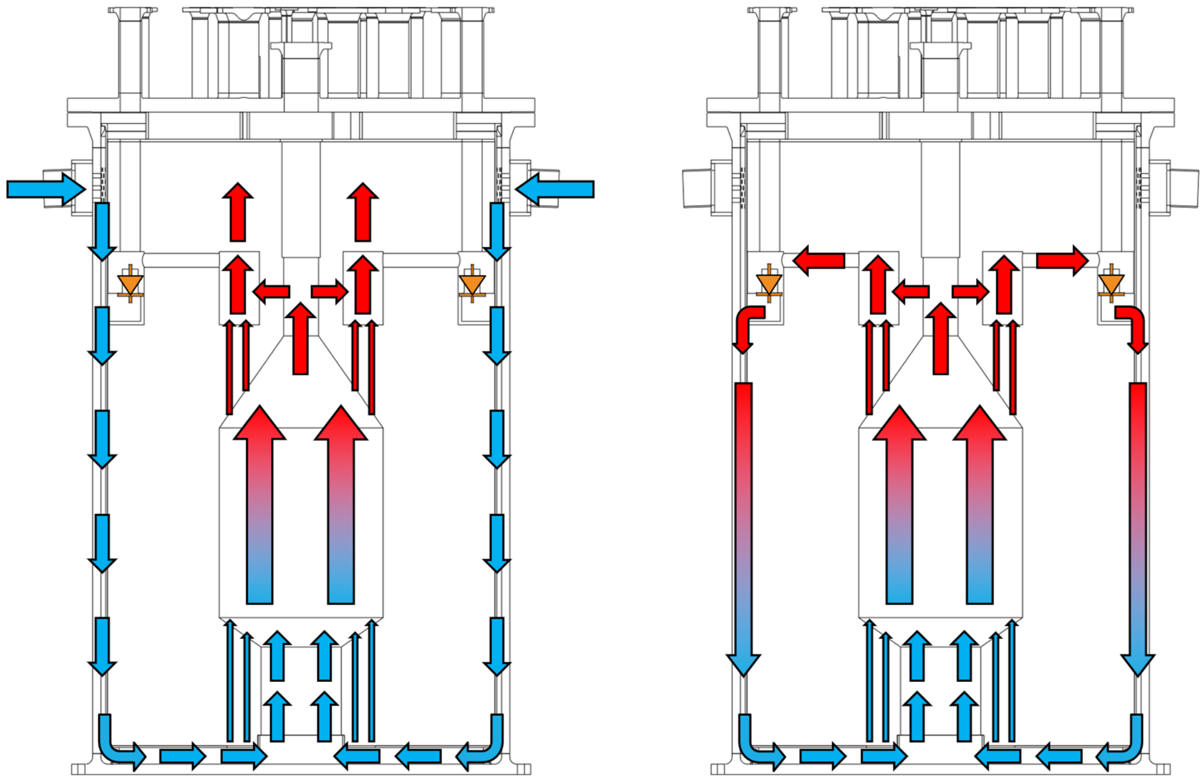
Table 4.6-1: Summary of Thermal Hydraulic Parameters

Parameter	Nominal Value
Core Power (MWth)	35
Reactor Inlet Temperature (°C)	550
Maximum Core Outlet Temperature (°C)	650
Maximum Reactor Mass Flow Rate (kg/s)	210
Core Pressure Drop at Maximum Flow Rate (kPa)	121
Core Volume (m^3)	2.0
Core Packing Fraction (%)	60
Total Pebbles (Fuel and Moderator)	36,000
Power Density (MW/m^3)	17.5

Notes:

1. Value does not account for bypass flow

Figure 4.6-1: Coolant Flow Path



(a) Normal Operation Coolant Flow Path

(b) Natural Circulation Coolant Flow Path

4.7 REACTOR VESSEL SUPPORT SYSTEM

4.7.1 Description

The reactor vessel support system (RVSS) provides structural support to the reactor vessel and supports the full weight of the reactor vessel with fuel and coolant, vessel internals, and all head-mounted components. The system transmits pressure, seismic, and thermal loads to the cavity structures during normal operation and design basis earthquakes. The RVSS provides adequate thermal management to support the vessel's thermal expansion while transitioning from room temperature at assembly to nominal operating temperature for primary coolant fill. The RVSS also supports the vessel's thermal expansion during postulated events.

The RVSS interfaces with the reactor vessel (see Section 4.3), the reactor thermal management system (RTMS) (See Section 9.1.5), and the safety-related portion of the Reactor Building (see Section 3.5). The safety-related portion of the Reactor Building is seismically isolated to reduce seismic loads (see Section 3.5.3). The RVSS is not shared by Unit 1 and Unit 2.

The bottom support consists of a support tray, ledge, support columns, support pads, base plate, vessel connector, and anchoring connector as shown in Figure 4.7-1. All the components are made of 316H stainless steel. The reactor vessel bottom head sits directly on top of the tray and is connected to the tray by the vessel connector to prevent uplift and shear. The ledge around the edge of the tray contains spilled Flibe in case of leakage. The tray is reinforced by 316H SS support columns, which are sized and spaced appropriately to provide structural support for the total weight of the vessel, vessel internals, head components, coolant, and fuel. The support columns are welded onto the support pad, which allows relative sliding with the underlying base plate to accommodate thermal expansion. The support pads have slotted holes to allow relative sliding with the anchoring connectors. The anchoring connectors prevent the reactor vessel and RVSS from uplift and shear. The RVSS is designed and fabricated using the technical guidance in ASME BPVC Section III, Division 5 (2017) (Reference 1) as shown in Table 3.6-2.

The RTMS provides thermal management for the bottom support with a load bearing metallic insulation material, which acts as a thermal break that reduces heat loss and cooling load for the RVSS support columns. The bottom insulation of the RTMS, as shown in Figure 4.7-1, protects the reactor building cavity concrete from thermal effects. The RVSS is also vertically anchored to the foundation through the bottom insulation. The bottom support insulation interface accommodates relative thermal expansion between the support columns and the insulation material.

There are no lateral seismic restraints for the reactor vessel and the head-mounted components. The RVSS is designed to keep the reactor vessel from uplift and shear during seismic events. The design also leverages seismic isolation of the Reactor Building to reduce seismic effects on the reactor vessel, RVSS, and the head-mounted components (see Section 4.3).

4.7.2 Design Basis

Consistent with PDC 2, the RVSS can withstand the effects of natural phenomena and to perform its safety function in the event of a design basis earthquake.

Consistent with PDC 4, the RVSS accommodates the environmental conditions associated with normal operation, maintenance, testing, and postulated events.

Consistent with PDC 74, the design of the reactor structural support system ensures the integrity of the reactor vessel during postulated events to support the geometry for passive removal of residual heat from the core and to permit sufficient insertion of the control and shutdown elements providing for reactor shutdown.

4.7.3 System Evaluation

The RVSS supports the reactor vessel in the event of an earthquake or other natural phenomenon thus ensuring the integrity of the reactor vessel and its ability to retain reactor coolant. The bottom support meets ASCE 43-19 (2019) (Reference 2) and precludes linear buckling in the vessel support columns under static and design basis earthquake loads. The bottom support is also vertically anchored to the cavity to prevent the vessel from uplift during a design basis earthquake. The vessel connectors meet Reference 2 and provide sufficient lateral and uplift support to the vessel and the vessel top head components. The reactor cavity is also seismically isolated to reduce seismic loads. Load combinations for the RVSS and safety-related portions of the Reactor Building are provided in Table 4.7-1 and Table 3.5-1. These design features demonstrate compliance with PDC 2 for the RVSS.

The RVSS is protected from discharging fluids by catch basins. Sensors and probes installed on catch basins including the bottom support tray can be used as a means of leak detection to preclude damage to the RVSS. There are no pressurized piping systems in proximity to the RVSS thus precluding by design any impacts from pipe whip hazards. The RVSS accommodates the reactor vessel temperature loading cycles in combination with relevant mechanical loading cycles to ensure creep-fatigue damages are precluded. The RVSS can also accommodate the growth of the reactor vessel due to thermal expansion between startup and equilibrium conditions. These design features satisfy PDC 4 for the RVSS.

PDC 74 states requires the design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated events (1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown. The RVSS maintains the integrity of the reactor vessel by removing heat via the RTMS, actively during normal operation and passively during postulated events. Fission product decay heat and other residual heat from the reactor core is transferred to the reactor vessel; then to the anchored surface by the RVSS. The support columns of the RVSS are sized and spaced to maximize heat transfer between the bottom support and the environment. The thermal break between the RVSS and the reactor building provided by the bottom support insulation ensures the concrete integrity meets ACI 349-13 to support maintenance and inspection of the vessel bottom head/vessel shell weld and to ensure conditions in the surrounding cavity do not exceed maximum allowable parameters. This demonstrates compliance with PDC 74 for the RVSS.

4.7.4 Testing and Inspection

The RVSS temperature will be monitored during operation for conformance with design limits. The RVSS will be included in an in-service inspection program, which will be submitted at the time of the Operating License Application.

4.7.5 References

1. American Society of Mechanical Engineers, ASME Boiler & Pressure Vessel Code, Section III, Division 5, ”High Temperature Reactors.” 2017.
2. ASCE 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.”
3. ACI 349-13, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary”

Table 4.7-1: Load Combinations for the Reactor Vessel Support System

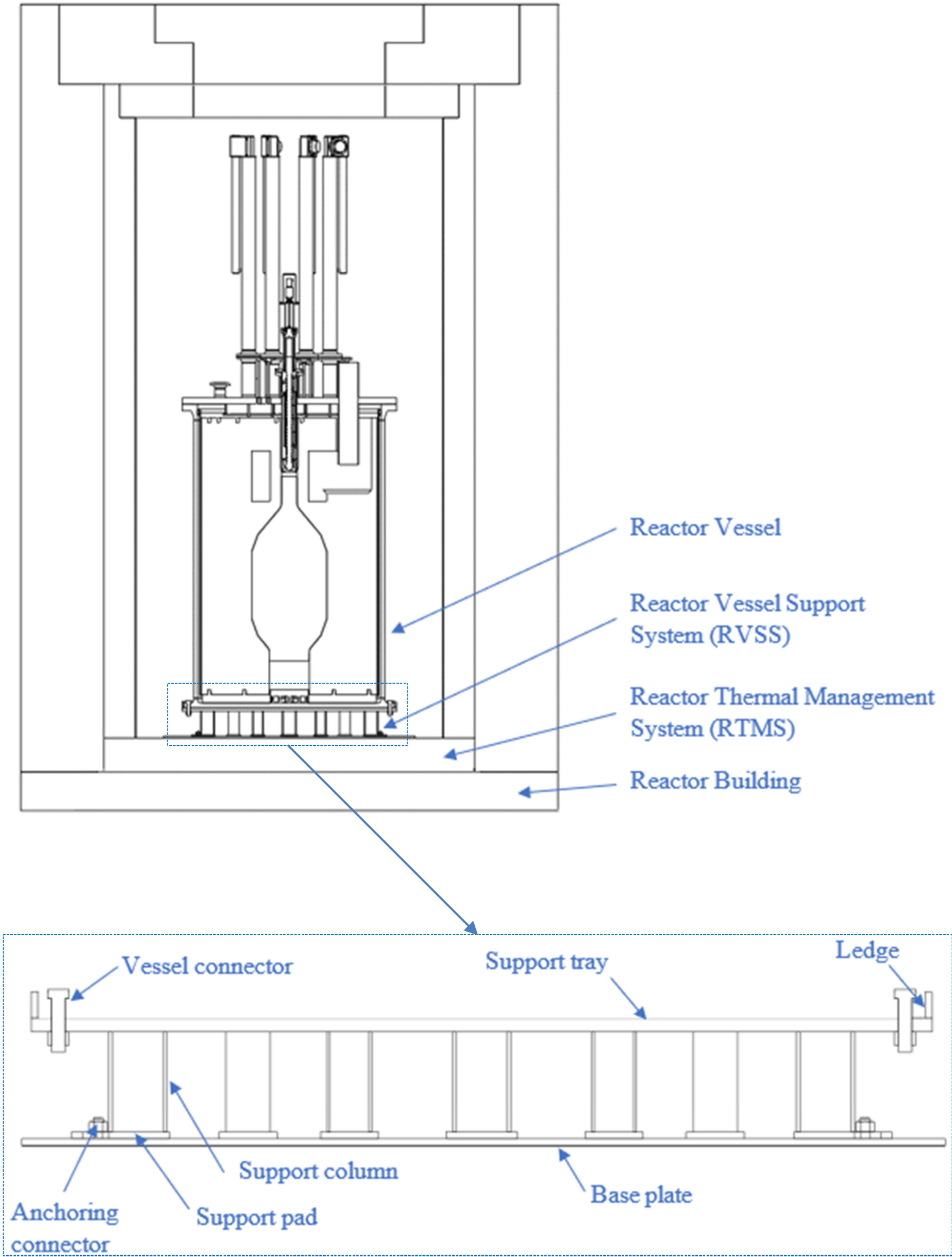
Service Level	Load Combination*
A	$D + L + T_o + R_o$
B	$D + L + T_o + R_o + E_o$ $D + L + T_i + R_i + E_o$
C	$D + L + T_o + R_o + E_{ss}$ $D + L + T_s + R_s + E_{ss}$
D	$D + L + T_a + R_a + W_t$ $D + L + T_a + R_a + E_{ss}$

*Load combination refers to the types of loads considered acting simultaneously. Application of load factors and specific details of load combination effects are per the applicable design standard.

Load Nomenclature:

- D — Dead loads
- L — Live loads
- T_o — Thermal loads during startup, normal operating, or shutdown conditions
- T_i — Thermal loads during Service Level B loadings
- T_a — Thermal loads during Service Level D loadings
- T_s — Thermal loads during Service Level C loadings
- R_o — Pipe reactions during startup, normal operating, or shutdown conditions
- R_i — Pipe reactions during Service Level B loadings
- R_a — Pipe reactions during Service Level D loadings
- R_s — Pipe reactions during Service Level C loadings
- E_o — Loads generated by 1/3 of design basis earthquake (DBE)
- E_{ss} — Loads generated by DBE
- W_t — Accidental loads due to missile impact effects

Figure 4.7-1: Reactor Vessel Support System



Kairos Power

Chapter 5

Heat Transport System

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 5 HEAT TRANSPORT SYSTEMS

5.1 PRIMARY HEAT TRANSPORT SYSTEM

5.1.1 Description

The primary heat transport system (PHTS) transfers heat from the reactor core by circulating reactor coolant between the packed bed of fuel elements (pebbles) and reflector in the reactor core and the intermediate heat transport system (IHTS) (Section 5.2) during normal operations. The PHTS includes a primary salt pump (PSP), an intermediate heat exchanger (IHX), a heat rejection subsystem (HRS), and associated piping. The heat rejection subsystem includes a heat rejection radiator (HRR), heat rejection blower, and associated ducting. The PHTS also includes capability for primary loop auxiliary heating to maintain the reactor coolant in the liquid phase when the reactor core is not generating heat, and capability to drain piping, the IHX, and the HRR to allow cooldown, inspection, and maintenance. The information presented in this section is applicable to both Unit 1 and Unit 2. Each unit has its own PHTS and there are no shared PHTS components between units. A process flow diagram of the PHTS is provided in Figure 5.1-1. The key design parameters for the PHTS are provided in Table 5.1-1.

The primary system functions of the PHTS are non-safety related and include the following:

- Transport heat from the reactor core to the IHTS to support nuclear heat generation and transport during normal operations.
- Contain and direct the reactor coolant flow between the reactor vessel and the IHX.
- Manage thermal transients (overall thermal balance) occurring as part of normal operations.
- Support the heat removal function through the HRR at low power during plant startup and normal shutdown.
- Support the tritium management function through the HRR interface with the TMS.
- Support void fraction limits in the reactor coolant flow through gas separation features, where applicable.
- Accommodate thermal expansion of the system and components in transitioning between the temperature at assembly and operation, and during transients.
- Circulate trace heated coolant during periods when fission heat is not sufficient to ensure minimum acceptable temperatures in the PHTS, including initial heat up.
- Provide capability to drain the PHTS.
- Prevent forced air ingress by the PSP and by the heat rejection subsystem blower when the PSP is not operating.
- Maintain the reactor coolant pressure in the IHX above the IHTS coolant pressure under normal operating conditions.
- Support reactor power level transitions (ramp up and ramp down in power).
- Provide for in-service inspection, maintenance, and replacement activities.

The PHTS interfaces with multiple systems including the reactor systems (e.g., reactor vessel, reactor startup system, and thermal management system), the IHTS, the reactor coolant auxiliary systems (e.g., inert gas system and inventory management system), the instrumentation and control system (e.g., reactor protection system), the plant auxiliary systems (e.g., radiation monitoring system, fire protection system, and remote maintenance and inspection system), the electrical system (e.g., backup power system and normal power system) and civil structures systems and components (e.g., plant site and reactor building). These systems are described in Chapters 4, 7, and 9.

The reactor coolant is maintained at a positive pressure differential with respect to the IHTS during normal operation. If a postulated IHX tube leak were to occur, reactor coolant would be driven into the IHTS to maintain reactor coolant chemistry and physical properties for investment protection.

The primary components of the PHTS are described in the following subsections.

5.1.1.1 Reactor Coolant

The reactor coolant is a chemically stable, molten mixture of fluorine, lithium, and beryllium (Flibe). A description of the reactor coolant material composition, coolant quality requirements, Flibe impurities, and thermophysical properties is provided in the “Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor” Topical Report KP-TR-005 (Reference 1). The reactor coolant performs safety functions associated with reactivity control and fission product retention. The composition of the reactor coolant also enables the reactor core to be designed with a negative coolant temperature coefficient of reactivity. This provides a safety benefit supporting reactivity control, low parasitic neutron absorption for effective fuel utilization, and minimal short-term and long-term activation of the coolant for improved operations and maintenance. The reactor coolant also serves as a fission product barrier providing retention of fission products that escape the fuel particle and fuel pebble barriers for fuel in the reactor core. This additional retention capability contributes to the functional containment and enhanced safety. The circulating activity of the reactor coolant is sampled (see Section 9.1.1) to remain within limits established in the technical specifications.

5.1.1.2 Primary Salt Pump

The PSP is a variable speed, cartridge style pump located on the reactor vessel head that controls system flow rate and pressure in the PHTS under normal operation. The PSP circulates the reactor coolant between the reactor core, where the Flibe is heated as it contacts with the fuel, and the IHX where the heat is transferred to the IHTS. PHTS flow rates are varied based on the operating power of the reactor. The design of the PSP operates continuously at full thermal power flow rates and temperatures, as well as at reduced power and flow rates.

The cantilever pump design extends the shaft down into the reactor coolant while keeping the bearings and seals in a lower temperature region above the coolant. The pump flow discharges horizontally above the reactor vessel head and has a high-point vent that is used for vacuum fill. The pump has a positive pressure inert gas space with a purge gas flow that discharges into the reactor vessel cover gas space. The pump motor rotor is directly mounted on the shaft and operates in the cover gas environment, eliminating the need for conventional shaft seals and providing a hermetic boundary for cover gas. The inert gas system is described in Section 9.1.2.

The design of the pump suction controls and prevents entrainment of cover gas at normal submergence levels. Residual gas in the PHTS at start up is removed by de-entrainment locations in the upper reflector. The pump casing design sets the inlet elevation of the anti-siphon surface for the hot leg when the external PHTS piping is drained and if a leak was to occur in the external portion of the PHTS.

5.1.1.3 Intermediate Heat Exchanger

The IHX serves as the heat transfer interface and coolant boundary between the PHTS and the IHTS. The IHX does not perform any safety-related functions. The reactor coolant is circulated from the PSP outlet nozzle through the primary piping before it enters the IHX, where heat is transferred from the reactor coolant to the intermediate coolant on the IHTS side.

The reactor coolant enters the IHX at approximately 600-650°C and leaves the IHX at approximately 550°C during normal, steady-state operation at full power. After transferring its heat, the reactor coolant leaves the outlet nozzle of the IHX and is returned to the inlet nozzle of the reactor vessel.

The IHX is located at an intermediate elevation between the reactor vessel coolant free surface and the IHTS coolant free surface resulting in a positive pressure differential due to hydrostatic head under shut down conditions. PSP and intermediate salt pump speeds are controlled to maintain positive pressure differential under normal operating modes. A pressure differential measurement between the IHX intermediate coolant inlet pressure (highest intermediate coolant pressure) and reactor coolant outlet pressure (lowest reactor coolant pressure) controls the speed of the PSP and ISP to maintain a positive pressure differential.

5.1.1.4 Primary Loop Piping

The primary loop piping consists of the interconnecting piping and small components not specifically allocated within the other architectural elements. This includes cross connection piping, valves, and interfaces with the inventory management system.

The primary loop piping does not perform any safety-related functions and is not credited to mitigate the consequences of postulated events.

The PHTS piping is made of austenitic stainless steel and designed to accommodate the reactor coolant temperature, pressure, and corrosion properties. The section of piping from the PSP discharge to the IHX inlet is termed the “hot leg” and the section of piping from the IHX outlet to the reactor vessel inlet is termed the “cold leg.” An anti-siphon feature is implemented in the design that can break the siphon from the reactor vessel if a leak in the PHTS occurs.

5.1.1.5 Primary Loop Thermal Management

The thermal management feature provides non-nuclear heating and insulation to the PHTS as needed for various operations including plant startup, plant shutdown, and supplemental heating during normal operation. The auxiliary heating maintains the PHTS piping, the IHX, and the HRR at or above the trace heating setpoint temperature. The source of the heat depends on the subsystem or component requiring the heat. For example, electrical heating is used in some areas of the plant that would be susceptible to coolant freezing with the use of insulation alone. Sufficient heating is provided to maintain reactor coolant temperature in external piping, the IHX, and the HRR above freezing throughout the filling, operation, and draining processes.

5.1.1.6 Heat Rejection Subsystem

The HRS provides for non-safety-related heat transfer from the reactor coolant to the atmosphere at low power during plant startup and normal shutdown conditions. Within the HRS, the HRR serves as the heat transfer interface for this function. The HRR is placed in series with the IHX along the reactor coolant flow path within the PHTS.

The HRS consists of the HRR, heat rejection blower, and associated ducting and thermal management. During normal plant operations, the heat rejection blower is off and the air flow path in the HRS ducting is blocked to suppress heat losses through the HRS. In these conditions, heat is transferred to the IHTS through the IHX. During plant startup and normal shutdown conditions, the HRS ducting is open and the heat rejection blower is active. In these conditions, heat is transferred to air passing through the HRS, which leaves the plant via a stack.

Tritium can permeate through the HRR into the surrounding air during both normal plant operations with the heat rejection blower off, and startup and normal shutdown conditions with the heat rejection blower on. During normal operations, air surrounding the HRR is isolated between the inlet and outlet of the HRS. The air is recirculated through a subsystem of the TMS which captures tritium permeating through the HRR and returns air at appropriate temperatures to provide thermal management for the HRR and limit heat losses. Further details of the tritium capture subsystem of the TMS which interfaces with the HRR are provided in Section 9.1.3. During startup and normal shutdown conditions, tritium capture is not conducted by the TMS and permeation losses through the HRR are released through the HRS as a gaseous effluent.

The transition from power operation to normal shutdown cooling involves a programmed runback (see Section 7.2) of the PSP and activation of the heat rejection blower to minimize the thermal transient experienced by the reactor vessel and the PHTS.

The heat rejection blower is tripped concurrent with the PSP to prevent forced air ingress during postulated HRR tube failures.

5.1.2 Design Basis

Consistent with PDC 2, the safety-related SSCs located near the PHTS are protected from the adverse effects of postulated PHTS failures during a design basis earthquake.

Consistent with PDC 10, the design of the reactor coolant supports the assurance that specified acceptable system radionuclide release design limits (SARRDLs) are not exceeded during any condition of normal operation, as well as during any unplanned transients.

Consistent with PDC 12, the design of the reactor coolant, in part, ensures that power oscillations cannot result in conditions exceeding specified acceptable SARRDLs.

Consistent with PDC 16, the design of the reactor coolant, in part, provides a means to control the release of radioactive materials to the environment during postulated events as part of the functional containment design.

Consistent with PDC 33, the design of the PHTS includes anti-siphon features to maintain reactor coolant inventory in the event of breaks in the system.

Consistent with PDC 60, the design of the PHTS supports the control of radioactive materials during normal reactor operation.

Consistent with PDC 70, the design of the PHTS supports the purity control of the primary coolant by limiting air ingress.

Consistent with 10 CFR 20.1406, the design of the PHTS, to the extent practicable, minimizes contamination of the facility and the environment, and facilitates eventual decommissioning.

5.1.3 System Evaluation

The design of the non-safety-related PHTS is such that a failure of components of the PHTS does not affect the performance of safety-related SSCs due to a design basis earthquake. In addition to protective barriers, the PHTS pipe connections to the reactor vessel nozzles have sufficiently small wall thickness, such that if loaded beyond elastic limits, inelastic response occurs in the PHTS piping, which is non-safety-related. These features, along with the seismic design described in Section 3.5, demonstrate conformance with the requirements in PDC 2.

While the primary side of the PHTS is a closed system, there are conceivable scenarios that may result in the release of radioactive effluents. The fuel design locates the fuel particles near the periphery of the fuel pebble, enhancing the ability of the fuel to transfer heat to the coolant. The thermal hydraulic analysis of the core (see Section 4.6) ensures that adequate coolant flow is maintained to ensure that

SARRDLs, as discussed in Section 6.2, are not exceeded. These features demonstrate conformance with the requirements in PDC 10.

The reactor coolant is designed, in part, to ensure that power oscillations cannot result in conditions exceeding specified acceptable system radionuclide design limits. The PHTS is designed such that (1) reactor coolant inlet temperature and coolant mass flow rate oscillations are suppressed or readily detected, (2) the reactor coolant in the PHTS does not experience significant gas entrainment (to avoid unexpected coolant void reactivity feedback), (3) the reactor coolant remains within defined specifications and additional neutron absorbers are not added to or removed from the system, and (4) the reactor coolant has high thermal inertia, making the reactor resistant to thermal-hydraulic instability events. These features, in part, demonstrate conformance with the requirements in PDC 12.

The functional containment is described in Section 6.2. The design relies primarily on the multiple barriers within the TRISO fuel particles to ensure that the radiological dose at the exclusion area boundary as a consequence of postulated events meets regulatory limits. However, the reactor coolant also serves as a distinct physical barrier for fuel submerged in Flibe by providing retention of fission products that escape the fuel. The design of the reactor coolant composition provides, in part, a means to control the accidental release of radioactive materials during normal reactor operation and postulated events (PDC 60), and supports, in part, demonstration of the functional containment aspects. The design aspects of the reactor coolant are discussed in Reference 1. The Flibe also accumulates radionuclides from fission products, and transmutation products from the Flibe and Flibe impurities. The retention properties of the Flibe are credited in the safety analysis as a barrier to release of radionuclides accumulated in the coolant, and radionuclide concentration is limited by technical specifications. The transport of radionuclides through Flibe is based on thermodynamic data that will be justified in the application for an Operating License. These features demonstrate conformance with the requirements in PDC 16.

The PSP casing design sets the inlet elevation of the anti-siphon surface for the hot leg should a leak occur in the external portion of the PHTS. In the event of a break in the external portion of the PHTS hot leg or breaches of inventory management system piping connected to the PHTS (see Section 9.1.4), reactor coolant level is expected to decrease and the cover gas moves into the pump well to break the siphon. This precludes coolant from being siphoned below the elevation of the PSP casing. These anti-siphon features demonstrate compliance with PDC 33.

Significant forced air ingress into the PHTS is excluded by design basis. Air ingress could affect the inventory of reactor coolant in the reactor vessel as well as affect the purity of the reactor coolant. Design features of the heat rejection subsystem and the reactor trip system will limit the quantities of air ingress during system leakage events by tripping the heat rejection blower and tripping the PSP. These design features satisfy PDC 33 and PDC 70. The effects of non-forced air ingress into the PHTS on safety-related Hermes components are bounded by the results of materials qualification programs as described in Section 4.3.

The fouling and plugging of the reactor coolant flow path through the vessel as a result of a reduction in coolant purity is not expected. However, the temperature of the reactor coolant in the downcomer and core can be monitored to determine decrease in heat removal capability that could occur as a result of fouling or plugging of passages. This demonstrates conformance with PDC 70.

The design of the PHTS controls the release of radioactive materials in gaseous and liquid effluents in the event the PHTS working fluid is inadvertently released to the atmosphere via leaks in the piping system. The PHTS SSCs that are part of the reactor coolant boundary are designed to the ASME B31.3 Code (for the piping) and ASME BPVC Section VIII (for the HRR and the IHX) such that leaks are unlikely.

Means are provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage in the PHTS SSCs. A postulated IHX tube failure could cause Flibe to leak into the intermediate coolant, as the Flibe is maintained at a higher pressure than the intermediate coolant, resulting in a spread of contamination to the IHTS. Such an event would be detected by a loss of inventory in the inventory management system (Section 9.1.4) and by detection capability in the IHTS. The compatibility of the primary to intermediate coolant interaction will be demonstrated as part of an application for an Operating License.

Tritium and other radionuclides will be present in the reactor coolant as part of normal operations of the plant. Control measures will be taken to minimize the release of radioactive material and ensure that they are also below allowable limits (see Section 9.1.3). Tritium (HT, T_2) in the reactor coolant will normally permeate through the IHX and HRR heat transfer surfaces (See Section 5.2 and Section 9.1.3). The reactor coolant contains radionuclides as a result of releases from defective fuel particles, as well as a result of activation of impurities in the Flibe itself. The reactor coolant thermophysical properties, impurities and limitations are provided in Reference 1. The reactor coolant activity is sampled during normal operations as described in Section 9.1.1. Postulated failures in the PHTS could cause the reactor coolant or cover gas to leak into the reactor building cell gas space and be released. Such events are evaluated in Section 13.1. These features demonstrate conformance with the requirements in PDC 60.

The PHTS (reactor coolant) contains radiological contaminants. Therefore, the design of the system minimizes contamination and supports eventual decommissioning, consistent with the requirements of 10 CFR 20.1406, as described in Chapter 11.

5.1.4 Testing and Inspection

Descriptions of any tests and inspections of the PHTS will be provided with the application for an Operating License.

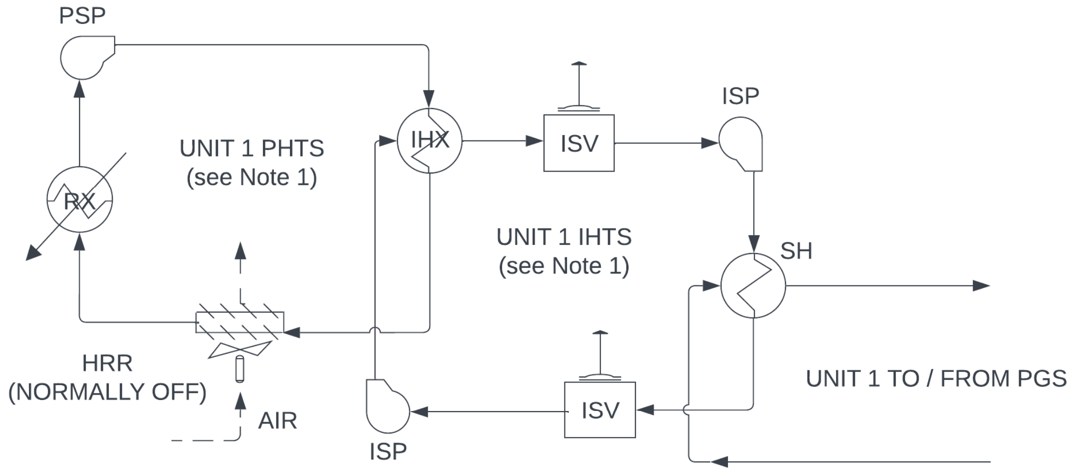
5.1.5 References

1. Kairos Power LLC Topical Report, “Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor” KP-TR-005-P-A (ML 20219A591). July 2020.

Table 5.1-1: Key Design Parameters of the Primary Heat Transport System (at nominal full power)

Parameter	Value
Thermal duty	35 MWth
Number of IHXs	1
Number of hot legs	1
Number of cold legs	2
Primary loop line size	8-12 in nominal pipe size
PHTS hot leg temperature	600-650^{o}C
PHTS cold leg temperature	550^{o}C
Nominal flow rate	210 kg/s
PHTS design pressure	525 kPa(g)

Figure 5.1-1: Primary Heat Transport System and Intermediate Heat Transport System Process Flow Diagram



Note 1: Duplicated and separate Unit 2 PHTS and Unit 2 IHTS to/from common PGS

5.2 INTERMEDIATE HEAT TRANSPORT SYSTEM

5.2.1 Description

The Intermediate Heat Transport System (IHTS) transfers heat from the PHTS (Section 5.1) by circulating intermediate coolant between the cooling side of the IHX and the power generation systems (Section 9.9) during normal plant operation. The IHTS includes intermediate salt pumps (ISPs), intermediate salt vessels (ISVs), a superheater, and associated piping. The IHTS transports tritium from the IHX to the tritium management system (TMS) in the cover gas portion of the ISVs. The TMS is described in Section 9.1.3. The IHTS also provides for fill/draining control of the IHTS piping, IHX, and superheater tube side.

The information presented in this section is applicable to both Unit 1 and Unit 2. Each unit has its own IHTS and there are no shared IHTS components between units. A process flow diagram of the IHTS showing both units is provided in Figure 5.1-1. The key design parameters for the IHTS are provided in Table 5.2-1.

The primary system functions of the IHTS are non-safety related and include the following:

- Transport heat from the PHTS to the steam system.
- Manage thermal transients (overall thermal balance) occurring as part of normal operations.
- Maintain intermediate coolant pressure below primary coolant pressure within the IHX.
- Facilitate tritium transfer from the intermediate coolant to the TMS to capture tritium permeating into the IHTS.

There is one safety-related function associated with the IHTS:

- Relieve IHTS pressure in the event of a superheater tube leak or rupture event.

The design of the IHTS allows for on-line monitoring, in-service inspection, maintenance, and coolant replacement activities. The IHTS design includes an intermediate inert gas system to control intermediate coolant chemistry, to minimize corrosion and control and recover tritium. Inert gas within the ISVs is circulated through the TMS to capture tritium in the gas (see Section 9.1.3). Gas composition and impurities within the ISVs inert gas are controlled to maintain conditions which facilitate tritium capture. The intermediate inert gas system is designed to support keeping the intermediate coolant pressure in the heat exchangers lower than the pressure in the PHTS.

The intermediate coolant is a eutectic mixture of sodium fluoride and beryllium fluoride (57mol%NaF-43mol%BeF_2, referred to as “BeNaF”). BeNaF has similar characteristics to Flibe in that it is thermodynamically stable, is compatible with structural materials, and has analogous chemical properties to the primary Flibe coolant.

The ISPs provide the motive force for the circulation of intermediate coolant between the IHX and the superheater and provide the needed pressure and flow rate in the IHTS. The intermediate coolant is circulated through the superheater where heat is transferred to saturated steam to produce superheated steam in the power generation systems (see Section 9.9).

The IHTS is equipped with safety-related rupture disks located in the intermediate inert gas system, made of austenitic stainless steel, which prevents overpressure in the IHTS during a postulated superheater tube leak or rupture event.

The intermediate piping serves as the flow conduit within the IHTS. The design of the piping accommodates continuous operation at full thermal power and operates under partial load conditions at reduced flow rate.

The design of the IHTS piping includes provisions for filling, draining, and high point venting, and accommodates thermal expansion between the ISPs, the ISVs, and the superheater.

The IHTS contains an auxiliary heating system to provide non-nuclear heating as needed for plant startup, shutdown, and supplemental heating during normal operation. The auxiliary heating maintains the IHTS piping at or above the trace heating setpoint temperature. The source of the heat depends on the subsystem or component requiring the heat. The selected heat source will be described in the application for an Operating License.

5.2.2 Design Basis

Consistent with PDC 60, the IHTS includes features that support the control of radioactive materials during normal reactor operation.

Consistent with PDC 73, the IHTS includes a passive barrier (IHX) for the reactor coolant system that is chemically compatible with the IHTS coolant and IHTS features that support the control of radioactive materials during normal reactor operation. The IHTS provides two passive barriers (IHX and superheater) between Flibe in the PHTS and steam in the steam system.

Consistent with 10 CFR 20.1406, the IHTS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and to facilitate eventual decommissioning.

5.2.3 System Evaluation

Tritium will be present in the intermediate coolant as part of normal operations of the plant. Control measures will be taken to minimize the release of radioactive material and ensure that it is also below allowable limits. Tritium which permeates through the IHX heat transfer surface is expected to enter the intermediate coolant in the chemical form of HT or T_2. As described in Section 9.1.3, anhydrous hydrogen fluoride will be added to the intermediate inert gas system to convert the tritium to a tritium fluoride that will move into the gas space of the ISVs. The TMS will capture tritium from the gas mixture. Removal of tritium from the ISV gas spaces reduces tritium inventory that is available for release, as described in Section 9.1.3. Postulated failures of the IHTS could cause intermediate coolant or cover gas to leak into the reactor building. Such events are evaluated in Section 13.1. These features demonstrate conformance with the requirements in PDC 60.

The IHTS coolant has the potential to be contaminated with Flibe due to a postulated leak of the IHX, as the Flibe is maintained at a higher pressure than the intermediate coolant. However, the two fluids are chemically compatible. Flibe is separated from the water in the power generation system by two passive barriers, the IHX and superheater boundaries. The IHTS is provided with safety-related rupture disks to mitigate the effects of a postulated superheater tube leak or tube rupture event. These features demonstrate conformance with the requirements in PDC 73.

The IHTS piping is designed to the ASME B31.3 Code. The superheater is designed to ASME BPVC Section VIII. The ISVs are designed to ASME BPVC Section VIII. The IHTS coolant has the potential to be contaminated with tritium or other radioactive materials in a postulated leak from the PHTS into the IHTS, via the IHX. As such, the IHTS includes features that support monitoring radioactive material releases from breaks and leaks in the piping system or via pressure relief equipment. Therefore, the design of the system minimizes contamination and supports eventual decommissioning, consistent with the requirements of 10 CFR 20.1406, as described in Chapter 11.

5.2.4 Testing and Inspection

Descriptions of any tests and inspections of the IHTS will be provided with the application for the Operating License.

5.2.5 References

None

Table 5.2-1: Key Design Parameters of the Intermediate Heat Transport System

Parameter	Value
Thermal duty	35 MWth
Number of IHXs	1
Number of hot legs	1
Number of cold legs	1
Number of superheaters	1
IHTS hot leg temperature	580 - 615°C
IHTS cold leg temperature	490 - 525°C
Nominal flow rate	140 - 250 kg/s
IHTS design pressure	Near ambient pressure

Chapter 6

Engineered Safety Features

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 6 ENGINEERED SAFETY FEATURES

6.1 SUMMARY DESCRIPTION

This section describes the engineered safety features (ESF) designed to mitigate the consequences of postulated events, ensuring that any potential dose consequences are within acceptable values. The ESFs credited for mitigation of postulated events are the functional containment and the decay heat removal system (DHRS).

Functional containment refers to an approach to radionuclide retention that includes multiple barriers between radioactive material at risk for release (MAR) and safety features inherent in KP-FHR technology. For fuel inside the reactor core, which can have high decay heat generation, the multiple barriers to release include the TRISO layers of the fuel (described in Section 4.2.1) and the Flibe (described in Section 5.1). For fuel in the pebble handling and storage system (PHSS) (described in Section 9.3), which has low decay heat generation, the TRISO layers of the fuel (described in Section 4.2.1) provide the barriers to release. The inherent safety features of KP-FHR technology that facilitate following the functional containment approach include a near-atmospheric operating pressure (described in Section 5.1), a robust fuel design with radionuclide retention capabilities qualified to withstand peak temperatures of 1600 °C (described in Section 4.2.1), and a coolant design with a high boiling point (described in Section 5.1). The functional containment, described in Section 6.2, is credited with radionuclide retention in the postulated events described in Section 13.1.

The DHRS is the ESF that removes heat from the reactor vessel in postulated events where the normal heat rejection system is unavailable. The DHRS, along with natural circulation flow within the core, provides heat removal from fuel in the reactor core for postulated events via thermal radiation and convection without the need for external sources of electrical power or operator intervention. The heat removal provided by the DHRS and natural circulation is adequate to ensure that the vessel temperature remains below design limits and the fuel integrity is not challenged. The DHRS consists of four independent trains to provide redundancy in the event of a single failure. Figure 6.3-1 illustrates the general arrangement of the DHRS relative to the reactor system. There is one DHRS per unit, and the DHRS does not share components between units. The design information presented in Section 6.3 applies to both units. The DHRS is credited for decay heat removal from the reactor vessel in all of the limiting postulated events described in Section 13.1.

6.2 FUNCTIONAL CONTAINMENT

The 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 non-LWR technologies differ from LWRs in operating conditions, coolants, and fuel forms that allow 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. This section describes the functional containment, which is made up of physical barriers, operating conditions, coolant design, and fuel form that limit the potential release of radioactive material.

The majority of the radioactive material at risk for release in a KP-FHR is held up by design in the TRISO fuel, which resides in the reactor core where the fuel can have high decay heat generation, or in the PHSS where fuel has low decay heat generation. As described in Section 4.2.1, the fuel design consists of TRISO-coated particles embedded in an annular shell inside a spherical pebble to form a fuel element. The physical design features of the TRISO fuel, including the melting temperature and TRISO layers contribute to the functional containment retention of fission products. The TRISO fuel design temperature of 1600 °C provides significant margin to failure in transient conditions. The fuel pebble design provides protection of the TRISO particles from mechanical damage as described in Section 4.2.1. The TRISO fuel particle form provides a set of physical barriers to retain the radionuclides, consisting of a porous carbon buffer layer, a dense inner pyrolytic carbon (IPyC) layer, a silicon carbon (SiC) layer, and a dense outer pyrolytic carbon (OPyC) layer. Each layer of the TRISO fuel form is a barrier that can prevent the release of radionuclides from the fuel. A description of the design, evaluation, and testing of the fuel for its effectiveness at retaining radionuclides is provided in Section 4.2.

When the fuel is submerged in the reactor coolant (Flibe) in the reactor core, the retention properties of Flibe act as an additional physical barrier for release of radionuclides as discussed in Section 5.1. During reactor operation, the Flibe will accumulate radionuclides from fission products that escape from defective layers of the TRISO fuel, and transmutation products from Flibe impurities including uranium. The retention properties of the Flibe are credited in the safety analysis for selected radionuclides accumulated in the coolant that are not aerosolized or evaporated due to postulated event conditions. Section 5.1 provides a discussion of the radionuclide retention capabilities for the Flibe reactor coolant.

The operating conditions of the primary system contribute to functional containment. The primary heat transport system, described in Section 5.1, operates at near-atmospheric pressure, preventing the potential energetic releases normally associated with highly pressurized primary systems.

The specified acceptable system radionuclide release design limits (SARRDLs) must not be exceeded to ensure fuel failures do not exceed expected values and result in an unacceptable dose during normal operations or a postulated event. The evaluation of fuel performance is discussed in Section 4.2.1. The fuel performance evaluation shows significant margin to failures that could occur for fuel in the reactor core during postulated events. Radionuclides that escape the fuel during normal operation, when it is submerged in Flibe will contribute to the circulating activity of the Flibe, as discussed in Section 5.1.1. The circulating activity of the Flibe and the radioactive effluent from the gas space are expected to be controlled by technical specification, as described in Chapter 14. The maximum hypothetical accident, described in Section 13.1.1, assumes a limiting value for circulating activity, and provides the dose consequences that bound all postulated events. The SARRDLs are met by controlling the reactor conditions (e.g., temperature and flux) that result in limiting allowable fuel conditions. The safety limits in Chapter 14 will ensure that the SARRDLs are not exceeded, and potential dose consequences remain below dose targets (dose targets will be set based on emergency plan). The SARRDLs and technical specifications will be described in the application for an Operating License.

The design bases, evaluation, and testing of the functional containment features described in this section are described in Chapter 4 and Chapter 5. Chapter 13 evaluates the integrated functional containment approach for fuel in the reactor core, fuel in the PHSS, and other MAR by analyzing the boundary dose associated with a maximum hypothetical accident. Section 13.2.1 demonstrates that this functional containment approach is sufficient to maintain acceptable dose consequences to the public.

6.3 DECAY HEAT REMOVAL SYSTEM

6.3.1 Description

The decay heat removal system (DHRS) removes residual decay heat from the reactor core during normal and off-normal conditions. The DHRS is credited in Chapter 13 for decay heat removal during postulated events that assume the primary heat transport system is unavailable, including the maximum hypothetical accident. The portions of the DHRS that must function to perform the decay heat removal credited in Chapter 13 are designated as safety-related and are all passive. There are no active safety-related portions of the DHRS, and the DHRS does not require electrical power to perform safety functions during postulated events. The DHRS is an ex-vessel system that continuously operates when the reactor is operating above a threshold power by removing energy from the vessel wall via thermal radiation and convective heat transfer to water-based thermosyphons. Inventory in the thermosyphons is boiled off and vents directly to the atmosphere outside of the reactor building.

The DHRS consists of annular thermosyphon thimbles in the reactor cavity, steam separators, and water storage tanks. These components are arranged into four independent cooling trains with inventory sufficient to sustain passive operation of the DHRS for up to 7 days as needed to mitigate a postulated event where normal cooling systems are unavailable. Each train is composed of one water storage tank, one steam separator, and six thimbles. The general arrangement of the DHRS is illustrated in Figure 6.3-1.

The operation of the DHRS is governed by two operational modes. When reactor power is less than a specified threshold power, parasitic losses from the reactor vessel due to convective losses from air ingress and parasitic thermal-radiation and conduction losses through solid structures are sufficient to maintain vessel temperatures below the design limit during a postulated event when the PHTS is unavailable. When the reactor power is above the threshold power, supplemental cooling by the DHRS is required. This threshold power depends on the reactor power history due to the accumulation of fission products in the core as a function of power. The threshold power is nominally 10 MW for a fresh core. As such, the DHRS operating modes are defined as:

- Low Decay Power Operation (Reactor power < threshold power)
  Thermosyphon thimbles in the reactor cavity are dry and isolated from the rest of the system. Water is held in four separate water storage tanks (one for each DHRS train) located above the thimbles and outside of the reactor cavity. Decay heat removal is achieved through parasitic cavity losses.
- High Decay Power Operation (Reactor power > threshold power)
  The thimbles are filled with water and the connected steam separator contains a free surface below the thimble outlet and above the thimble inlet. The separator is continuously and passively replenished from the water storage tank as water in the thimbles is boiled off and vented to atmosphere outside the reactor building, thereby removing heat from the reactor vessel.

These operating states require a transition period. The transition period occurs at the threshold power, where decay heat loads exceed the removal rate by natural parasitic losses. The isolation valves on the thimble feedwater lines open, which allows water to flow from the water storage tank to the thimbles, as indicated by a positive flow rate. The peak flow rate is limited by frictional losses due to the line sizes and gravitational head associated with the water storage tank locations. The temperature of the evaporator tubes contained in the thimbles decreases from standby temperature (550 °C) during low decay power operation to the nominal boil-off operating temperature (100 °C) as the evaporator is wetted. The transient quenching process time is dependent on the thimble feedwater flow rate. Steady-state conditions occur upon completion of the fill with a pseudo-stable liquid level in the separators.

The continuous operation of the DHRS does not require a control actuation to transition from normal operation to passive heat removal. However, event monitoring and the capability for active actuation are provided. The primary interfacing systems through which these occur are described in Chapter 7.

The DHRS is located in the reactor building, which is described in Section 3.5 and contains the reactor cavity and the reactor cell. The DHRS thimbles and steam separators are located within the reactor cavity, but do not have direct contact with the reactor vessel shell. Energy is transferred from the vessel to the DHRS through thermal radiation and convection. The reactor auxiliary heating system (RAHS) is located in the free space between the reactor vessel and the reactor cavity insulation (see Section 9.1.5), but the overall performance of the RAHS does not adversely affect the DHRS removal efficiency because it is deactivated while the DHRS is actively removing heat. The water storage tanks are located outside of the reactor cavity, within the reactor cell. The primary biological shield is a concrete structure which separates the reactor cavity from the reactor cell. This provides direct structural support for the DHRS thimble units and separation and shielding of the water storage tanks from the reactor cavity environments. It also provides through-ports for the steam return and thimble feedwater lines. The primary biological shield is described in Section 4.4. The DHRS primary mode of heat removal is venting steam produced in the thimbles to the atmosphere through the water storage tanks.

The primary components of the DHRS are described in the following subsections.

6.3.1.1 Water Storage Tanks

The DHRS contains four water storage tanks which supply cooling inventory to the DHRS thimbles. These tanks are located outside of the reactor cavity, within the reactor cell, at a higher elevation than other DHRS components. This location enables gravity-driven flow to the thimbles and steam separators. Each water storage tank is coupled to a set of six thimbles through a feedwater line and steam return line, which pass through the primary biological shield. These lines are distributed to individual thimbles through the steam separator located in the upper reactor cavity.

At least three storage tanks must be available for the DHRS to adequately perform its function during postulated event conditions. Each tank holds sufficient inventory such that the thimbles connected to it may be operated continuously for up to 7 days as needed to mitigate postulated events resulting in a loss of the water storage tank feedwater supply. In addition, tank water level is monitored to ensure DHRS operability. Each storage tank is located in an independent location such that damage at one location does not preclude operation of the entire DHRS when required for decay heat removal. This location also provides additional assurance that failures of the water storage tank do not result in leaking into the reactor cell, and that vented or leaked water and steam do not mix with Flibe.

The key water storage tank parameters are provided in Table 6.3-1.

6.3.1.2 Steam Separators

The steam separators provide an interface between the water storage tanks and the individual thimbles that the tanks supply. The steam separator achieves this function by controlling the water level inside its volume through the use of a passive float valve located on the thimble feedwater line. The controlled free surface in the separator is located above the thimble feedwater port and below the steam vent port. The throughput of water is therefore a function of the boil-off rate in the thimbles. Figure 6.3-2 provides a notional diagram of the DHRS separator and float valve.

The float valve consists of a free hollow float which blocks the feedwater line when water level exceeds a threshold value and allows for continuous flow at all other float positions. There are no independent moving mechanical parts beyond the float itself. The valve is designed with sufficient reliability to support the safety case and provide a passively controlled flow of feedwater to the thimbles. The valve is fail-open by design with sufficient redundancy to ensure reliable operation upon demand. Fail-open performance floods the separator volume upon failure and does not affect the net heat removal performance of the thimbles.

The separators are contained within the leak barrier (described in Section 6.3.1.4); therefore, failures of the separator pressure boundary do not preclude the heat removal function of the DHRS. The water ejected from the separator due to a failure of the pressure boundary is captured in the monitored leak barrier. This initiates shutdown of the reactor if it has not already occurred. The leak barrier is a pressure boundary, which ensures that water does not leak directly into the reactor cavity or cell.

The key steam separator parameters are provided in Table 6.3-2.

6.3.1.3 Thimbles

The DHRS thimbles are annular thermosyphons located circumferentially around the outside of the reactor vessel. The thimbles remove heat from the reactor vessel through continuous boil-off of the thimble feedwater supply. A thimble consists of a centrally located guide tube contained within an evaporator tube. The entire unit is fully enclosed within an outer thimble casing, which is part of the leak barrier (see Figure 6.3-3). Heat from the reactor vessel is incident upon the leak barrier, which re-irradiates to the evaporator tube. Fluid is fed from the steam separator to the guide tube and back up the evaporator tube through buoyancy-driven flow. The density differential associated with this flow is developed in the evaporator region, where heat is absorbed in the fluid via convective heat transfer, causing nucleation and flow boiling. The two-phase mixture is ejected into the steam separator, which returns liquid to the static level and allows steam to flow freely out the steam return line. The fluid re-circulation and steam production rate in the thimble is a function of the reactor vessel surface temperature, resulting in a variable flow rate that accommodates the reactor vessel conditions.

The thimbles are supported by the weld joint to the steam separators and are seismically restrained. The thimbles are located in the free gas space between the vessel shell surface and the insulation lining the primary biological shield. The distribution of thimbles is such that failure of all six thimbles from a single train does not cause the reactor vessel to exceed its temperature limits.

The thimbles include an outer casing, which functions as part of the leak barrier system as described in Section 6.3.1.4. Individual thimbles may be either plugged or replaced during maintenance periods. Each train of DHRS includes one redundant thimble, such that the loss of a single thimble will not inhibit operation of the train.

The key thimble parameters are provided in Table 6.3-3.

6.3.1.4 Leak Barrier

Components in the reactor cavity are designed to prevent water leaks and flooding. For this reason, DHRS components located inside the reactor cavity are dual-walled. This includes the thimbles, separators, and thimble feedwater and steam-return piping. The outer casing of the thimbles serves as a dual-wall. The continuous and connected dual-wall may be pressurized for periodic leak checking of the gas region during normal operation. This confirms integrity of the water pressure boundary and the leak barrier. In addition, continuous leak detection of the internal water pressure boundary is possible by monitoring for a relative rise in humidity in the gas space and a drop in the external surface temperature, which would indicate the formation of a leak. Therefore, this system provides a reliable mechanism for prevention of flooding into the reactor cavity. This secondary barrier also provides protection for the DHRS from external hazards associated with Flibe coolant leaking from the PHTS in the event of a failure.

The leak barrier is designed to meet the same pressure and temperature conditions of the DHRS pressure boundary, as described in Table 6.3-3. The outer casing of the thimbles is part of the leak barrier. The leak barrier terminal boundary is outside the reactor cavity and above the free surface of water in the DHRS water storage tanks. This ensures that leaked water can continue to be held in the event of a failure of the primary DHRS pressure boundary. Water in the leak barrier will continue to boil off and remove heat, as in the primary DHRS pressure boundary. However, in the event that leaks are detected in this region, the reactor will be shut down to determine the source and repair as needed.

6.3.2 Design Bases

Consistent with PDC 1, the safety-related portions of the DHRS are designed, fabricated, and tested in accordance with generally recognized codes and standards.

Consistent with PDC 2, the DHRS is designed to perform its safety function in the event of a safe-shutdown earthquake and other natural phenomena hazards.

Consistent with PDC 3, the DHRS is designed to perform its safety function in the event of a fire hazard.

Consistent with PDC 4, the DHRS is designed to perform its safety function in the environmental conditions associated with normal operation, maintenance, testing and postulated events.

Consistent with PDC 10, the DHRS is designed to provide an adequate amount of heat removal to ensure that the specified acceptable system radionuclide release design limits (SARRDLs) are not exceeded during normal operation including postulated events.

Consistent with PDC 34, the DHRS will transfer an adequate amount of decay heat from the reactor core, such that the SARRDLs are not exceeded during normal and off-normal operations.

Consistent with PDC 35, the DHRS is designed to remove an adequate amount of decay heat during and following postulated events.

Consistent with PDC 36, the DHRS is designed to allow for periodic inspection of components to ensure the integrity and capability of the system.

Consistent with PDC 37, the DHRS is designed to permit appropriate periodic functional testing to ensure the structural integrity, operability, and performance of the system.

6.3.3 System Evaluation

Selected portions of the DHRS, as described in Section 6.3.1, perform a safety-related heat removal function for postulated events described in Chapter 13. These components are designed to the codes and standards shown in Table 6.3-4 and the Quality Assurance Program requirements described in Section 12.9. These features demonstrate conformance with the requirements in PDC 1.

The DHRS is primarily located in the safety-related portion of the reactor building. Section 3.5 discusses design features to address the effects of postulated seismic events on safety-related SSCs. The DHRS steam vent lines are not safety-related but may cross the isolation moat discussed in Section 3.5. SSCs that cross a base-isolation moat may experience differential displacements as a result of seismic events. The steam vent lines are designed so that postulated failures from differential displacements do not preclude a safety-related SSC from performing its safety function. Design features addressing differential displacement are discussed in Section 3.5. The reactor building also provides civil structural support for the DHRS and protection of safety-related components from external hazards such as wind, tornadoes, floods, and wind-induced missile events. The DHRS design requirements for seismic and other natural hazards demonstrate conformance with the requirements in PDC 2.

The DHRS is designed and located to minimize the probability and effect of fires and explosions by the use of low combustible materials and physical separation. These design features, in conjunction with the fire protection plan described in Section 9.4, provide assurance that the DHRS demonstrate conformance with the requirements in PDC 3.

The DHRS is designed with materials that will withstand the radiation environment of the reactor cavity and environmental temperatures up to 750°C to ensure the DHRS is capable of performing its safety function under conditions associated with normal operation, maintenance, testing, and postulated events. The DHRS is designed against equipment failures that could result from Flibe spills. Pipe whip and other similar dynamic failures are avoided by the low-pressure design of the DHRS and the use of restraints. Each component of the DHRS is designed such that failure of one component does not cascade and cause failures of nearby safety systems, including other DHRS components. These design considerations demonstrate conformance with the requirements in PDC 4.

Natural circulation in the reactor core transfers decay heat from the fuel to the reactor vessel shell when normal cooling is not available, as described in Section 4.6.3. Thermal-hydraulic calculations demonstrate that the DHRS is capable of passively removing a sufficient amount of decay heat from the reactor vessel without reliance on electric power for up to 7 days as needed to mitigate postulated events, such that the reactor vessel temperature remains below its design limit and is decreasing. In addition, fuel temperatures remain below their design limits. The DHRS is designed with sufficient redundancy, leak detection capability, and isolation to ensure the safety function can be performed assuming a single failure. The system includes four independent loops and maintains the ability to perform its function with the loss of a single loop. Isolation of the four water storage tanks from one another ensures that damage at one tank location does not result in a total loss of DHRS inventory. The thimbles, separators, and thimble feedwater and steam-return piping are all contained within the leak barrier. The leak barrier provides leak detection capability and ensures that a failure of the primary DHRS pressure boundary does not prevent the system from performing its heat removal function. These DHRS design features, along with the natural circulation characteristics of the reactor core, demonstrate conformance with the requirements in PDC 34 and PDC 35.

The DHRS design includes the capability for online monitoring of leaks to monitor for system integrity and to ensure that DHRS inventory remains sufficient to perform the safety-related heat removal function. The water level in the storage tanks is also capable of being monitored to ensure that sufficient inventory is present at the onset of a postulated event to provide sufficient cooling capacity. The DHRS is also sufficiently accessible to perform inspections for system integrity. These features satisfy PDC 36.

When the reactor is above threshold power, the DHRS is an “always on” operating condition which provides an ongoing demonstration of system availability. The transition from normal to postulated event operation can also be functionally tested. These features demonstrate conformance with the requirements in PDC 37.

6.3.4 Testing and Inspection

The details of the inspection and testing program for DHRS will be described in the application for an Operating License.

Water storage tank inventory is monitored to ensure the DHRS operability. The DHRS continuous operation is also monitored to ensure DHRS availability when demanded. DHRS operability is controlled by a technical specification, as described in Chapter 14.

6.3.5 References

1. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Sec. III Div. 5, "BPVC Section III-Rules for Construction of Nuclear Facility Components-Division 5-High Temperature Reactors," 2017.
2. Not Used.
3. American Society of Civil Engineers, ASCE/SEI 43-19, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities," 2020.
4. American Society of Civil Engineers, ASCE/SEI 4-16, "Seismic Analysis of Safety-Related Nuclear Structures," 2017.
5. American Concrete Institute, ACI 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," 2014.

Table 6.3-1: Water Storage Tank Parameters

Parameter	Value
Material	Stainless Steel
Design pressure (psig)	30
Design temperature (°F)	274
Minimum volume per tank (gal)	2900
Number per reactor	4

Table 6.3-2: Steam Separator Parameters

Parameter	Value
Material	Stainless Steel
Design pressure (psig)	30
Design temperature (°F)	274
Number per storage tank	1
Nominal thimble feedwater rate (lb/s)	0.040

Table 6.3-3: Thimble Parameters

Parameter	Value
Material	Stainless Steel
Design pressure (psig)	30
Design temperature (°C)	750
Number per steam separator	6
Length (in)	144
Thimble wall outer diameter (in)	2.875

Table 6.3-4: Applicable Design Codes and Standards for the DHRS

Code	Title	Applicability
ASME Sec. III Div. 5 Class B (Reference 1)	ASME Boiler and Pressure Vessel Code – High Temperature Reactors	The DHRS metallic pressure boundary and supports will be designed and fabricated using the technical guidance in ASME, Section III, Division 5, as shown in Table 3.6-2.
ASCE 43-19 (Reference 3)	Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities	Provides design criteria for seismic analysis of reactor components (including DHRS).
ASCE 4-16 (Reference 4)	Seismic Analysis of Safety-Related Nuclear Structures	Provides additional design criteria for safety-related systems (including DHRS) that expand upon ASCE 43-19.
ACI 349-13 (Reference 5)	Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary	Applicable to cavity support structures for DHRS panels and potentially the condenser pool construction.

Figure 6.3-1: Functional Diagram of the DHRS

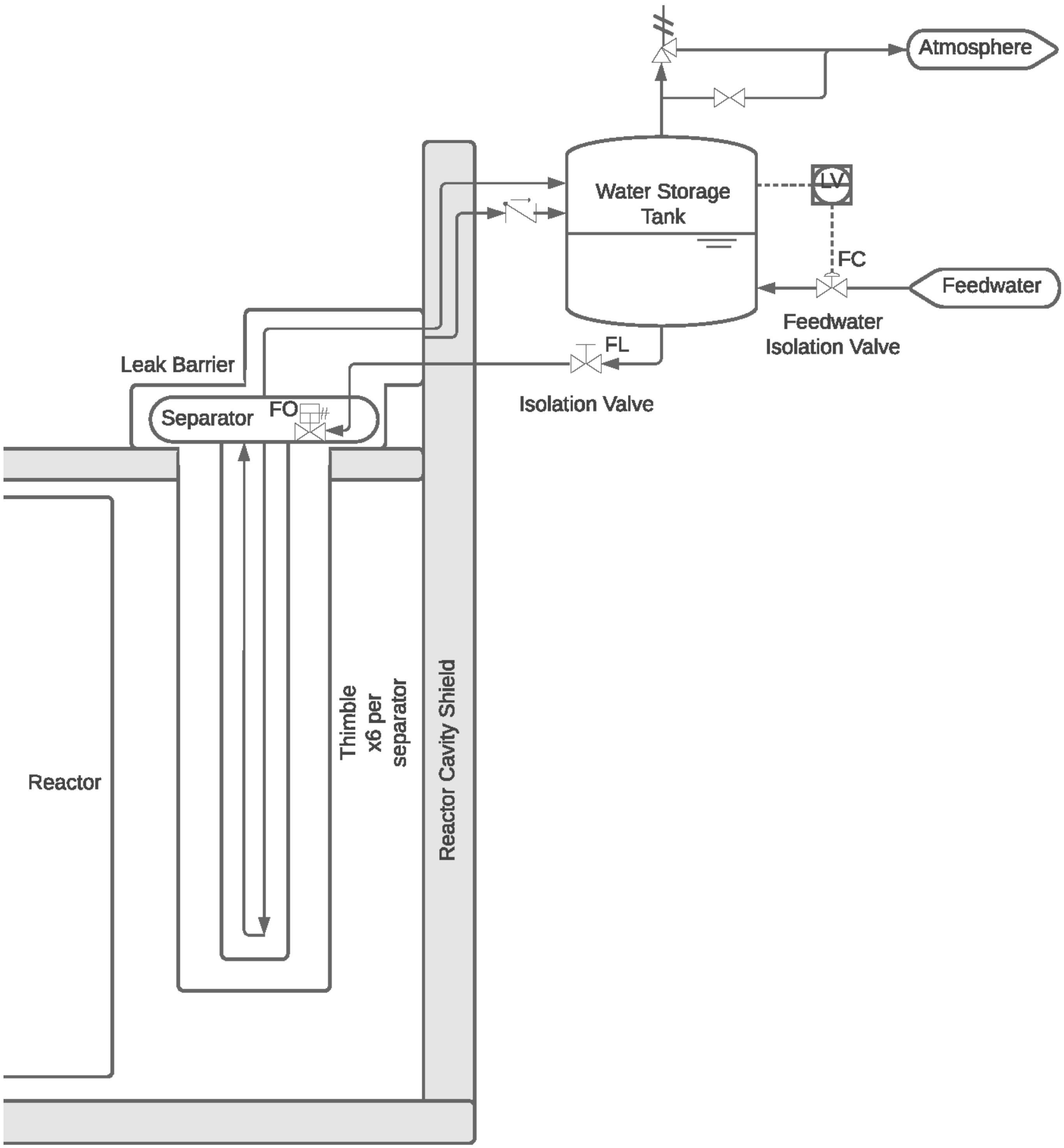


Figure 6.3-2: Notional Diagram of the DHRS Separator and Float Valve

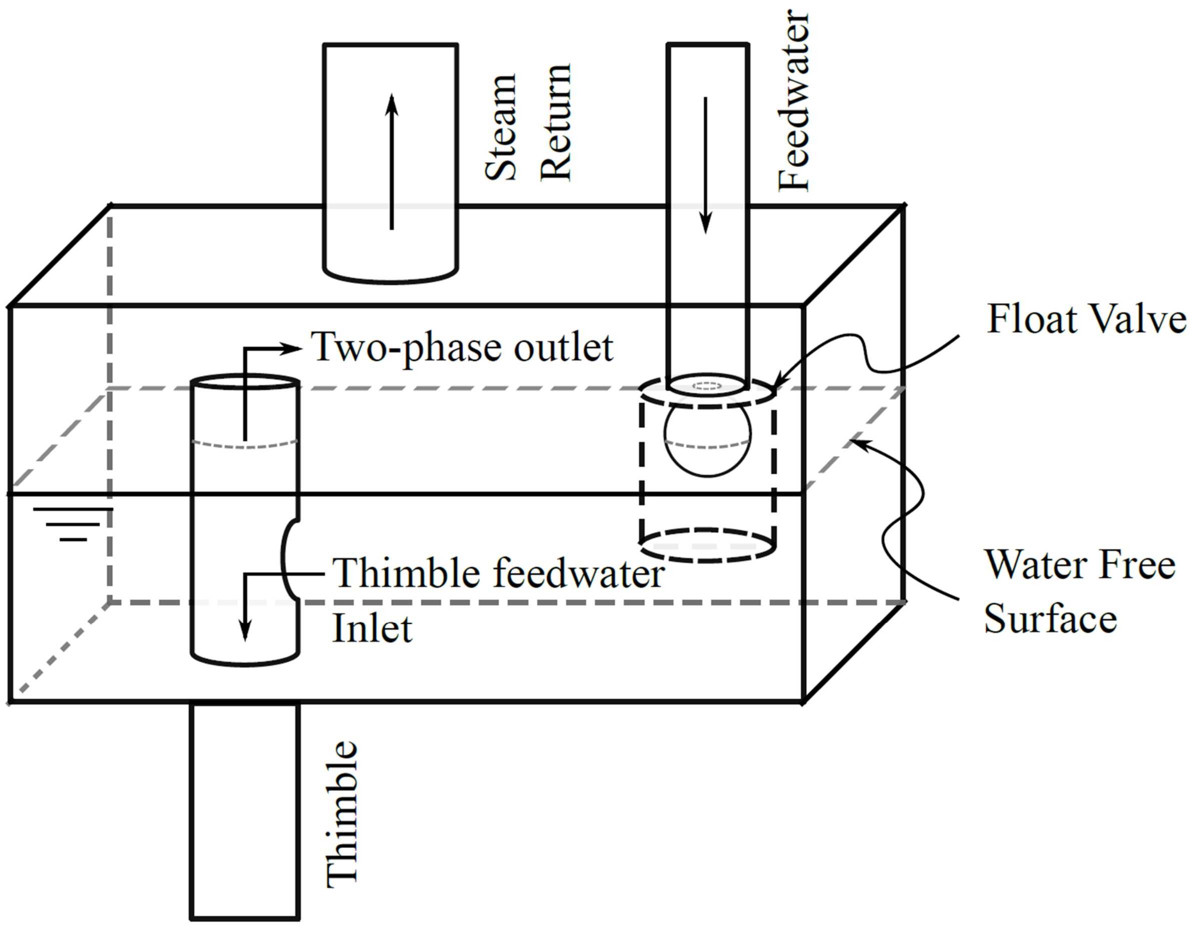
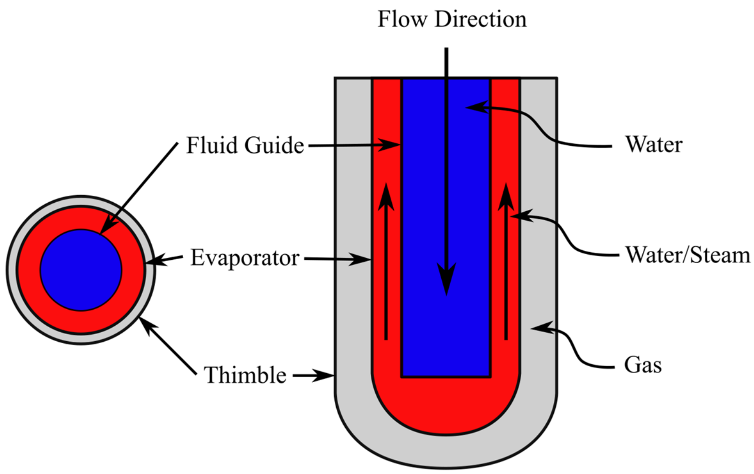


Figure 6.3-3: Annular Thimble Geometry



Chapter 7

Instrumentation and Control Systems

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 7 INSTRUMENTATION AND CONTROLS

7.1 INSTRUMENTATION AND CONTROLS OVERVIEW

7.1.1 Summary Description

The instrumentation and control (I&C) systems monitor and control plant operations during normal operations and planned transients. The systems also monitor and actuate protection systems in the event of unplanned transients. The I&C systems described in this chapter are applicable to both Unit 1 and Unit 2. The safety-related portions of the I&C systems are unit specific. Portions of the non-safety related I&C systems that control shared systems are shared between the units. I&C is composed of four parts, described in the bulleted list below. Each of the four parts are described in further detail in subsequent subsections of this chapter. The architectural design of the system accounts for interconnection interfaces for plant I&C structures, systems, and components (SSCs). Figure 7.1-1 provides an overview of the I&C system architecture.

- The plant control system (PCS) provides the capability to reliably control the plant systems during normal, steady state, and planned transient power operations, including normal plant startup, power maneuvering, and shutdown (see Section 7.2). The portions of the PCS that control the shared systems are shared between Unit 1 and Unit 2.
- The reactor protection system (RPS) provides protection for reactor operations by initiating signals to mitigate the consequences of postulated events and to ensure safe shutdown (see Section 7.3). The safety-related RPS is not shared between Unit 1 and Unit 2.
- The main control room and remote onsite shutdown panels provide the capability for plant operators to monitor plant systems, control plant systems, and to initiate plant shutdown (see Section 7.4). Unit 1 and Unit 2 share a common main control room. Each unit is provided with a unit-specific remote onsite shutdown panel.
- Sensors provide input to multiple control and protection systems (see Section 7.5). Safety-related sensors are not shared between Unit 1 and Unit 2. Non-safety related sensors that control and monitor shared systems are shared between Unit 1 and Unit 2.

The I&C system implements IEEE Standard 603-2018 (Reference 1) and IEEE Standard 7-4.3.2-2003 (Reference 2) and other consensus standards for safety-related I&C functions. The particular application of consensus standards is discussed for each I&C subsystem in the following sections.

The I&C system incorporates the principles of independence, redundancy, and diversity. Features reflecting those principles are discussed in the specific subsystem descriptions. The RPS is the safety-related system credited for tripping the reactor and actuating engineered safety features. Accordingly, the RPS is isolated and independent from the other I&C systems and uses input signals from independent instrumentation. RPS instrumentation signals are provided to the PCS via a data diode, which is part of the RPS hardware platform (See Section 7.3.3). The RPS incorporates redundancy and diversity in the system design as discussed in Section 7.3. The I&C system includes the capability for both manual and automatic control.

Section 7.5 describes the sensors used at the facility. Sensors for temperature, pressure, neutron count rates, level, flow, radiation level, and other analog and digital field detectors provide input to the plant control system and reactor protection system. Independent instruments are provided for RPS and PCS. Each section about specific I&C subsystems includes a discussion of the instruments that support that subsystem and the type of instrumentation used (i.e., analog or digital).

7.1.2 Calibration of Trips, Interlocks, and Annunciators

Safety limits (or analytical limits (ALs)) are defined by the operating limits in the plant safety analysis.

Systems having significant safety functions (for example technical specification limiting conditions for operation) that do not directly protect a plant safety limit, will be analyzed in the same fashion as those having safety limits. The technical specifications are described in Chapter 14.

Setpoints for safety-related instrumentation will be calculated in accordance with the guidance of ANSI/ISA 67.04.01-2018 (Reference 3). The setpoint nomenclature as defined in the Regulatory Information Summary RIS-2006-17 (Reference 4), will be applied to setpoint calculations developed to support licensing activities. Operational considerations such as drift, linearity, hysteresis, and operational margins are considered in the development of specific instrument loop setpoints. Consideration is also given to fixed instrument errors and environmental affects in the selection of instrument setpoints.

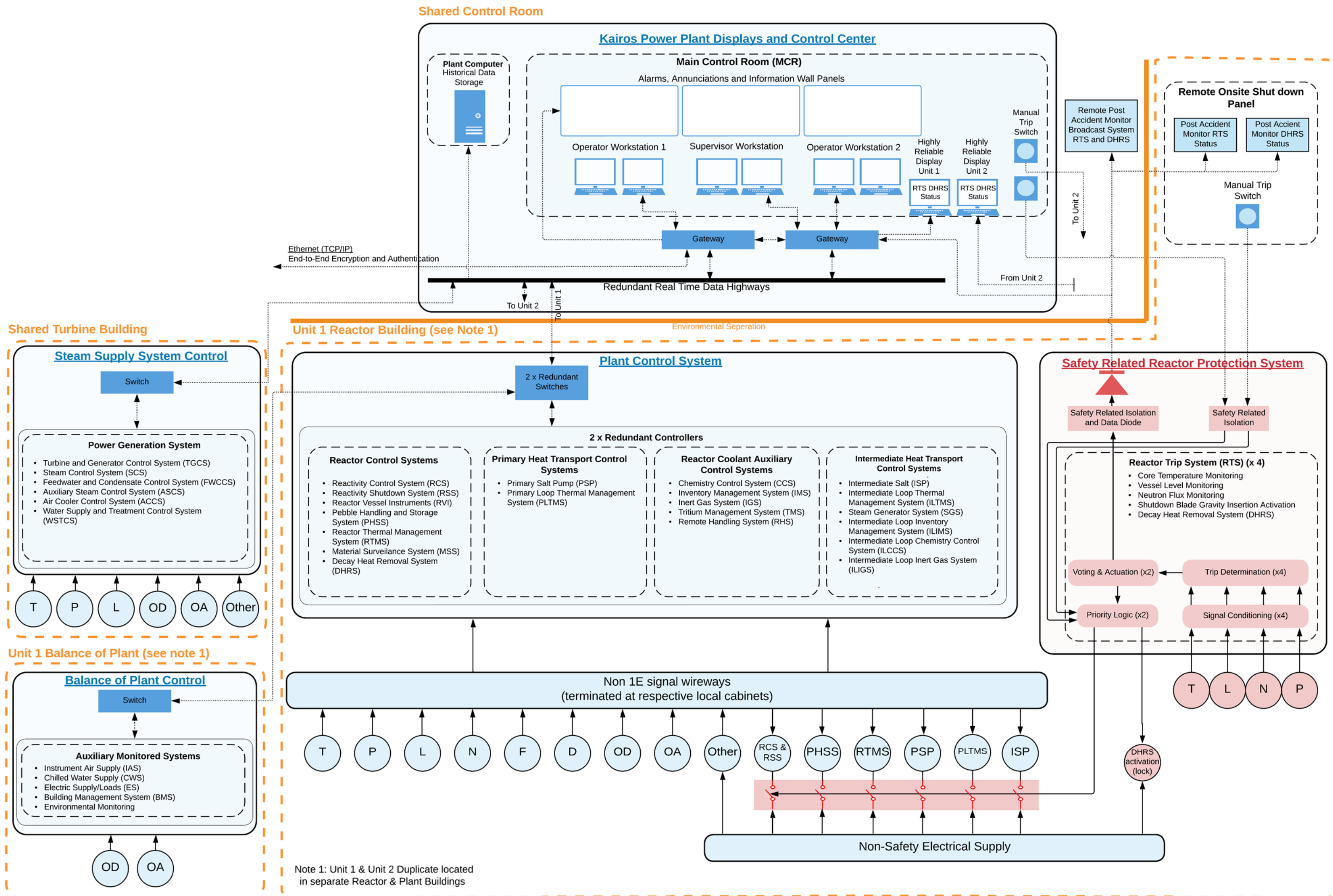
The RPS include sensors, trips, and interlocks to shut down the reactor when operating parameters exceed operational limits. This includes release of the control and shutdown elements within a set of defined parameters after the onset of a postulated event. Specific trips and interlocks are discussed in Section 7.3. However, RPS actuation setpoints for trips and interlocks are calculated based on the following design principles:

- Simulation models: Time to reach operational limits based on system qualification (environments, process conditions, etc.) as demonstrated by actual empirical data collected during simulation testing
- RPS Technical Specifications: Measurement time, process parameters as informed by safety case assumptions and bounded by Technical Specification limits
- Mechanical design and testing - response time for actuation to complete: Time to detect, process, and actuate the required controls; this time should be less than the time between event onset and parameter reaching a limiting condition for continued operation
- Tiered (graded) approach to protection: The RPS utilizes highly reliable safety-related parameters as the final level of protection for public health and safety.

7.1.3 References

1. Institute of Electrical and Electronics Engineers, Standard IEEE 603, “Standard Criteria for Safety Systems for Nuclear Power Generating Stations.” 2018.
2. Institute of Electrical and Electronics Engineers, IEEE Standard 7-4.3.2, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations." 2003.
3. Instrument Society of America, ANSI/ISA-67.04.01, “Setpoints for Nuclear Safety-Related Instrumentation.” 2018.
4. Nuclear Regulatory Commission, Regulatory Issue Summary 2006-17, “NRC Staff Position on The Requirements of 10 CFR 50.36, ‘Technical Specifications,’ Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels.” August 24, 2006.

Figure 7.1-1: Instrumentation and Controls System Architecture



Legend for Figure 7.1-1

T	Temperature
P	Pressure
L	Level
F	Flow
N	Neutronics
R	Radiation Monitor
D	Discrete (Digital Input of Output/Actuation)
OA	Other analog field instruments
OD	Other digital field instruments

7.2 PLANT CONTROL SYSTEM

7.2.1 Description

The PCS is a non-safety related control system which controls reactor startup, changes in power levels, reactor shutdown, heat transport, and power generation system. The PCS implements these functions through a series of subsystems which include:

- Reactor control system (RCS)
- Reactor coolant auxiliary control system (RCACS)
- Primary heat transport control system (PHTCS)
- Intermediate heat transport control system (IHTCS)
- Power generation control system

The PCS is a microprocessor-based distributed control system that individually controls plant systems using applicable inputs. The subsystems listed above are integrated into the PCS using non-safety related signal wireways which are terminated at local cabinets and using redundant, non-safety, real time data highways.

The control subsystems communicate with each other to ensure each subsystem has access to parameters such as measurements and actuation events from other subsystems.

This allows the PCS to maintain plant and unit parameters within the normal operating envelope. The RCS, RCACS, PHTCS, and IHTCS are unit-specific subsystems. The power generation control system is shared between Unit 1 and Unit 2. The PCS also provides data to the control consoles located in the main control room (see Section 7.4). Figure 7.1-1 shows the elements of the PCS.

The plantwide sensor inputs are used to verify interlock and permissive rules for the various plant states. The sensor data is also used to provide feedback and alarms to the operators via the control consoles. The PCS is powered by AC and DC power supplies which are discussed in Chapter 8.

The PCS uses non-safety related sensor inputs as well as safety-related sensor inputs from the plant protection system (See Section 7.3.3). The PCS includes the input parameters shown in Table 7.2-1. The sensors are described in Section 7.5. The instrumentation provides input signals using non-safety related signal wireways that are terminated at local cabinets.

Control outputs are generated using a control transfer function based on the sensor inputs and setpoints provided by the control system. The setpoints are adjusted automatically based on the plant operating mode, or in some cases by the operator via the main control room consoles. Plant operators do not directly control PCS outputs.

The PCS does not provide any safety-related functions during any mode of operation or postulated event. The PCS is electrically and functionally isolated from the safety-related RPS (see Section 7.3) using a safety-related isolation device as shown in Figure 7.1-1. The RPS isolation devices ensure electrical isolation between the electrical system and the non-safety related SSCs that PCS normally controls that are deactivated by the RPS when a reactor trip is demanded.

The subsystems of the PCS are described below.

7.2.1.1 Reactor Control System

The RCS controls and monitors systems and components that support normal operation, planned transients, and normal shutdown of the reactor. The RCS controls the systems listed in Figure 7.1-1 and supports the following capabilities:

- Reactivity control and planned transients/adjustments in power level
- Monitoring of core neutronics
- Pebble handling and storage
- Monitoring and control of temperature in the reactor

The RCS controls reactivity for normal operations and normal shutdown using reactor control elements and reactor shutdown elements in the reactivity control and shutdown system (RCSS) (see Section 4.2). The RCS is capable of incrementally changing the position of reactor control elements and of releasing the control and shutdown elements. The RCS is only capable of withdrawing elements one at a time and the RCS includes a limit on the rate at which a control element can be withdrawn, as also discussed in Section 4.2.2. In this way the design precludes, with margin, the potential for prompt criticality and rapid reactivity insertions. The RCS inputs include reactor outlet temperature and reactor inlet temperature sensors and source and power range neutron excore detectors. The RCS also provides a reactor monitoring function to monitor plant components that are associated with reactor functions. The RCS uses source and power range sensors that are located outside the reactor vessel for reactor control.

The RCS controls pebble insertion and extraction, in-vessel pebble handling, and ex-vessel pebble handling in the pebble handling and storage system (PHSS) (see Section 9.3). The RCS is capable of counting linearized pebbles external to the vessel, controlling the rate of pebble insertion and removal from the vessel, and controlling pebble distribution within the PHSS.

The RCS controls the reactor thermal management system (RTMS) (see Section 9.1.5) to monitor the temperature of the primary system to maintain it within the normal operating envelope and to implement planned transients. The RCS controls external heating elements in the RTMS to prevent overcooling.

7.2.1.2 Reactor Coolant Auxiliary Control System

The RCACS controls and monitors systems and components that support normal operation in the core. The system supports the following capabilities in the core:

- Chemistry control in the primary system
- Inventory management system control
- Inert gas system control in the primary loops
- Tritium management system monitoring and control

The RCACS controls the chemistry control system (see Section 9.1.1) to monitor reactor coolant chemistry. The monitoring systems provide information to facilitate maintaining coolant purity and circulating activity within specifications for the system.

The RCACS receives input from the inventory management system (see Section 9.1.4) which monitors primary coolant level during normal operations. The system also provides control for changes to primary inventory during planned primary filling and draining operations.

The RCACS also controls the inert gas system (see Section 9.1.2). During normal operation, the system provides control signal to maintain cover gas pressure and flow, monitors venting gas for impurities above specified limits in the gas space of the primary system. During startup, the system monitors and controls inert gas flow and temperature to support initial heating of the primary system.

The RCACS receives input from the tritium management system (see Section 9.1.3) and provides control signal to remove tritium from the cover gas in the primary system.

7.2.1.3 Primary Heat Transport Control System

The PHTCS controls and monitors systems and components that support normal operation of the primary heat transport system (PHTS). The system supports the following capabilities:

- Control of the flow rate through the PHTS
- PHTS thermal management
- Primary loop draining, filling, and piping monitoring, including PHTS external piping

The purpose of the PHTCS is to control the transport of primary coolant through the PHTS, to maintain the primary coolant in a liquid state, and to monitor the inventory of primary coolant in the PHTS. The PHTCS maintains the parameters in the PHTS within the normal operating envelope. The PHTCS controls the primary salt pump (PSP)and the primary loop thermal management subsystem (PLTMS). The sensors used by the PHTCS are discussed in Section 7.5.

The PHTCS provides control signal for the PSP (see Chapter 5). The control system manipulates the primary coolant flow rate by variable frequency to maintain PHTS parameters within the normal operating range. The PHTCS does not provide a safety function; however, as discussed in Section 7.3, the RPS trips the PSP on a reactor trip, as a protection feature for the reactor system related to the pump.

The PHTCS maintains the primary coolant in liquid phase throughout the PHTS to prevent localized over- or under-heating. The control system uses temperature as input to provide control signal to the PHTS auxiliary heaters.

7.2.1.4 Intermediate Heat Transfer Control System

The IHTCS controls and monitors systems and components that support normal operation of the intermediate loop which removes heat from the primary loop. The system supports the following capabilities:

- Control of the flow rate through the intermediate loop
- Intermediate loop heating
- Intermediate loop draining, filling, and piping monitoring
- Chemistry control in the intermediate loop
- Maintain positive pressure differential between the PHTS and IHTS during normal operations

The purpose of the IHTCS is to control the transport of intermediate coolant through the intermediate loop, to maintain the intermediate coolant in a liquid state, and to monitor the inventory of intermediate coolant in the intermediate loop. The monitoring systems provide information to facilitate maintaining intermediate coolant purity within specifications for the system. The IHTCS does not perform a safety function. The IHTCS maintains the parameters in the intermediate loop within the normal operating envelope. The IHTCS controls the intermediate salt pump (ISP), the intermediate loop auxiliary heating system, the intermediate coolant inventory system, the intermediate coolant chemistry control system, and the intermediate inert gas system. The IHTCS controls the ISP by changing the intermediate coolant flow rate by variable frequency to maintain intermediate loop parameters within the normal operating range. The IHTCS controls the intermediate loop auxiliary heating system to maintain the intermediate coolant in liquid phase throughout the intermediate loop to prevent localized over- or under-heating. The control system uses temperature information as input to provide control signal to the intermediate loop auxiliary heaters. The IHTCS monitors the intermediate coolant inventory system, the intermediate coolant chemistry control system, and the intermediate inert gas system and provides information to facilitate maintaining the intermediate coolant within specifications for the system. The IHTCS monitors the IHX intermediate coolant inlet pressure and reactor coolant outlet pressure and controls the speed of the ISP to maintain a positive pressure differential.

7.2.1.5 Power Generation Control System

The power generation control system controls and monitors systems and components that support normal operation of the turbine-generator which converts heat from the intermediate loop into electrical power. The system supports the following capabilities:

- Monitoring of turbine-generator system parameters and initiation of turbine-generator trips and pump runbacks
- Control of steam flow rate from each unit's superheater to the turbine, and auxiliary steam loads
- Control of condensate and feedwater flow rate and temperature to the evaporator
- Control of the position of turbine control and bypass valves
- Control of the removal of heat from the air-cooled condenser

The purpose of the power generation control system is to control the conversion of thermal energy into mechanical energy. The power generation control system does not perform a safety-related function. The power generation control system maintains the parameters within the turbine generator, main steam, condensate, and feedwater systems within the normal operating envelope.

7.2.2 Design Bases

Consistent with Principal Design Criteria (PDC) 13, the PCS is designed to monitor variables and systems over their anticipated ranges for normal operation, and over the range defined in postulated events.

7.2.3 System Evaluation

The PCS is designed to monitor plant and unit parameters and maintain systems within normal operating range. The PCS is also designed to control planned transients associated with anticipated operational occurrences and maintain the affected reactor in a shutdown state. These functions are consistent with PDC 13. The PCS does not perform a safety-related function. Finally, the PCS is designed so that it cannot interfere with the RPS's ability to perform its safety functions; see Section 7.3 for more information about the isolation of the RPS from the PCS.

The PCS is a digital system that controls the reactor power about a point set by the operator. The control system uses linear average temperature and flow rate in the primary system as variable inputs to control power level so that it remains within the normal operating envelope. The PCS controls electrical power generation about a point set by the operators using steam flow rates, feedwater flow rates, and feedwater temperatures as inputs to control the positions of turbine control valves, turbine bypass valves, and feedwater regulating valves to balance the turbine load from each unit. The system design meets the applicable portions International Electrotechnical Commission (IEC) standard 61131 for industrial controllers (Reference 1), and the applicable portions of the cyber security standard IEC 62443 (Reference 2). Table 7.2-2 lists other standards applied to the PCS. Applicable portions of IEEE 1012-2017 (Reference 3) are used for verification and validation of PCS components, which is consistent with the non-safety related classification of the PCS.

Action in the PCS is designed to accurately and reliably provide control signal for all modes of normal operation. The PCS is also designed to provide timely control signals, with further analysis of timeliness to be provided in an application for the Operating License.

The PCS includes interlocks and inhibits that prohibit or restrict operation of the reactor, PHSS, and the power generation system unless certain operating conditions are met. The following interlocks are included in the control system design:

- An interlock that prohibits reactivity control element withdrawal until there is sufficient neutron count rate to ensure that nuclear instruments are responding to neutrons.
- Interlocks are also provided related to startup power level and pebble handling as detailed in Table 7.2-3.
- An interlock that prevents the opening of a unit's main steam isolation valve following a reactor trip until there is sufficient steam production to ensure that a turbine imbalance will not occur.

The PCS initiates automatic turbine generator trip signals if certain conditions are detected. In the event of a turbine generator trip, the PCS initiates runbacks of the RCSS, PSP, ISP, and feedwater pumps on both units to decrease reactor thermal power and heat transport to a level that can be safely rejected using normal shutdown cooling if the condenser is available or using main steam power relief valves and/or main steam safety valves if the condenser is not available.

In the event of a single unit reactor trip, the PCS will initiate signals to close the main steam isolation valve, open turbine bypass valves, regulate flow control valves through the unit specific superheater and runback feedwater flow to the affected unit to maintain a minimum flow to the steam generator, ensure balanced steam supply to the turbine, and prevent overcooling of the intermediate loop, as discussed in Section 9.9. A turbine generator runback will also be initiated to establish turbine generator output within the capacity of a single unit's superheater to allow the unaffected unit to remain online. Should the grid be unable to absorb the communicated power loss of a single unit trip, the turbine generator will lose grid synchronization and trip, in which case steam from the remaining unit will bypass the turbine while the reactor ramps down in power or grid connection is re-established.

The unit-specific plant controls are grouped and located on unit-specific operating panels in the main control room so that operators for each unit can easily reach and manipulate the controls. The shared system controls are grouped and located on a separate operating panel within the main control room that is accessible to both unit's operators and requires coordination between each unit's operators before controls can be manipulated. Displays of the results of operator actions are readily observable. See Section 7.4 for more information about the human interface for the PCS.

The PCS is not safety-related and no safety-related SSCs cross the seismic isolation moat, discussed in Section 3.5. However, any portion of the PCS that crosses the moat includes flexible design features to accommodate design displacements from postulated seismic events to the extent necessary to prevent damage of SSCs in the PCS from affecting a safety-related SSC's ability to perform its safety function. Specific design features and the SSCs to which they are applied, will be provided in the Operating License application.

Additional information about the PCS that is dependent on the final design of the reactor and the power generation system's SSCs will be provided in the Operating License Application, including: (1) further specifics about the hardware and software, (2) software flow diagrams for digital computer systems, (3) a description of how the operational and support requirements will be met, and (4) the basis for reliability of PCS systems and reliability targets.

7.2.4 Testing and Inspection

Functional tests will be performed prior to initial startup and tests and inspections consistent with the standards discussed in Section 7.2.3.

7.2.5 References

1. International Electrotechnical Commission, IEC 61131, "Programmable Controllers.” 2020.
2. International Electrotechnical Commission, IEC 62443, “Cybersecurity.” 2015.
3. Institute of Electrical and Electronics Engineers, IEEE 1012-2017, “System, Software, and Hardware Verification and Validation.” 2017.

Table 7.2-1: Plant Control Variables

Control Variables (Inputs)	Primary Loop • PSP speed • Control rod drive position • Inert gas pressure Intermediate Loop • ISP speed • Loop temperature Power Generation System • Steam flow • Feedwater flow	
	Air Cooling • Blower speed	
Controlled Variables (Outputs)	Primary Loop • Neutron flux (self-powered neutron detectors and ion cambers) • Reactor outlet temperature • Primary coolant mass flow rate Intermediate Loop • Intermediate coolant mass flow rate Power Generation System • Electrical power output • Valve position • Condenser vacuum	
Constrained Variables (Outputs)	Primary Loop • Excess reactivity margin • Reactor inlet temperature	

Table 7.2-2: Standards Applicable to the Plant Control System

Identifier	Standard
1	Institute of Electrical and Electronics Engineers, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations," IEEE Std 7-4.3.2-2003, Annex C, Sections C.2.2.2, C.2.2.3 and C.2.3, Piscataway, N.J.
2	IEC 61131: 2020 SER, “Programmable Controllers”
3	IEC 62443: 2015, “Cybersecurity”

* The software development process will follow Annex C, Sections C.2.2.2, C.2.2.3 and C.2.3

Table 7.2-3: Plant Control System Interlocks and Inhibits

Input Signal to the Plant Control System	Interlock or Inhibit
High radiation detected in pebble handling area	Movement of pebbles stops within a specified time delay *Purpose*: Minimize effects of a PHSS transfer line break
Abnormal positioning of pebble in PHSS	Movement of pebbles stops within a specified time delay *Purpose*: Prevent damage to PHSS system
Neutron Flux detected on Source Range and is below 0.5 count/second	Block reactivity control element withdrawal *Purpose:* Prevent inadvertent rapid positive reactivity insertion
DHRS operating	RTMS blocked from operating *Purpose*: Prevent inadvertent actuation of RTMS.

7.3 REACTOR PROTECTION SYSTEM

7.3.1 Description

The RPS provides protection for reactor operations by initiating signals to mitigate the consequences of postulated events and to ensure safe shutdown. The RPS is the only portion of the I&C system that is safety-related and that is credited for tripping the reactor and actuating engineered safety features. RPS SSCs are unit-specific and are not shared between Unit 1 and Unit 2. The purpose of the RPS is to actuate upon receipt of a trip signal in response to out-of-normal conditions and provide automatic initiating signals to protection functions. There are three possible trip sources that can cause the RPS to actuate and three protection functions that result from RPS actuation, shown below in Figure 7.3-1. The three possible trip sources are:

- Process variables reach or exceed specified setpoints, as measured by RPS sensors
- Manual initiation from the main control room or remote onsite shutdown panel
- Plant electric power is lost (with a time delay)

The three KP-FHR protection functions that result from RPS actuation are:

- Actuate the RCSS that inserts control and shutdown elements into the reactor core
- Inhibit actions from the PCS so that it does not interfere with the functioning of the RPS
- Ensure an actuation of the decay heat removal system (DHRS) that passively removes heat from the PHTS to the atmosphere

Actuation of the RPS to trip the reactor includes several actuations that stop specific non-safety related SSCs, normally controlled by PCS, to ensure that those non-safety related SSCs to do not prevent a safety-related SSC from performing its safety function. The non-safety related functions that are stopped are shown in Figure 7.3-1. RCSS element withdrawal is inhibited after a loss of power, to prevent inadvertent positive reactivity insertion when power returns (see also Table 7.3-2). The PSP is stopped to maintain Flibe inventory in the core. The ISP is stopped to limit inadvertent overcooling of the PHTS. Pebble extraction and insertion in the PHSS is stopped to prevent removing pebbles from the core in the event of a PHSS extraction line break. Finally, RTMS and PLTMS actuations are prohibited to prevent a challenge to the heat removal capability of the DHRS. These inhibitions are accomplished through safety-related trip devices as shown in Figure 7.1-1.

The RPS is built on a logic-based platform that does not utilize software or microprocessors for operation. It is composed of logic implementation using discrete components and field programmable gate array (FPGA) technology. The RPS is isolated from other I&C systems, including the main control room and the remote onsite shutdown panel, using safety-related isolation hardware. Isolation is achieved at the point of signal generation either through features built into the hardware platform or through separate isolation devices. The RPS includes the following safety-related (except as noted otherwise) elements:

- Separate channels of sensor electronics and input devices
- Redundant and separate groups of signal conditioning
- Redundant and separate groups of trip determination
- Manual reactor trip switches in the main control room (switches are non-safety related)
- Safety-related components to provide electrical isolation from the non-safety-related highly reliable DC power system power supply
- Multiple reactor trip devices and associated cabling (cabling is non-safety related)
- RPS isolation hardware
- Two divisions of reactor trip system (RTS) voting and actuation equipment

Reactor trip functions are hardcoded into FPGA logic and are not dependent on plant operating state. Operating conditions are compared against the trip setpoints and actuate protection functions according to established programmable logic. The RPS cabinets are located within the safety-related portion of the Reactor Building within an environmentally separated enclosure, discussed further in Section 7.3.3.

The RPS performs safety-related functions as shown in Figure 7.1-1, which include RTS actuation and ensuring actuation of the DHRS. Both functions are described in more detail in Sections 7.3.1.1 and 7.3.1.2. Operator interface for the RPS is discussed in Section 7.4. The RPS uses inputs from the reactor core temperature, reactor vessel level, and source and power range neutron excore detectors. The sensors that provide input to the RPS are safety-related and described further in Section 7.5. The four source range and four power range excore detectors monitor neutron flux. The power range excore detectors are located in azimuthally symmetric locations outside the reactor vessel at mid-core elevation. The source range excore detectors are located in optimal locations for best detectability of criticality. The power range and source range excore detectors are used to monitor core power during normal operation and are used as input to the rate trip. The source range detectors are used during reactor startup. Final design for the neutron flux monitoring will be provided with the application for the Operating License.

7.3.1.1 Reactor Trip System

The RTS actuates the RCSS that allow for insertion of control and shutdown elements into the reactor core. Upon receipt of a trip signal, the RTS removes power from coils on the reactivity shutdown elements, which drop by gravity into the reactor (See Section 4.2.2 for more information about the shutdown elements). The RTS receives trip signals generated from automatic or manual sources.

The RTS is built on a logic-based platform that does not utilize software or microprocessors for operation. It is composed of logic implementation using discrete components and FPGA technology. The RTS is isolated from other I&C systems using safety-related isolation hardware.

The RTS receives input from sensors through hardwired, analog, safety-related signal wireways that are terminated at local cabinets. Section 7.5 provides additional information about the sensors that provide input to the RTS. Using the inputs from the sensors, the RTS automatically opens the reactor trip devices when setpoints are reached. The system uses both undervoltage coils as well as shunt trip coils to provide the means to open the trip devices. The reactivity shutdown element position coils fail open on loss of power.

The main control room and the remote onsite shutdown panel each have the capability to provide a manual trip signal to the RTS. Section 7.4 includes a discussion of the human interface with the RTS.

Table 7.3-2 provides a list of interlocks implemented for RPS systems. If normal power is not available and the RPS does not detect a transfer to backup power within a defined time period, the RPS removes power from the RTS, causing the control and shutdown elements to drop into the core. The RPS includes an interlock that inhibits movement of reactivity control elements, and a manual reset is required before reactivity control elements can be withdrawn. The purpose of this interlock is to prevent inadvertent insertion of positive reactivity when normal power is lost and subsequently restored.

On actuation, the RTS will trip the PSP. A manual reset prevents the pump from inadvertently restarting after power return. To limit overcooling, the ISP trips concurrently with the PSP. An interlock prevents starting the ISP if the PSP is not running.

7.3.1.2 Decay Heat Removal System

The DHRS provides passive residual heat removal that requires no electrical power to operate, as discussed in Section 6.3. Although the DHRS is always operating above a certain threshold of fission product accumulation level, the decay heat removal portion of the RPS provides actuation signal to DHRS to ensure the DHRS is operating when there is a RPS actuation signal. The RPS actuation signal to DHRS is achieved by removing power to the water tank isolation valve that allows passive coolant flow. The water tank isolation valves fail in place upon loss of power. The decay heat removal portion of the RPS can receive the actuation signal from either an automatic or manual source.

The decay heat removal portion of the RPS uses core temperature and neutron detectors as inputs through hardwired, analog, safety-related signal wireways that are terminated at local cabinets. Section 7.5 provides additional information about the sensors that provide input to the RPS.

The decay heat removal portion of the RPS also includes a manual actuation capability from the main control room and the remote onsite shutdown panel. Section 7.4 includes a discussion of the human interface with the decay heat removal portion of the RPS.

Table 7.3-2 provides a list of interlocks implemented for RPS systems. Before sufficient fission products and subsequent decay heat is produced in the core, for example during startup, DHRS has no safety function. During this period, the decay heat removal portion of the RPS includes a manual inhibition of the DHRS that is available to plant operators to allow for additional thermal management capabilities. Once decay heat is produced at a sufficient rate in the core, the RPS blocks the manual inhibition capability utilizing safety-related actuations. After shutdown, once fission product decay heat production has dropped to levels not requiring DHRS, the RPS removes the block on the manual inhibition capability. The parameters the RPS uses to determine if the manual inhibition is to be permitted or blocked are neutron excore detectors (source and power range) and core temperature.

7.3.2 Design Bases

- Consistent with PDC 1, the RPS is designed using relevant industry codes and standards and the Quality Assurance program.
- Consistent with PDC 2, the RPS is designed to withstand and be able to perform during natural phenomena events.
- Consistent with PDC 3, the RPS is designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions.
- Consistent with PDC 4, the RPS is designed for the environmental conditions associated with normal operation, maintenance, testing, and postulated events.
- Consistent with PDC 10 and 20, the RPS provides reactor trip and decay heat removal actuation that ensure radionuclide release design limits are not exceeded during normal operation.
- The RPS implements PDC 13 in that the system includes sensors that monitor core temperature, vessel level, and power level. The sensors monitor variables and systems over their anticipated ranges for normal operation and for postulated event conditions.
- Consistent with PDC 15, the RPS provides reactor trip and decay heat removal actuation to ensure that the design conditions of the reactor coolant boundary are not exceeded during normal operation.
- Consistent with PDC 20, the RPS provides automatic reactor trip and decay heat removal actuation to ensure radionuclide release design limits are not exceeded as a result of postulated events. The RPS is also designed to identify postulated event conditions and initiate passive insertion of reactivity shutdown elements and passive decay heat removal.

- Consistent with PDC 21, the RPS is designed with sufficient redundancy and independence to assure than no single failure results in loss of its protection function. Individual components of the RPS may be removed from service for testing without loss of required minimum redundancy. The RPS is designed to permit periodic testing.
- Consistent with PDC 22, the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated event conditions, do not result in loss of the protection function for the RPS. The RPS is designed with sufficient functional and component diversity to prevent the loss of function for the RPS.
- Upon loss of electrical power or detection of adverse environmental conditions, the RPS fails to a safe state, consistent with PDC 23.
- The RPS system functionally independent from the control systems, consistent with PDC 24.
- Consistent with PDC 25, the RPS is designed to ensure that radionuclide release design limits are not exceeded upon reactor trip actuation, including in the event of a single failure of the reactivity control system.
- Consistent with PDC 28, the RPS setpoints are designed to limit the potential amount and rate of reactivity to ensure sufficient protection from postulated events involving reactivity transients. The limits are set such that reactivity events cannot result in damage to the reactor coolant boundary greater than limited local yielding, and cannot sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core.
- The RPS is designed to be redundant and diverse to assure there is a high probability of accomplishing its safety-related functions in postulated events, consistent with PDC 29.
- The RPS is designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the safety function to be performed.
- The RPS is designed in accordance with IEEE Std 603-2018 (Reference 1).

7.3.3 System Evaluation

The RPS provides automatic reactor trip (1) if plant parameters exceed the normal operation envelope (PDC 20), (2) in the event of station blackout, and (3) manually using signal from the main control room or remote onsite shutdown panel. The RPS also ensures that the DHRS is running when the reactor trips. The RPS is consistent with NUREG-1537, “Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors,” by meeting IEEE 603-2018. Table 7.3-1 provides a list of the consensus standards to which the RPS is designed.

Chapter 13 describes the postulated events to which the RPS is designed to respond. The RPS uses the same set of operating parameters in the trip and actuation logic for all modes of reactor operation. The setpoints are established to ensure that the design conditions of the reactor coolant boundary are not exceeded during operation within the design basis. This is consistent with PDC 25 because maintaining the reactor coolant boundary within design basis bounds will ensure that radionuclide release design limits are not exceeded. The setpoints are established and calibrated using the method described in Section 7.1.2.

Reactor trips implemented by the RPS meet IEEE 603-2018, Section 4. The primary plant trip signal is based on core temperature measurement. In addition, the plant will also have a trip signal for high flux rate based on input from the neutron detector sensors and a trip of the reactor upon detection of a break in the PHSS extraction line. When the temperature or flux rate are outside the normal operating range or when a PHSS extraction line break is detected, the primary plant trip deenergizes the RSS trip device, the DHRS loop trip device, and the PCS inhibitor trip device. Redundant trip devices are provided for each signal pathway. Note that the cabling to the trip devices is not classified as safety-related because the trip devices accomplish their safety function without reliance on the input cabling.

However, the cables to the trip devices are design to IEEE 603-2018. See Figure 7.3-1 for a schematic of the RPS trip logic. Trip setpoints are established and calibrated using the methods described in Section 7.1.2. The PCS inhibitor trip device functionally isolates the RPS from the PCS. This includes tripping the PSP, discussed in Section 7.2.1.3. The RPS also provides alarm signals to the main control room, which will be described in the Operating License application.

Consistent with PDCs 10, 15, and 20, the RPS provides reactor trip and decay heat removal actuation to ensure that the design conditions of the reactor coolant boundary are not exceeded during normal operation, including anticipated operational occurrences. With power, the RPS provides a trip actuation that opens a trip device, removing power from the reactor protection features (shut down elements and decay heat removal), as discussed in Sections 7.3.1.1 and 7.3.1.2. In the event that the RPS loses power, the RPS fails to a safe state, consistent with PDC 23. With loss of power, the RPS trip devices fail open, and power is removed from the aforementioned reactor protection features.

The reliability of the RPS is such that there is a high probability the RPS will accomplish its safety-related functions if a postulated event occurs, consistent with PDCs 22 and 29. No single failure results in loss of the RPS protective functions, consistent with PDC 21 and Section 5 of IEEE 603-2018. Specifics of the minimum redundancy in the RPS to permit periodic testing without compromising the function of the RPS will be provided in an application for the Operating License.

The RPS is functionally independent from the PCS, consistent with PDC 24 and Section 6 of IEEE 603-2018. The system does not share components with the PCS and takes inputs from separate, dedicated sensors. However, safety-related sensors that provide input to the RPS also provide signals to the PCS via a safety-related data diode that uses one-way fiber optic channels. The data diode is integrated into the RPS hardware platform. Consistent with PDC 13, the system uses sensors that monitor variables and systems over their anticipated ranges for normal operation and for postulated event conditions. As discussed in Sections 7.3.1, the RPS uses as input core temperature and vessel level from safety-related sensors. The sensors are discussed in Section 7.5, including the range over which the sensors monitor reactor variables.

Consistent with PDC 3, the RPS is designed to perform its safety function in the event of a fire hazard. The RPS is designed and located to minimize the probability and effect of fires and explosions by the use of low combustible materials and physical separation. These design features, in conjunction with the fire protection program described in Section 9.4, provide assurance that the RPS conforms to PDC 3.

Consistent with PDC 4 and 22, the RPS is designed for the environmental conditions associated with normal operation, maintenance, testing, and postulated events. A description of how the operational and support requirements will be met, including a description of the enclosure that houses the RPS cabinets, will be provided in an application for the Operating License.

The RPS is located in the safety-related portion of the Reactor Building. The Reactor Building is designed to protect internal SSCs from external hazards as discussed in Chapter 3. Consistent with PDC 22, the RPS's location in the safety-related portion of the Reactor Building ensures that natural phenomena will not result in a loss of protection for the RPS.

No portion of the RPS that performs a safety function crosses the seismic isolation moat that is described in Section 3.5. The RPS includes a block to the PCS to prevent any PCS SSCs from interfering with a safety-related SSC's performance of its safety function. The RPS block is accomplished by removing power to a safety-related relay. The safety-related relay is also located in the safety-related portion of the Reactor Building, so no other flexible design features to address differential displacement are required for the RPS to accomplish the block to the PCS during postulated seismic events. This is consistent with PDCs 2 and 4.

The RPS is under the Quality Assurance Program as described in Section 12.9, which is consistent with PDC 1.

Consistent with 10 CFR 50.36, technical specifications contain limiting safety system settings, limiting conditions of operation, surveillance requirements, and action statements applicable to the RPS. Implementation of technical specifications do not interfere with the ability of the RPS to perform its protective function, consistent with PDC 22.

7.3.4 Testing and Inspection

RPS parameters to which operability controls are applied are reactor core temperature, reactor vessel level, source and power range neutron excore detectors. Surveillance intervals are established based on operating experience, engineering judgement, and available vendor recommendations.

Operability tests are performed prior to startup and tests and inspections consistent with the standards discussed in Section 7.3.3.

7.3.5 References

1. Institute of Electrical and Electronics Engineers, Standard IEEE 603-2018, “Standard Criteria for Safety Systems for Nuclear Power Generating Stations.” 2018.

Table 7.3-1: Codes and Standards Applied to the Reactor Protection System

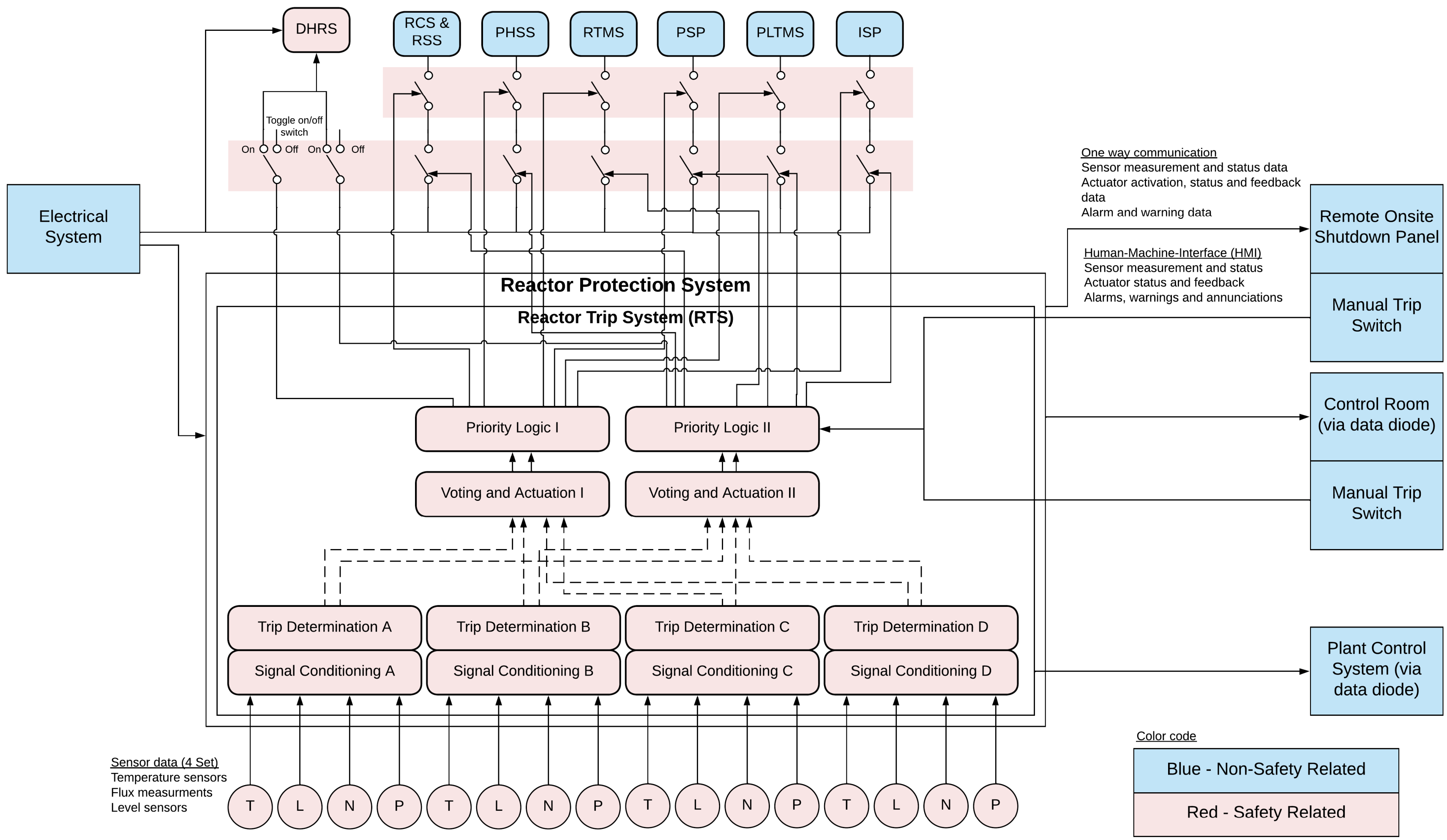
Identifier	Standard or Guidance
1	Electric Power Research Institute, “Guidelines on Evaluation and Acceptance of Commercial Grade Digital Equipment for Nuclear Safety Applications,” TR-106439, November 14, 1996.
2	Institute of Electrical and Electronics Engineers, “IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations,” IEEE Std 323-2003, Piscataway, N.J.
3	Institute of Electrical and Electronics Engineers, “IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” IEEE Std 379-2014, Piscataway, N.J.
4	Institute of Electrical and Electronics Engineers, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” IEEE Std 384-1992, Piscataway, N.J.
5	Institute of Electrical and Electronics Engineers, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” IEEE Std 603-2018, Piscataway, N.J.
6	Institute of Electrical and Electronics Engineers, “ IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations,” IEEE Std 7-4.3.2-2003, Piscataway, N.J.
7	Institute of Electrical and Electronics Engineers, “IEEE Standard for Software Quality Assurance Plans,” IEEE Std 730-2002, Piscataway, N.J.
8	Institute of Electrical and Electronics Engineers, “IEEE Standard for Software Configuration Management Plans,” IEEE Std 828-2005, Piscataway, N.J.
9	Institute of Electrical and Electronics Engineers, “IEEE Standard for Software and System Test Documentation,” IEEE Std 829-2008, Piscataway, N.J.
10	Institute of Electrical and Electronics Engineers, “IEEE Recommended Practice for Software Requirements Specifications,” IEEE Std 830-1998, Piscataway, N.J.
11	Institute of Electrical and Electronics Engineers, “IEEE Standard for Software Unit Testing,” IEEE Std 1008-1987 (R2009), Piscataway, N.J.
12	Institute of Electrical and Electronics Engineers, “IEEE Standard for Software Verification and Validation,” IEEE Std 1012-2004, Piscataway, N.J.
13	Institute of Electrical and Electronics Engineers, “IEEE Standard for Software Reviews and Audits,” IEEE Std 1028-2008, Piscataway, N.J.
14	Institute of Electrical and Electronics Engineers, “IEEE Standard for Developing a Software Project Life Cycle Process,” IEEE Std 1074- 2006, Piscataway, N.J.
15	American National Standards Institute/International Society of Automation, “Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants,” ANSI/ISA 67.02.01-1999, Research Triangle Park, North Carolina.
16	Institute of Electrical and Electronics Engineers, “Standard for Flame-Propagation Testing of Wire & Cable,” IEEE Std 1202-2006, Piscataway, N.J.
17	International Society of Automation, “Setpoints for Nuclear Safety-Related Instrumentation,” ISA-67.04.01-2018, Research Triangle Park, North Carolina.
18	Institute of Electrical and Electronics Engineers, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” IEEE Std 497-2002, Piscataway, NJ.

Table 7.3-2: Reactor Protection System Interlocks and Inhibits

Input Signal to the Reactor Protection System	Interlock or Trip
Fission product accumulation in the core exceeds a defined level	DHRS is actuated *Purpose*: ensure decay heat removal
Fission product accumulation in the core exceeds a defined level	Manual reset for DHRS prohibited *Purpose*: DHRS cannot be disengaged while the core generates decay heat
Low power level AND a minimum defined fission product accumulation in the core is reached*	Manual reset for DHRS available *Purpose:* Prevent overcooling while shutdown
DHRS manual reset is available after RPS actuation NOTE: see row above for the initial conditions for DHRS manual reset availability	Reactor Auxiliary Heating System actuation available. *Purpose*: Allow additional thermal management capabilities following a reactor trip
Loss of normal power AND No transfer to backup power within a defined time period	Movement of reactivity control elements inhibited with manual reset required *Purpose*: prevent inadvertent positive reactivity addition to the core by preventing withdrawal of reactivity control elements when power returns following a reactor trip
Loss of normal power AND Actuation of the RTS	After the RTS trips the PSP, manual reset is required to restart the PSP *Purpose*: Prevent inadvertent restart of the PSP when power is restored
Actuation of the RTS	After the RTS trips the PSP and ISP, the ISP is prevented from restarting unless the PSP is running *Purpose*: Limit overcooling
PSP not running	Trip the ISP and lock out restart of the ISP until the PSP is running. *Purpose*: Limit overcooling
Detection of a break in the PHSS extraction line	Trip the pebble extraction and insertion machines

* The fission product accumulation is based on the operating time and power level relationship.

Figure 7.3-1: Reactor Protection System Trip Logic Schematic



7.4 MAIN CONTROL ROOM AND REMOTE ONSITE SHUTDOWN PANEL

7.4.1 Description

The main control room (MCR) is shared between Unit 1 and Unit 2. The MCR provides means for operators to monitor the behavior of each unit and shared systems, control performance of each unit and the shared systems, and manage the response to postulated event conditions in each unit. Unit-specific remote onsite shutdown panels (ROSP) provide separate means to shut down each unit and monitor plant parameters in response to postulated event conditions. Figure 7.4-1 shows the architecture of the MCR and ROSP.

7.4.1.1 Main Control Room

The MCR contains equipment related to normal operation of the plant. These include operator and supervisor workstation terminals, which provide alarms, annunciations, personnel and equipment interlocks, and process information. These pieces of equipment are the main point of interaction (human/system interface (HSI)) between operators and the PCS and the information coming from the RPS. The terminals are connected to the main plant network through a network switch. The system uses redundant fiber optic communication channels between the PCS and the MCR. Communication from the RPS to the MCR utilizes the data diode discussed in Section 7.3.3 for one-way communication.

The MCR consoles display plant parameters to allow operators to monitor conditions during and following postulated events. Dedicated consoles are provided to control and monitor each unit individually and to control and monitor shared systems. The MCR consoles contain a manual trip switch that propagates through a gateway and through safety-related isolation, which allows operators to initiate a reactor trip, but this is not a credited safety-related function nor credited in the accident analyses (see Chapter 13).

The MCR also contains a central alarm panel for the fire protection system so that operators can monitor the status of fire protection equipment inside the Reactor Building. The central alarm panel includes controls for the ventilation and extinguishing systems related to the response to fires.

7.4.1.2 Remote Onsite Shutdown Panel

The ROSP provides a HSI for plant staff to monitor unit-specific indications from the reactor protection system including operating status of the RTS and the DHRS in the event that the MCR becomes inaccessible or uninhabitable. The ROSP features one-way (read-only) communication with reactor protection system instrumentation signals and the ability to initiate a trip signal from the manual trip button that actuates reactor protection systems. The ROSP is not safety-related and is located in the safety related portion of the Reactor Building for each unit.

7.4.2 Design Bases

Consistent with PDC 19:

- The design of the main control room allows actions to be taken to operate the reactor under normal operating conditions and to monitor it under postulated event conditions.
- The main control room is designed to provide radiation protection allowing access and occupancy of the control room under postulated event conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the event.
- The main control room is designed to be habitable, allowing access and occupancy of the main control room during normal operations and under postulated event conditions.
- An ROSP for each unit is located outside the control room that (1) provides the capability to promptly shutdown the reactor and includes instrumentation and controls to monitor the unit

during shutdown, and (2) provides the capability for subsequent safe shutdown of the reactor through the use of suitable procedures.

7.4.3 System Evaluation

7.4.3.1 Main Control Room

The MCR is located in an auxiliary building separate from the Reactor Building. There are no operator actions performed nor safety-related SSCs located in the MCR that are credited for mitigating the consequences of postulated events described in Chapter 13. Therefore, the MCR and the building that houses the MCR are designed to local building code standards.

The MCR consoles are designed to allow operators to manipulate plant parameters to control the reactor within an acceptable envelope during normal operating conditions, including planned transients. However, no operator actions are credited in the safety analysis of postulated events described in Chapter 13. Although the controls in the MCR are not credited in the safety analysis, the MCR consoles are designed as follows:

- MCR displays implements the guidance from NUREG-1537, Section 7.6, with respect to ease of operators use. The plant controls are grouped and located in the MCR so that operators can easily reach and manipulate the controls. Displays of the results of an operator's actions are readily observable.
- The screen element organization and appearance of the consoles are designed to allow operators to perform actions to operate the reactor under normal operating conditions and to monitor it under postulated event conditions, consistent with PDC 19.
- The MCR consoles are digital interfaces that consider IEEE 7-4.3.2-2003 (Reference 1), as it relates to hardware design, and Regulatory Guide 1.152, Revision 2 "Criteria for Use of Computers in Safety Systems of Nuclear Power Plants." The control consoles in the MCR are designed to display plant parameters that indicate plant status. The MCR consoles display the following information:
  - Plant sensor data and digitally processed parameter outputs based on plant sensor data
  - Indications of PCS and RPS system and equipment status
  - Current and past operating parameter and system information for a duration relevant to inform process and maintenance trending
- Administrative controls are applied to the consoles in the main control room to prevent unauthorized access. MCR console screens are password-protected and include interlocks such as swipe cards and multi-operator coordinated logins to prevent unauthorized access and systems actuation.
- The two reactor units will be controlled individually to achieve criticality and produce thermal power. However, the steam supply from both reactors is regulated through common flow control valves to ensure balanced steam supply to the turbine as well as prevent coolant feedback from one system to the other.

The MCR is located at a distance from the Reactor Buildings such that the radiological consequences of unfiltered air in the MCR during postulated events does not exceed 5 rem TEDE for the duration of the event. The environmental control features for the MCR are separate from the environmental control features for the Reactor Buildings. The analysis of operator dose depends on the final design of the reactor's safety-related SSCs and the analysis will reflect the methods described in Chapter 13. Accordingly, a description of the analysis of operator dose will be provided in the application of the Operating License.

Further, Section 2.2 describes potential chemical hazards related to anhydrous ammonia and chlorine from offsite highway traffic. Sensors are provided for the MCR for anhydrous ammonia and chlorine. When levels of either of those chemicals are detected to be above a threshold value, the ventilation system for the MCR will be turned off and administrative procedures applied until the hazard dissipates.

The design features described above demonstrate conformance with PDC 19.

7.4.3.2 Remote Onsite Shutdown Panel

Consistent with PDC 19, each unit's ROSP is located outside the main control room. A manual trip switch is provided on the ROSP console to open the reactor trip device described in Section 7.3.3; however, this is not a safety-related function. The ROSPs are used in the event the MCR becomes uninhabitable. Communication between the RPS and the ROSP for a unit uses safety-related hardwired communication channels in protected ducting and cable trays. The ROSP displays the parameters necessary to monitor the reactor during shutdown. Suitable procedures for safe shutdown of the reactor will be discussed further in the application for an Operating License.

The ROSP screen design implements the guidance from NUREG-1537, Section 7.6, with respect to ease of operator use. The ROSP controls are grouped and located so that operators can easily reach and manipulate the controls. Displays of the results of an operator's actions are readily observable.

7.4.4 Testing and Inspection

Operability tests will be performed prior to startup and tests and inspections consistent with the standards discussed in Section 7.4.3.

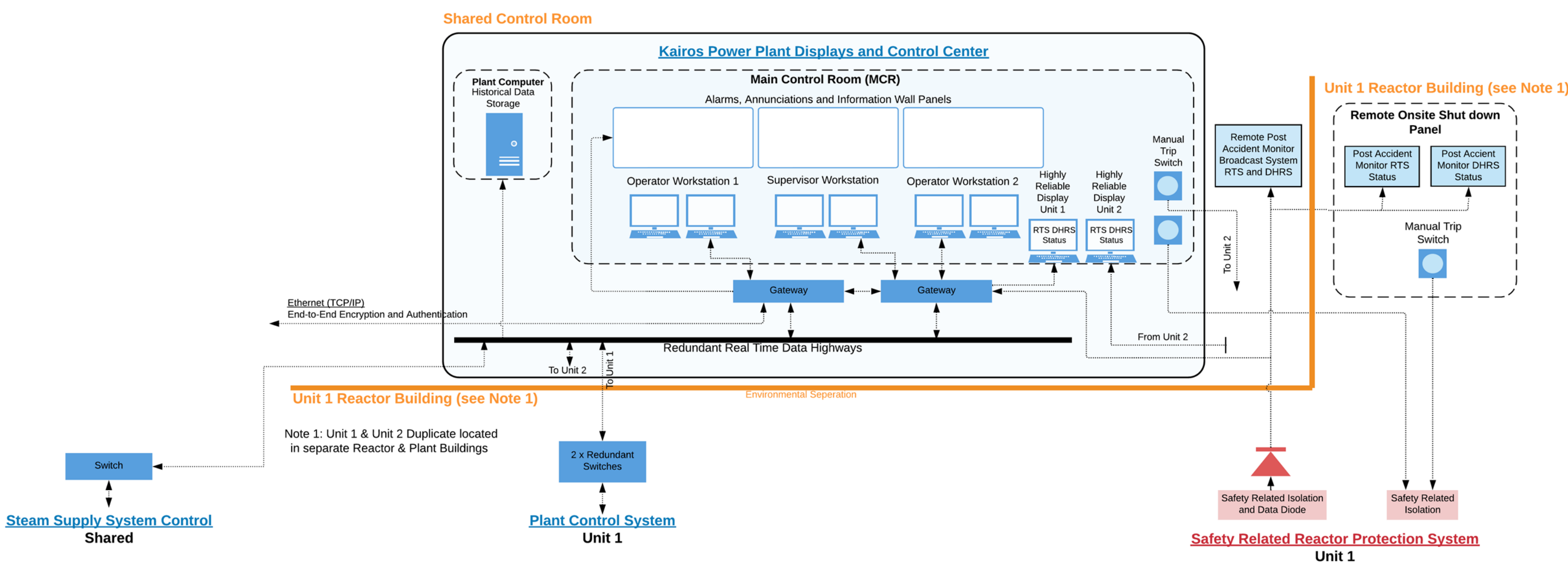
7.4.5 References

1. Institute of Electrical and Electronics Engineers, IEEE Standard 7-4.3.2, "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations." 2003.

Table 7.4-1: Codes and Standards Applied to the Main Control Room and Remote Onsite Shutdown Panel

Identifier	Standard or Guidance
1	Institute of Electrical and Electronics Engineers, "IEEE Standard for Software Quality Assurance Plans," IEEE Std 730-2002, Piscataway, N.J.
2	Institute of Electrical and Electronics Engineers, "IEEE Standard for Software Configuration Management Plans," IEEE Std 828-2005, Piscataway, N.J.
3	Institute of Electrical and Electronics Engineers, "IEEE Standard for Software and System Test Documentation," IEEE Std 829-2008, Piscataway, N.J.
4	Institute of Electrical and Electronics Engineers, "IEEE Recommended Practice for Software Requirements Specifications," IEEE Std 830-1998, Piscataway, N.J.
5	Institute of Electrical and Electronics Engineers, "IEEE Standard for Software Unit Testing," IEEE Std 1008-1987 (R2009), Piscataway, N.J.
6	Institute of Electrical and Electronics Engineers, "IEEE Standard for Software Verification and Validation," IEEE Std 1012-2004, Piscataway, N.J.
7	Institute of Electrical and Electronics Engineers, "IEEE Standard for Software Reviews and Audits," IEEE Std 1028-2008, Piscataway, N.J.
8	Institute of Electrical and Electronics Engineers, "IEEE Standard for Developing a Software Project Life Cycle Process," IEEE Std 1074-2006, Piscataway, N.J.
9	American National Standards Institute/International Society of Automation, "Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants," ANSI/ISA 67.02.01-1999, Research Triangle Park, North Carolina.
10	Institute of Electrical and Electronics Engineers, "Standard for Flame-Propagation Testing of Wire & Cable," IEEE Std 1202-2006, Piscataway, N.J.

Figure 7.4-1: Architecture of the Main Control Room and the Remote Shutdown Onsite Panel



7.5 SENSORS

7.5.1 Description

Sensors are used to provide information about temperature, pressure, neutron count rates, level, flow of the primary coolant and area radiation levels as input to multiple control and protection subsystems. Independent sensors are provided to the reactor protection system and the plant control system. Each section about specific I&C subsystems includes a discussion of the sensors that support that subsystem and the type of sensor used (i.e., analog or digital).

Temperature, pressure, level, and flow sensors measure and monitor plant operating process parameters and are used to control operations and initiate reactor protective actions. Neutron source range sensors provide indication of power level during the initial stages of startup. Gamma radiation monitors provide information about area radiation levels during all plant modes of operation.

7.5.2 Design Bases

Consistent with PDC 1, safety-related sensors are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the safety function to be performed.

Consistent with PDC 2, safety-related sensors are designed to be protected from adverse effects of natural phenomena.

Consistent with PDC 3, safety-related sensors are designed and located to minimize the probability and effect of fires and explosions. Consistent with PDC 13, safety-related sensors monitor process variables and systems over their anticipated ranges for normal operation and for postulated events.

Consistent with PDC 21, RPS sensors are designed with sufficient redundancy and independence to assure that no single failure results in loss of protection function. RPS sensors are designed to permit periodic testing and individual safety-related sensors may be removed from service for testing and maintenance without loss of required minimum redundancy.

Consistent with PDC 22, the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated event conditions do not result in loss of the protection function for RPS sensors. The RPS sensors are designed with sufficient functional and component diversity to prevent the loss of function for the RPS control systems.

Consistent with PDC 24, RPS sensors are functionally independent from the non-safety related sensors.

Consistent with PDC 29, RPS sensors are designed to be redundant to assure there is a high probability of accomplishing the safety-related functions of the RPS in postulated events.

Consistent with 10 CFR 50.36, technical specifications address testing of sensors.

7.5.3 System Evaluation

Safety-related sensors are those that provide input to the RPS safety functions discussed in Section 7.3. Their safety function is to provide sensor input for those plant parameters needed by the RPS to perform its safety functions. Safety-related sensors are also used as inputs to the PCS discussed in Section 7.2, which is not a safety-related function of the sensors. Sensors that provide input to the PCS but not the RPS are classified as non-safety related. In this way the RPS sensors are functionally independent from the non-safety-related sensors, consistent with PDC 24.

The range over which safety-related sensors are designed to monitor process variables reflects the range for postulated events and bounds, with margin, the range for normal operation. For example, safety-related sensors are designed to monitor temperatures between the freezing temperature of the primary coolant, and up to the acceptance criterion for peak vessel and core barrel temperature documented in Table 13.1-1, 750°C. Table 7.5-1 provides the range for parameters over which safety-related sensors are designed to operate. By monitoring variables over the range in Table 7.5-1, the safety-related sensors meet PDC 13.

Non-safety related sensors are designed to monitor process variables over the range of normal operation. For example, the non-safety related sensors monitor temperatures between the temperature of the cold leg of the primary heat transfer loop, 550°C, and the temperature at the outlet of the reactor vessel, 650°C. The parameter range over which sensors that are non-safety related are provided in Table 7.5-2. By monitoring variables over the range in Table 7.5-2, the non-safety related sensors meet PDC 13.

RPS sensors are designed to be redundant so that no single failure results in the RPS losing its ability to perform its safety function. This is consistent with PDC 21. Similarly, RPS sensors are designed to be redundant so that there is a high probability that the necessary inputs are provided to the RPS during normal operations and during a postulated event, including natural phenomena, consistent with PDCs 2, 22 and 29. The number of RPS sensors of each type needed will be consistent with the safety analysis and will be specified in the application for the Operating License (OL). The number of RPS sensors of each type also accounts for sensors that are removed from service for periodic testing, which is discussed further in Section 7.5.4.

The OL application will specify sensors of each type (temperature, pressure, etc.) that are suitable for the environment in which they will function. The sensors are rated to perform in environments described in Tables 7.5-1 and 7.5-2.

Safety-related sensors are designed to operate in normal operating and postulated event environmental conditions. In this way the safety-related sensors are designed to be consistent with the safety functions of RPS that the safety-related sensors support. Non-safety related sensors do not support a safety function and are designed to operate in normal operation environments.

Consistent with PDC 3, the RPS sensors are designed to perform their safety function in the event of a fire hazard. The RPS sensors are designed and located to minimize the probability and effect of fires and explosions by the use of low combustible materials and physical separation. These design features, in conjunction with the fire protection program described in Section 9.4, provide assurance that the RPS sensors conform to PDC 3.

Consistent with PDC 1, safety-related sensors are included in the quality assurance program discussed in Section 12.9 and are maintained as described in Section 7.5.4.

There are no operator actions credited in the safety analysis, therefore there are no sensors for which operators rely on to initiate a safety function.

7.5.4 Testing and Inspection

Consistent with 10 CFR 50.36, technical specifications contain limiting safety system settings, limiting conditions of operation, surveillance requirements, and action statements for sensors. Operability controls are applied to safety-related sensors. Surveillance intervals for sensors are specified based on operating experience, engineering judgement, and vendor recommendation if available.

The safety-related sensors are designed to enable periodic testing. Testing and inspections will be performed so that there is a high probability that the RPS receives the input needed to perform its safety functions described in Section 7.3. The RPS sensors are designed so that individual RPS sensors can be removed from service for testing either without losing minimum redundancy or with a demonstration of acceptable reliability for operation of the protection system (PDC 21). Operability tests are performed prior to startup. Tests and inspections are performed consistent with the standards discussed in Section 7.5.3.

7.5.5 References

None

Table 7.5-1: Parameter Range for Safety-Related Sensors

Parameter	Range
Temperature	450°C – 750°C
Vessel Level	To be provided in Operating License application
Area Radiation	To be provided in Operating License application
Source Range Neutronics	To be provided in Operating License application
Power Range Neutronics	To be provided in Operating License application

Table 7.5-2: Parameter Range for Non-Safety Related Sensors

Parameter	Range
Temperature	550°C – 650°C
Pressure	To be provided in the Operating License application
Flow Rate in the Reactor Vessel	To be provided in the Operating License application
Vessel Level	To be provided in the Operating License application
Area Radiation	To be provided in the Operating License application

Chapter 8

Electric Power Systems

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

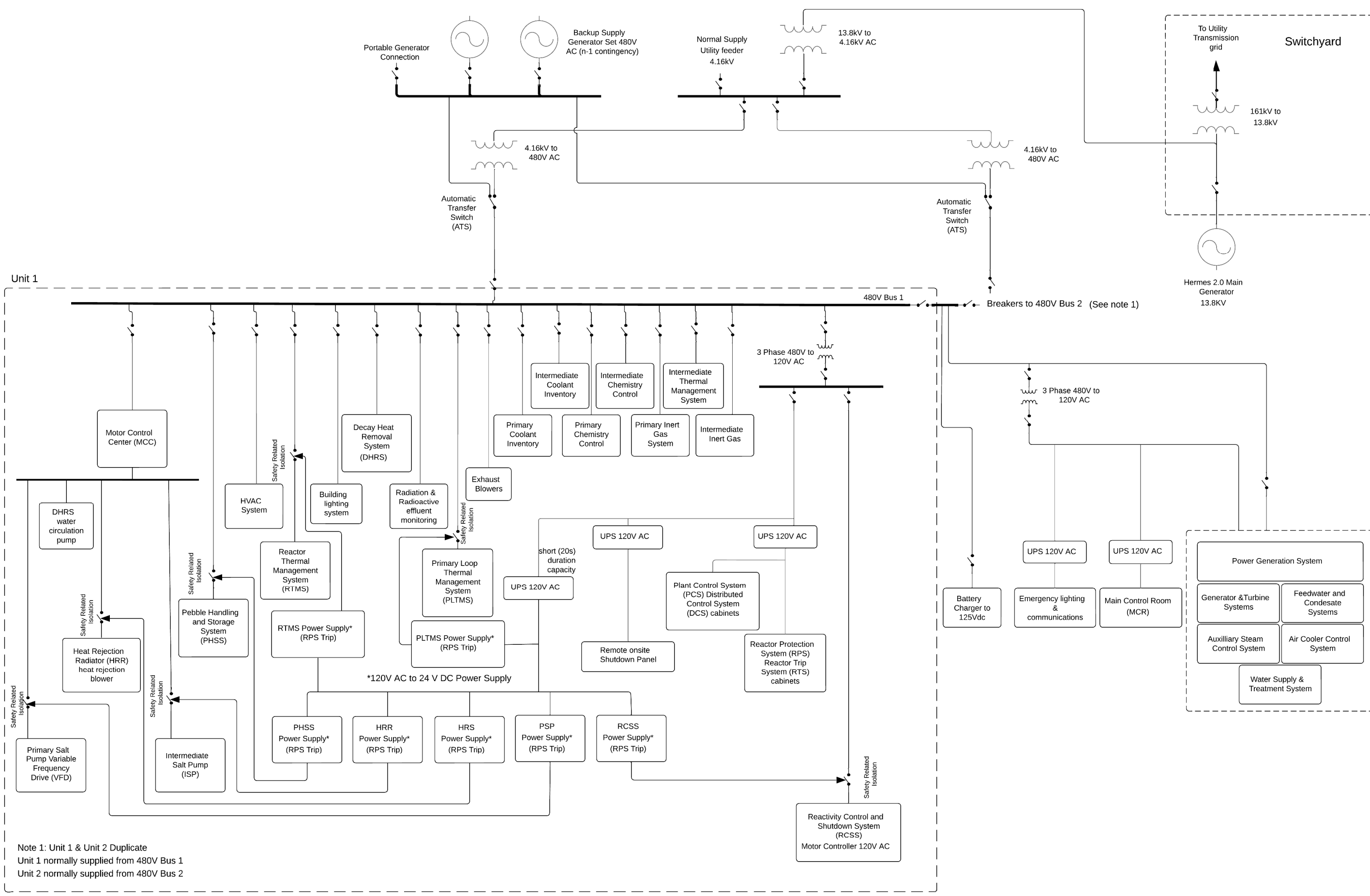
CHAPTER 8 ELECTRIC POWER SYSTEMS

8.1 SUMMARY DESCRIPTION

The purpose of the electrical system is to provide power to support internal operation of plant equipment and to distribute the electric power produced to offsite users. The electrical system consists of the non-Class 1E normal power system (discussed in Section 8.2) and the backup power system (discussed in Section 8.3). During normal operations, offsite AC electrical power is provided either from the offsite local utility to the normal AC power system, or by onsite AC power generated from an operating turbine generator. AC power is shared between Unit 1 and Unit 2. If the normal AC power source fails, the backup power system supplies plant power to designated loads. The backup power system utilizes backup generators and uninterruptible power supplies (UPS) to achieve this function.

Owing to the passive plant design, safety-related structures, systems, and components (SSCs) do not require electric power to perform safety-related functions following a postulated event. Therefore, AC and DC power from off-site or backup power sources is not required to mitigate a postulated event. A simplified diagram of the major electrical system components is provided in Figure 8.1-1.

Figure 8.1-1: Electrical Configuration Diagram



8.2 NORMAL POWER SYSTEM

8.2.1 Description

The normal power system is capable of being supplied by either an offsite power source from the local utility or by AC power generated from the onsite turbine generator system. The local utility provides a medium voltage feeder from offsite. The turbine generator provides output voltage at 13.8 kV. From the point of connection, an appropriate step-down transformer is provided from each of these sources and reduces the voltage to the nominal bus voltage of 480 V, which is distributed to Unit 1 and Unit 2 plant electrical loads as depicted in Figure 8.1-1. A loss of voltage or degraded voltage condition on the normal power system does not adversely affect the performance of safety-related functions.

8.2.1.1 AC Electrical Power

AC power is distributed to the plant electrical loads during startup and shutdown, normal operation, and off-normal conditions. AC power generated from the turbine generator system is provided to an onsite switchyard, via a step up transformer, for distribution to the offsite electrical grid. The AC electrical power components include the following:

- A 4.16 kV/480 V step down transformer connected to a single 4.16 kV offsite electrical power supply from the local utility
- Incoming 13.8 kV feeder from the turbine generator system and associated 13.8 kV/480 V step down transformer
- The low voltage AC electrical power distribution with nominal bus voltages of 480 V and 120 V
- A 13.8 kV/161kV transformer from the turbine generator system to the onsite electrical switchyard

Selected onsite plant electrical loads are supplied with continual AC electrical power via uninterruptible power supplies (UPS). Each UPS provides a highly reliable power supply during normal operations and is automatically configured to provide backup power during a loss of normal electrical power event. The backup function of the UPS is described in Section 8.3.1.2.

8.2.1.2 DC Electrical Power

DC electrical power supply is limited to switchgear control power of 125VDC and unit-specific instrumentation and control functions that require 24 VDC electrical power for operation. The cabinets associated with these functions are equipped with 120 VAC to 24 VDC power supplies, as shown in Figure 8.1-1. AC electrical power is supplied to these cabinets via UPS to ensure continuous, failure-tolerant DC power during normal operation and for a specified maximum duty cycle following a total loss of AC electrical power. DC electrical power is not shared between Unit 1 and Unit 2.

8.2.1.3 Offsite Electrical Power

Offsite AC electrical power is supplied to the facility by a single connection to the existing local utility 4.16kV supply or through a new single overhead connection from the onsite 161kV electrical switchyard to the TVA electrical transmission grid adjacent to the site. The facility does not rely on electrical power to perform safety-related functions, therefore, neither offsite electrical power source is a preferred source. During power operations, the facility is expected to receive power from the onsite turbine generator.

8.2.2 Design Bases

The normal power system does not perform any safety-related functions and is not credited for the mitigation of postulated events. The system is also not credited with performing safe shutdown functions.

8.2.3 System Evaluation

The normal power system is provided to permit functioning of plant SSCs that require electrical power and for providing electric power generation to offsite users. The passive plant design features, based on fundamental physics principles, do not rely on electrical power for safety-related SSCs to perform their safety functions during postulated events. These features demonstrate conformance with the requirement in PDC 17.

As discussed above, the normal power system is not relied on for safety-related SSCs to perform their safety functions following postulated events. Therefore, there are no safety-related portions of the normal power system, and no tests or inspections are required to demonstrate conformance with the requirement in PDC 18.

The design of the normal power system is such that malfunction of the system will not cause reactor damage or prevent safe reactor shutdown. The normal power system ensures that adequate independence is maintained between the non-safety related equipment and circuits of the normal power system and Class 1E instrumentation and control (I&C) equipment and circuits (see Section 8.3.3).

The normal power system is not safety-related, but portions of the system may cross the isolation moat discussed in Section 3.5. The SSCs that cross a base-isolation moat may experience differential displacements as a result of seismic events. The normal power system is designed so that postulated failures of SSCs in the system from differential displacements do not preclude a safety-related SSC from performing its safety function. Design features addressing differential displacement are discussed in Section 3.5. These features demonstrate conformance with the requirements in PDC 2.

To mitigate fault risk, grounding and lightening protection will be implemented in the design. The switchyard protection scheme is consistent with the design approach for protective devices, feeders, branch circuits, and transformers in accordance with industry guideline IEEE Standard 242-2001 (Reference 1). These features demonstrate conformance with the requirements in PDC 2.

The normal power system is designed in accordance with National Fire Protection Association (NFPA) 70, "National Electrical Code" (Reference 2) and Institute of Electrical and Electronics Engineers, IEEE-C2,"National Electrical Safety Code (NESC)" (Reference 3).

8.2.4 Testing and Inspection

Protection devices are capable of being tested, calibrated, and inspected.

8.2.5 References

1. Institute of Electrical and Electronics Engineers, IEEE Standard 242, "IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems (IEEE Buff Book)." 2001.
2. National Fire Protection Association, NFPA 70, *National Electrical Code*. 2020.
3. Institute of Electrical and Electronics Engineers, IEEE-C2, "National Electrical Safety Code (NESC)." 2023.

8.3 BACKUP POWER SYSTEM

8.3.1 Description

The purpose of the backup power system (BPS) is to provide AC electrical power to the essential facility loads (for investment protection purposes) when the normal AC power supplies are not available. The system includes backup generators (shared between the two units) and uninterruptible power supplies (UPS), as well as electrical equipment and circuits used to interconnect the backup generators to the low voltage AC electrical power distribution. In addition, the power distribution design is equipped with a plug-in connection for use with a portable 480 VAC generator to provide power to essential loads in the event the backup generators are unavailable.

8.3.1.1 Backup Generators

The backup generators automatically start in the event of a loss of offsite power and provide backup electrical power to the essential facility loads. There will be at least one redundant generator by design (n-1 contingency), which ensures that sufficient backup power will be supplied in the event of a single generator failure. The backup generators are shared between Unit 1 and Unit 2 and are sufficiently sized to supply the essential loads on both units simultaneously. The backup generators are located on an enclosed skid installation outside the reactor buildings and include conventional components such as:

- Engine starter
- Combustion air intake and engine exhaust
- Engine cooling
- Engine lubricating oil
- Engine fuel (including fuel storage and transfer)
- Generator excitation, protective relaying, and associated instrumentation and controls

The backup generators are provided with controls to facilitate manual startup and shutdown, either locally or from a transfer switch in the main control room (MCR) (see Section 7.4), and to provide for monitoring and control during backup generator operation.

The backup generator switchgear is connected to a distribution switchgear which provides power to 480 V motor control centers (MCCs) and distribution panels. On a loss of normal power, the backup generators start up and the automatic transfer switch (ATS) transfers power supply from the normal power feed to the backup generator feed. A load shedding scheme is employed to ensure that only essential loads are supplied with backup power. A list of the specific essential loads that receive backup power will be provided in the application for an Operating License.

8.3.1.2 Uninterruptible Power Supplies

Selected unit specific electrical loads are also supplied with continuous AC electrical power via uninterruptible power supplies (UPS), as depicted in Figure 8.1-1. Each UPS provides a highly reliable power supply during normal operations and is automatically configured to provide backup power during a loss of normal electrical power event. The UPS are sized to provide sufficient power to those selected loads to maintain functionality during backup generator startup, and for their respective specified maximum duties as described in Section 8.3.3. The UPS are generally not shared between Unit 1 and Unit 2 unless the essential load is common to both units.

8.3.2 Design Bases

The BPS does not perform any safety-related functions and is not credited for the mitigation of postulated events. The system is also not credited with performing safe shutdown functions.

8.3.3 System Evaluation

The normal and backup power systems are designed to prevent interference with safety-related functions. If the backup generators fail during a loss of normal power event, the UPS supplying the reactor protection system (RPS) block loads (as shown in Figure 8.1-1) will fail by design to ensure proper fail-safe functions. This UPS is sized to provide short-term backup power to the RPS block loads, and to lose power on failure of the backup generators. The fail-safe functions are described in further detail in the following paragraphs and in Section 7.3.

To ensure fail-to-safety in the event of a complete loss of AC electrical power, the reactivity control and shutdown system (RCSS) is equipped with a safety-related clutch that requires 24 VDC to remain closed. On a loss of power, the relay opens, and the shutdown elements drop into the reactor by gravity.

To ensure fail-to-safety in the event of a complete loss of AC electrical power, the primary salt pump (PSP) and intermediate salt pump power supplies are equipped with relays requiring 24 VDC to remain closed. On a loss of power, the relays open to prevent inadvertent pump restart on power restoration. A manual reset is required to restart the pumps.

On activation of the decay heat removal system (DHRS), the reactor protection system will remove 24 VDC from the activation circuit relay to prevent inadvertent shut down of the DHRS by operator error.

Equipment for monitoring reactor status will be supplied by UPS until the normal power supply or backup generators are restored.

The BPS is provided to permit functioning of SSCs following a loss of normal power. The passive design features of the reactor, based on fundamental physics principles, do not rely on AC or DC electrical power for safety-related SSCs to perform their safety functions during postulated events. Safe shutdown of the reactor does not rely on AC electrical power from the BPS. These features demonstrate conformance with the requirements in PDC 17.

As discussed above, the BPS is not relied on for safety-related SSCs to perform their safety functions following postulated events. Therefore, there are no safety-related portions of the BPS, and no tests or inspections are required to demonstrate conformance with the requirement in PDC 18.

The backup power system is not safety-related, but portions of the system may cross the isolation moat discussed in Section 3.5. SSCs that cross a base-isolation moat may experience differential displacements as a result of seismic events. The backup power system is designed so that postulated failures of SSCs in the system from differential displacements do not preclude a safety-related SSC from performing its safety function. Design features addressing differential displacement are discussed in Section 3.5. These features demonstrate conformance with the requirement in PDC 2.

The backup power system is designed in accordance with NFPA 70, “National Electrical Code” (Reference 1).

8.3.4 Testing and Inspection

The BPS does not perform any safety functions. Periodic inspection and testing are performed on the BPS for operational purposes.

8.3.5 References

1. National Fire Protection Association, NFPA 70, “National Electrical Code.” 2020.

Kairos Power

Chapter 9

Auxiliary Systems

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 9 AUXILIARY SYSTEMS

This chapter provides an overview description of the auxiliary systems at the reactor facility. Auxiliary systems are those systems not previously described elsewhere in this safety analysis report.

Additional 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.

9.1 REACTOR COOLANT AUXILIARY SYSTEMS

The reactor coolant auxiliary systems (RCAS) are a collection of systems that provide support for the functionality and performance of the reactor coolant (Flibe). The major functions of the system are as follows:

- Remove fission products, activation products, and other chemical impurities and particulates from the reactor coolant.
- Maintain the cover gas atmosphere (pressure and composition) in the head space above the core.
- Provide removal and storage of tritium.
- Control inventory, filling, and draining processes for systems containing reactor coolant, including transfer of coolant into the reactor.
- Provide active and passive thermal management to reactor system components.

The functions of the RCAS are implemented via the following subsystems:

- Chemistry control system (Section 9.1.1)
- Inert gas system (Section 9.1.2)
- Tritium management system (Section 9.1.3)
- Inventory management system (Section 9.1.4)
- Reactor thermal management system (Section 9.1.5)

These systems are further described in the subsections which follow.

9.1.1 Chemistry Control System

9.1.1.1 Description

The chemistry control system (CCS) is used during normal plant operations to monitor the coolant chemistry in the reactor vessel system and primary heat transport system (PHTS), through the interface with the Inventory Management System (IMS), for compliance with Flibe specifications described in Section 5.1. Each unit has its own CCS and there are no components shared between the units. The system extracts coolant samples for offline analysis of the Flibe chemistry, including the content of dissolved radionuclides in the Flibe and loading of insoluble materials. A description of the offline sample analysis equipment will be provided with the application for an Operating License. If the Flibe is not within limits, the IMS may be used to remove and replace a sufficient amount of reactor coolant to restore conformance to the Flibe specification.

The CCS is not credited with performing any safety-related functions.

The CCS is shown in Figure 9.1.4-1.

9.1.1.2 Design Bases

The CCS does not perform any safety-related functions and is not credited for the mitigation of any postulated events. The system is also not credited for performing safe shutdown functions.

Consistent with principal design criteria (PDC) 2, safety-related structures systems and components (SSCs) located near the CCS are protected from the adverse effects of postulated CCS failures during a design basis earthquake.

Consistent with PDC 4, safety-related SSCs located near the CCS are protected from the adverse effects of postulated CCS failures during dynamic events.

Consistent with PDC 70, the CCS is designed to monitor the purity of reactor coolant within specified design limits in consideration of chemical attack, fouling and plugging of passages and radionuclide concentrations, and air or moisture ingress.

Consistent with 10 CFR 20.1406, the CCS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning.

9.1.1.3 System Evaluation

Portions of the CCS may be located in proximity to SSCs that perform safety-related functions. Those safety-related SSCs will be protected from seismic induced failures of the CCS by either seismically mounting the applicable CCS components, confirming sufficient physical separation, or by the erection of barriers to preclude adverse interactions. The CCS is designed to preferentially fail in a way that does not impact the reactor vessel system. This satisfies the requirements of PDC 2.

The CCS is designed such that safety-related systems in proximity to the CCS are protected against the dynamic effects potentially created by the failure of the CCS equipment by either confirming sufficient physical separation, the erection of barriers to preclude adverse interactions, or designing safety-related components to survive adverse interactions. This satisfies the requirements of PDC 4.

The CCS periodically monitors the reactor coolant chemistry using offline sample analysis to ascertain whether the coolant is within the Flibe specifications. The sample analysis examines materials dissolved within the salt (e.g., metal fluoride corrosion products) as well as entrained materials (e.g., fission products and activation products). If the Flibe is not within the specification in KP-TR-005, “Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor,” (Reference 1), or the circulating activity limits in the technical specifications, the IMS (see Section 9.1.4) may be used to remove and replace a sufficient amount of reactor coolant to restore conformance to the Flibe specification. This satisfies the requirements in PDC 70 for monitoring the purity of the reactor coolant.

The CCS interfaces with the IMS and supports containment of fission products and activation products from the reactor vessel system and PHTS. Therefore, the system is designed to minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406.

9.1.1.4 Testing and Inspection

The CCS sample analysis monitors will be periodically calibrated. The components of the CCS are located such that they are accessible for periodic inspection and testing.

9.1.1.5 References

1. Kairos Power, LLC, “Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor,” KP-TR-005-P-A. July 2020.

9.1.2 Inert Gas System

9.1.2.1 Description

The Inert Gas System (IGS) provides argon gas flow to multiple locations in the reactor vessel, pebble handling and storage system (PHSS), primary salt pump (PSP), inventory management system (IMS), reactivity control and shutdown system (RCSS), and the chemistry control system (CCS). Each unit has its own IGS and there are no components shared between the units. The IGS provides cover gas cleanup from impurities such as oxygen, water, and particulates. The cover gas helps to remove tritium and other gases for further downstream treatment as well as prevent accumulation of aerosols in other components.

Radiation monitoring is provided in the IGS for radioactivity in the inert gas space.

The major functions of the IGS are:

- Maintain an inert environment for components that use argon
- Provide inert gas as a purging flow to system components during normal operation and maintenance
- Remove impurities from the cover gas
- Provide transport of tritium for downstream treatment
- Provide reactor coolant motive force during filling and draining operations

The IGS is designed to operate during startup, normal operation, postulated events, shutdowns, and maintenance. During plant start-up, air in the system is removed by a purge with argon gas. The IGS also supports reactor coolant motive force during the reactor coolant filling process and can provide vacuum pressures for high point vent filling if needed. Additionally, during startup, the IGS maintains an inert atmosphere during the heating of the reactor vessel and internal components before Flibe is transferred into the system. Once the plant is operating normally, reactor coolant level control is no longer required to be supported by the IGS.

The IGS provides gas flow to components for Flibe vapor/aerosol control and impurity treatment of the gas. The IGS also aids in reactor coolant draining operations during shutdown by providing required motive pressure. The IGS has a backup argon supply for redundant gas flow as needed.

The IGS interfaces with components on the reactor vessel head to provide purge flow which discharges into the reactor vessel head volume to act as an inert gas blanket above the Flibe free surface. Other IGS interfaces include PHSS, PSP, IMS, RCSS drive mechanisms, and the CCS. The IGS primarily provides Flibe vapor control in components to prevent long term salt deposits from forming in locations below the freezing temperature. The gas flow rates, temperatures, and pressures of the IGS are regulated for each component individually to meet the design requirements. Gas flow is directed through a vent line to the tritium management system (TMS) to capture tritium that has been entrained in the cover gas. Direct interface with the TMS includes gas temperature and pressure control to meet design requirements.

The flow within the IGS is from lower temperature, less contaminated locations to higher temperature, more contaminated locations. Ultimately, the IGS flows through a cleanup system where the argon is filtered, cooled, stripped of tritium, compressed, and stored for reuse. The gas leaving the cleanup system will be recirculated to the major components to maintain the inert atmosphere.

The IGS is not credited with performing any safety-related functions.

Table 9.1.2-1 provides a summary of the key components in the IGS. The IGS design and operating parameters are provided in Table 9.1.2-2. A high-level process flow diagram of the system is provided in Figure 9.1.2-1.

9.1.2.2 Design Bases

The IGS is designed to meet the following PDCs:

Consistent with PDC 2, the safety-related SSCs located near the IGS will be protected from the adverse effects of IGS failures during a design basis earthquake.

Consistent with PDC 4, Environmental and Dynamic Effects Design Basis, safety-related SSCs located near the IGS will be protected from the adverse effects of IMS failures during dynamic events.

Consistent with PDC 64, the IGS is designed to monitor radioactive releases.

Consistent with 10 CFR 20.1406, the IGS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning.

9.1.2.3 System Evaluation

The IGS is not safety-related but may be located in proximity to or may be connected to SSCs that perform safety-related functions. Those safety-related SSCs will be protected from seismically induced failures of the IGS by either seismically mounting the applicable components, confirming sufficient physical separation, or by the erection of barriers to preclude adverse interactions. Also, the IGS is located in safety-related and non-safety related portions of the Reactor Building. As a result, portions of the IGS may cross the isolation moat discussed in Section 3.5. SSCs that cross the base isolation moat may experience differential displacements as a result of seismic events. The IGS is designed so that postulated failures of SSCs in the system from differential displacements do not preclude safety-related SSCs from performing their safety function. Design features addressing differential displacement are discussed in Section 3.5. This satisfies the requirements of PDC 2 for the IGS.

The IGS, excluding the supply tanks and the piping to the buffering tank, is a low-pressure system thus precluding pipe whip. Nearby safety-related systems are not affected by the presence of inert argon gas that might escape during a system failure. These features satisfy PDC 4 for the IGS.

Radiation monitoring is provided in the cover gas space for the evaluation of radioactivity levels in the gas. This monitor supports the evaluation of the radioactive material releases that might occur as a result of a system or fuel failure. This design feature, in part, satisfies PDC 64.

The IGS contains radiological contaminants; therefore, the system is designed to minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406.

9.1.2.4 Testing and Inspection

The IGS backup argon system will be periodically checked for available quantity and for leakage during plant operation. The argon volumes and gas purity will be included in the technical specifications. The IGS includes sampling systems which contain radiation monitoring to evaluate gas radioactivity levels. A limit on circulating gas activity is expected to be included in the technical specifications supporting the determination of the specified acceptable radiological release design limit. IGS system instrumentation is used to track pressures, temperatures, and flow rates.

9.1.2.5 References

None

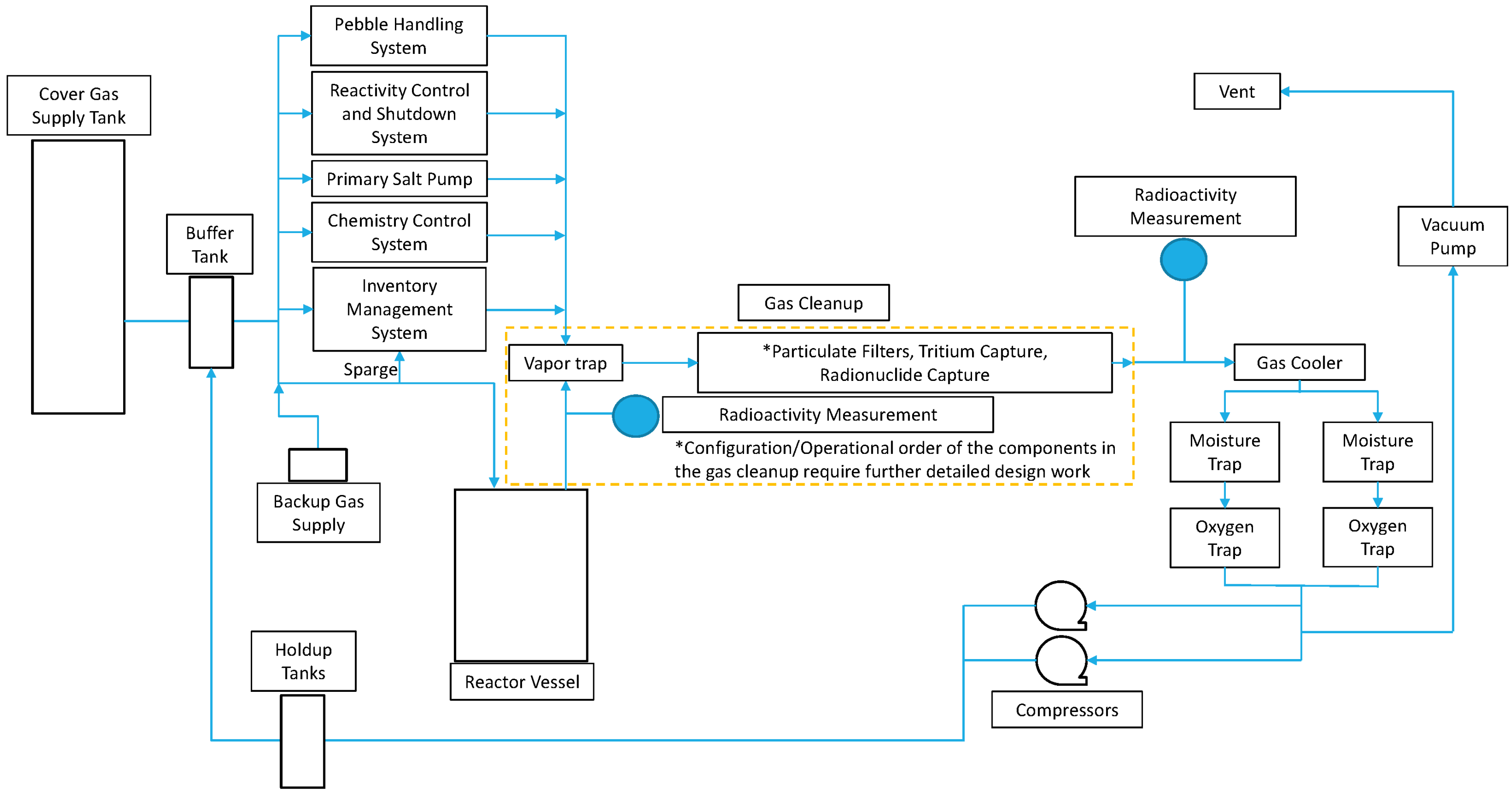
Table 9.1.2-1: Key Components in the Inert Gas System

Inert Gas System Argon Supply	
Component	**Function**
Cover Gas Supply Tank	Replenishment of argon gas to IGS
Argon Buffer Tank	Inventory control
Argon Compressors	Pressurization of IGS
Vacuum Pump	Initial purge of system volume
Cooling System	Cool hot argon gas to low temperatures
Pressure Relief Components	Function for overpressure scenarios
Inert Gas System Argon Cleanup	
Filter	Removes solid particulates, such as graphite dust, Flibe, and corrosion products
Radionuclide Cleaning Equipment	Ability to assist with some radionuclide cleanup in the gas space NOTE: Tritium will be controlled by the separate tritium management system
Vapor Trap	Molten salt vapor removal from gas
Holdup Tanks	Holds gas for a time period
Common Components	
Radiation Monitors	Measure for presence of and changes in radioactivity
Backup Argon Supply	Provides backup supply of argon for redundancy

Table 9.1.2-2: Inert Gas System Design and Operating Parameters

Parameter	**Value**
Working Fluid	Argon
Temperature Range (°C)	25 to 650
Pressure Range (bar(g))	-0.7 to 1.9
Operating Pressure (bar(g))	<0.14

Figure 9.1.2-1: Process Flow Diagram for the Inert Gas System



9.1.3 Tritium Management System

9.1.3.1 Description

The tritium management system (TMS) provides capture of tritium from gas streams in various plant locations in order to reduce environmental releases. Tritium is produced primarily by neutron irradiation of lithium in the salt coolant, such as from lithium-7, lithium-6 remaining after initial enrichment, and lithium-6 produced from transmutation of beryllium-9. Multiple TMS subsystems are integrated into other systems based on the expected tritium distribution among possible transport pathways and the feasibility of tritium capture in each environment. Predictions for the distribution of tritium in primary systems are made using the tritium transport methodology developed for mechanistic source term calculations for KP-TR-012, "KP-FHR Mechanistic Source Term Methodology Topical Report," (Reference 1). The Tritium Management System is a non-safety related system that provides for the collection and disposition pathway. Each unit has its own TMS and there are no components shared between the units.

The primary system functions include:

- Tritium separation from argon in the inert gas system (IGS)
- Tritium separation from argon in the intermediate heat transport system (IHTS) cover gas
- Tritium separation from dry air in the heat rejection radiator (HRR) enclosure
- Tritium separation from dry air in Reactor Building cells
- Final collection and disposition of tritium

Tritium separation from argon in the IGS

Tritium can enter the argon gas in the IGS by direct evolution from the salt to the cover gas in the reactor vessel. Similar evolution phenomena exist in other systems where a Flibe-argon interface is present, such as the chemistry control system (CCS), primary salt pump (PSP), and inventory management system (IMS). Tritium can also enter argon through desorption of sorbed tritium in fuel and moderator pebbles during recirculation in the pebble handling and storage system (PHSS). The sources of tritium from the previously mentioned systems are circulated with argon from the IGS, which then routes the gas flow for tritium cleanup. The TMS subsystem in the IGS (TMS-IGS) uses getter beds to capture tritium from the argon flow.

A simplified process flow diagram for the tritium capture system in the TMS-IGS is shown in Figure 9.1.3-1. The TMS-IGS tritium capture system receives argon flow from the IGS after the gas has been treated with a salt vapor trap and particulate filters. The argon temperature is adjusted with a heat exchanger in the TMS-IGS to bring the gas stream to the getter bed operating temperature. A set of upstream instrumentation monitors the tritium activity and oxygen impurity levels in the gas stream, which are used to inform the saturation or consumption rates of the active getter alloy. Tritium is captured from the argon stream using beds with a fixed packing of getter alloy. An additional tritium measurement is taken downstream of the tritium capture beds. Active bed tritium inventory is monitored based on the difference between upstream and downstream tritium measurements. The tritium capture beds can be bypassed during operations where IGS flow is required but tritium capture is not necessary, such as initial startup sequences. Following the TMS-IGS, the argon is returned to the IGS for further gas treatment.

Tritium separation from argon in the IHTS cover gas

Tritium permeation (as HT or T_2) through the intermediate heat exchanger (IHX) tubing into the BeNaF intermediate coolant (see Section 5.2) is anticipated during normal operating conditions. Tritium permeating into the IHTS is expected to remain in the chemical form of HT or T_2, and these forms could permeate out of the IHTS piping or through the superheater into the steam cycle. Permeation of tritium beyond the IHTS will be mitigated by collecting tritium in the cover gas after converting the HT/T_2 to TF.

The conversion of HT/T_2 to TF is accomplished through isotopic exchange reactions with HF added to the IHTS. Anhydrous HF will be introduced into the IHTS through minor additions to the argon cover gas of the intermediate salt vessel (ISV), where HF will then diffuse into the coolant and equilibrate with the cover gas composition.

The tritium conversion to TF within the BeNaF coolant enhances tritium evolution to the ISV cover gas. A tritium capture subsystem in the ISV cover gas sequesters tritium from the IHTS. The process for tritium capture in the ISV cover gas is illustrated in Figure 9.1.3-2. Argon with HF additive, converted TF, and residual HT/T_2 from the ISV cover gas is supplied by the intermediate inert gas system to the TMS tritium capture subsystem located in the IHTS (TMS-IHTS). A means of oxidation, such as a catalytic conversion bed, is present prior to the molecular sieve beds to convert tritium present and increase the fraction of HTO/T2O available for capture by the molecular sieve desiccant. A heat exchanger between the tritium conversion bed and tritium capture bed reduces the temperature of the gas stream from the operating temperature of the oxide catalyst to near-room temperature to increase the water loading capacity of the molecular sieve desiccant. The concentration of tritium in the gas stream is measured before and after the tritium conversion bed and tritium capture bed, which allows the tritium inventory in the beds to be monitored over time. Since HF is removed along with TF on each pass through the tritium management system, an HF Addition System downstream of the Tritium Conversion Bed and Tritium Capture Bed will replenish the HF additive to the argon. Following the removal of tritium and reestablishment of argon and HF composition, the gas mixture is returned to the intermediate inert gas system.

Tritium separation from dry air in the HRR enclosure

As with the IHX, tritium is expected to permeate through the heat transfer surface of the HRR into the surrounding air. During startup and normal shutdown conditions when the heat rejection blower is on, tritium permeation through the HRR will be discharged through the HRS as a gaseous effluent (see (section 11.1.5). For normal power operations when reactor thermal power, tritium generation rates, and tritium permeation rates are increased, the heat rejection blower will be off and the air flow path in the HRS ducting will be blocked. Air within the isolated HRR enclosure in these scenarios will be recirculated through a tritium capture subsystem interfacing with the HRR to allow for mitigation of HRR permeation releases, as shown in Figure 9.1.3-3.

In the HRR enclosure tritium capture subsystem (TMS-HRR) loop, air suction is drawn from the HRR enclosure and reduced in temperature by a regenerative heat exchanger cooled by returning air. The TMS-HRR provides for tritium capture of both HT and HTO through oxidation of HT in a tritium conversion bed containing high surface area oxidizing catalyst. Process stream temperatures are then reduced by an externally cooled non-regenerative heat exchanger, which then allows for efficient separation of HTO from the recirculating air by molecular sieve desiccant in the tritium capture bed downstream. Tritium concentrations within the TMS-HRR recirculating air loop are assessed before and after the conversion and capture beds to monitor tritium inventory over time. Following tritium capture, the recirculating air is used to cool incoming air in the regenerative heat exchanger and is then discharged into the HRR enclosure.

Tritium separation from dry air in Reactor Building cells

Tritium permeates through the structural metals of the primary heat transport system (PHTS) at a lower overall rate than the intermediate heat exchanger due to the reduced Flibe-facing surface area. Tritium capture is carried out in the environments surrounding the reactor vessel, primary loop piping, and intermediate loop piping to collect tritium which permeates through structural metals, as well as any tritium released from limited gas leakage out of interfacing systems, such as the IGS, PHSS, CCS, and IMS, during normal operations or maintenance activities. The tritium which permeates through metallic boundaries or leaks from the inert cover gas of these systems is expected to predominantly exist in the form of HT or T_2. Reactor Building environments where tritium capture occurs are isolated into building cells where favorable conditions for tritium capture can be readily maintained. Molecular sieve capture beds are used for tritium capture systems in Reactor Building cells and are designed to accommodate additional moisture loads produced from in-leakage of ambient air. The molecular sieve capture beds are used to capture tritiated water in the cells along with moisture introduced from air ingress. A means of oxidation, such as a catalyst bed, are present prior to the building cell molecular sieve beds to convert any unoxidized tritium present and increase the fraction of HTO/T_2O available for capture by the sieve. To minimize tritium effluent, the exhaust flow used to maintain the cells at negative pressure is extracted from the TMS outlet flow and directed to the reactor building filter and exhaust system described in section 9.2. An example process diagram for the integration of a tritium capture system into the reactor cell HVAC system is shown in Figure 9.1.3-4; tritium capture systems integrated into Reactor Building cells other than the reactor cell follow a similar process.

Final collection and disposition of tritium

The previously described tritium capture systems each produce a unique stream of tritium capture materials. Tritium capture in the IGS results in the formation of a stable metal tritide from the getter alloy, while the IHTS, HRR enclosure, and Reactor Building cell capture systems produce tritiated water stored in a molecular sieve. The molecular sieve from the IHTS and the HRR enclosure will result in higher specific tritium activities than the building cell molecular sieve due to lower levels of moisture ingress, and thus lower dilution from H_2O. Following their in-service duty cycles, the tritium capture materials are stored in sealed canisters which can withstand pressure increases caused by tritium decay into helium-3. For the molecular sieve vessels, a catalytic recombiner material is added to convert hydrogen or HT/T_2 produced by radiolysis back to a water form to allow for re-adsorption by the sieve. Tritium capture materials which are intended to be shipped from the site will be contained in a package which meets appropriate Department of Transportation regulations. In accordance with 10 CFR 71.51, Type A and Type B packaging canisters are used. When tritium content would exceed the limit of 1,080 Ci, Type B canisters are used for temporary storage and shipping as needed. Canisters with a tritium content of less than 1,080 Ci are shipped from the site using Type A canisters.

9.1.3.2 Design Bases

The TMS satisfies the following Principal Design Criteria (PDC):

Consistent with PDC 2, safety-related SSCs located near the TMS are protected from the adverse effects of TMS failures during a design basis earthquake.

Consistent with PDC 13, proper instrumentation is provided to measure tritium inventories in the TMS and demonstrate compliance with imposed inventory limits.

Consistent with PDC 60, tritium capture functions performed by the TMS assist in controlling releases of radioactive materials to the environment.

Consistent with PDC 64, the TMS is designed support the monitoring of tritium releases.

Consistent with 10 CFR 20.1406, the TMS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning.

9.1.3.3 System Evaluation

The TMS does not perform any safety-related functions and is not credited for the mitigation of any postulated events. The system is also not credited for performing safe shutdown functions.

Portions of the TMS may be located in proximity to SSCs that perform safety-related functions. Those safety-related SSCs will be protected from seismic induced failures of the TMS by either seismically mounting the applicable TMS components, confirming sufficient physical separation, or by the erection of barriers to preclude adverse interactions. This satisfies the requirements of PDC 2 for the TMS.

The total tritium inventory in the TMS is monitored and maintained below a specified limit. The TMS tritium inventory upper bound limit for tritium not stored inside Type B containers, is set such that dose corresponding to a full release of TMS tritium in a postulated event is bounded by the tritium release dose from reactor vessel system and PHTS tritium inventories included in the maximum hypothetical accident (MHA) analysis. By maintaining the potential tritium release doses bounded by the MHA (see Table 13.2-1), the hypothetical tritium releases from the TMS satisfy the accident dose limits of 10 CFR 100.11.

The TMS tritium inventory includes active capture beds and previously used beds stored in on-site Type A canisters. Tritium capture beds packaged in Type B canisters located on-site are not included in the total tritium inventory. Tritium inventories are maintained below a specified TMS limit, either through radioactive decay, or shipment of Type A canisters off-site for disposal or beneficial use. Shipment of used beds which include greater than 1,080 Ci of tritium requires a certified Type B canister. Tritium stored in approved Type B canisters are excluded from the inventory limit that could be released in a postulated event. This is acceptable because credible environmental hazards associated with the canister storage location in the plant are less severe than the hypothetical transportation accident conditions required for Type B canister qualification.

The total tritium inventory in the TMS is monitored in order to comply with the inventory limits set by MHA assumptions and dose limits in 10 CFR 100.11. Quantities included as part of the TMS total tritium inventory are the tritium stored in active capture beds of TMS subsystems plus the tritium inventory of previously used beds in storage inside the plant in containers not qualified for tritium containment. During operation of TMS capture systems, the buildup of tritium inventory is monitored over time by measuring the difference in tritium activity in process streams upstream and downstream of the active capture beds. The total tritium inventory of capture beds is also measured with a non-destructive analysis method after each bed’s in-service duty cycle is complete. In compliance with PDC 13, tritium monitoring sensors are selected to provide measurements over a range of anticipated tritium activities where measurements are needed.

The TMS maintains a minimum level of overall tritium capture capacity in order to minimize tritium releases from the plant and satisfy PDC 60. Tritium releases in effluents are controlled within the effluent limits in 10 CFR 20.

Radiation monitoring is provided in the TMS for the evaluation of tritium levels in TMS subsystems. This monitor supports evaluation of radioactive material releases that might occur as a result of a system failure. This design feature, in part, satisfies PDC 64.

The system contains radiological contaminants; therefore, the system is designed to minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406.

9.1.3.4 Testing and Inspection

The TMS tritium inventory is monitored by measurement, or by bounding calculations when measuring equipment is inoperable.

Tritium capture functions performed by the TMS also assist in maintaining the quantity of tritium in the primary reactor coolant and intermediate reactor coolant below an upper bound limit set for each coolant.

9.1.3.5 References

1. Kairos Power LLC, “KP-FHR Mechanistic Source Term Methodology Topical Report,” KP-TR-012-P-A. May 2022.

Figure 9.1.3-1: Process Flow Diagram for the Tritium Capture System in the Inert Gas System

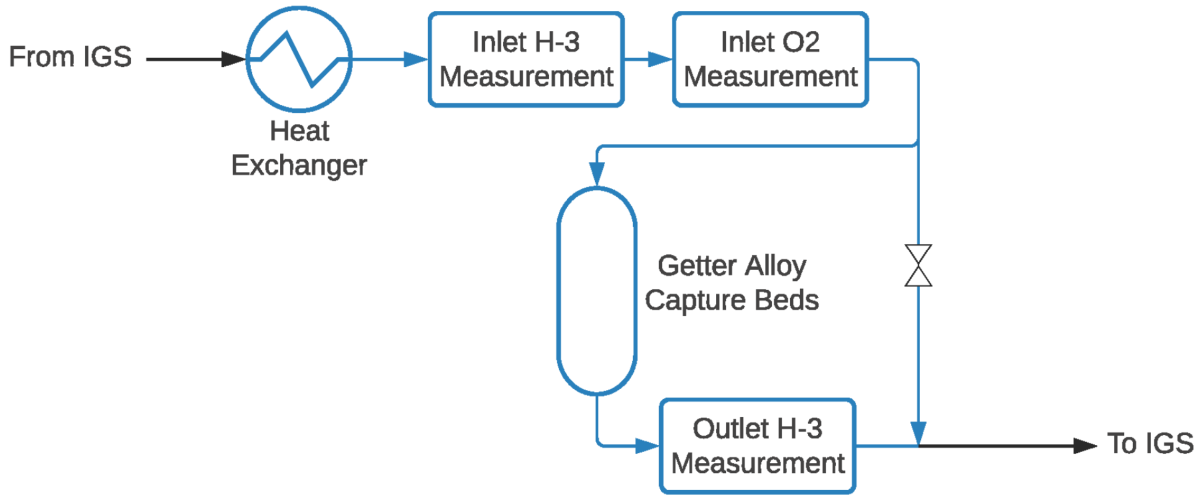


Figure 9.1.3-2: Process Flow Diagram for the Tritium Capture System in the Intermediate Heat Transport System

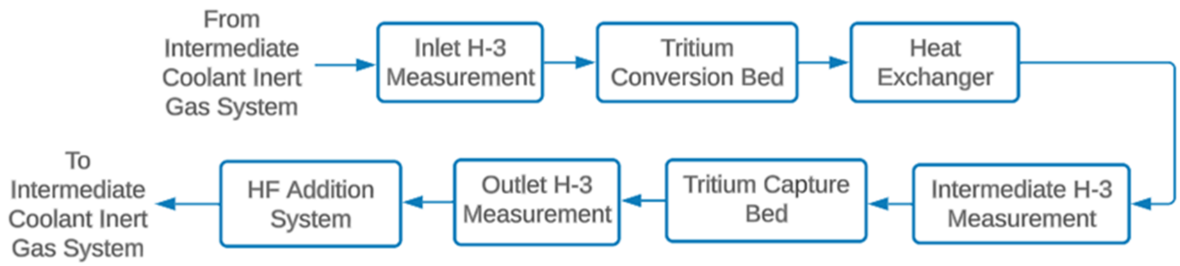


Figure 9.1.3-3: Process Flow Diagram for the Tritium Capture System in the Heat Rejection Radiator

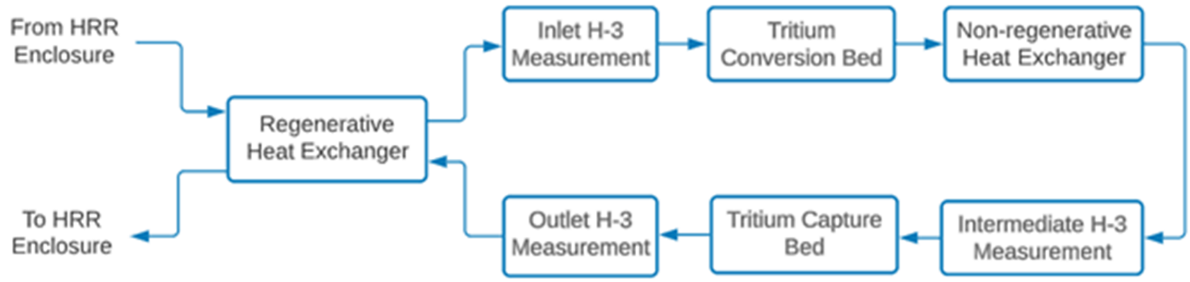
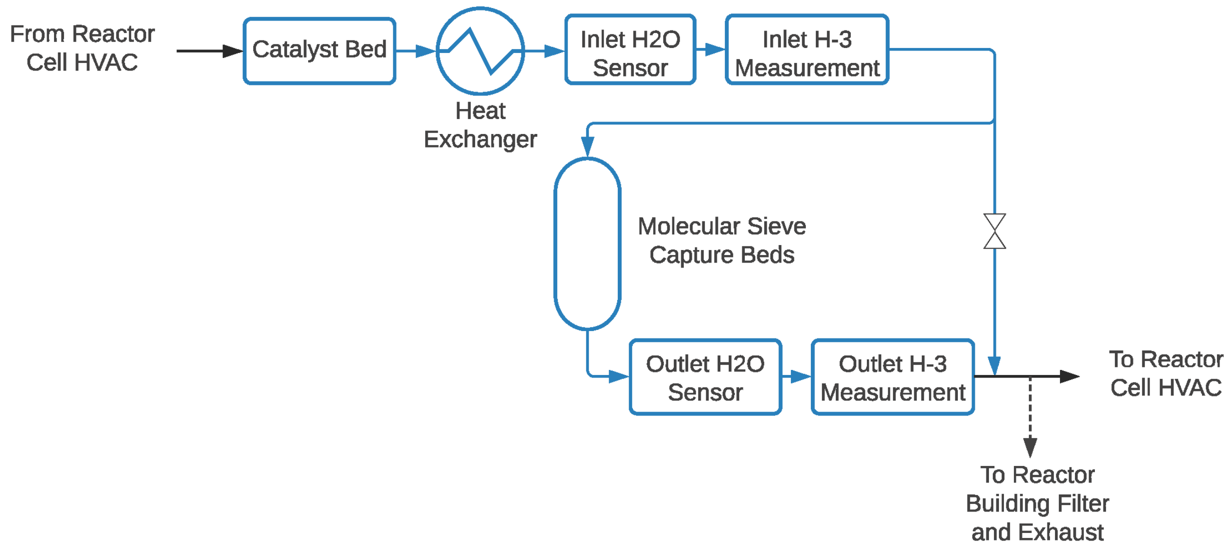


Figure 9.1.3-4: Process Flow Diagram for the Tritium Capture System in the Reactor Cell



9.1.4 Inventory Management System

9.1.4.1 Description

The inventory management system (IMS) consists of tanks, pumps, valves, and lines used to add and remove reactor coolant, and to maintain the desired level and volume within reactor coolant containing systems and components, i.e., the reactor vessel (RV; see Section 4.3), primary heat transport system (PHTS; see Section 5.2), and chemistry control system (CCS; see Section 9.1.1). Each unit has its own IMS and there are no components shared between the units.

Additionally, the IMS includes an interface for new reactor coolant delivery and used reactor coolant removal: the solid inventory management system, described in Section 9.1.4.1.4 below.

The IMS free space is filled with inert gas from the inert gas system (IGS; see Section 9.1.2). The IGS can apply inert gas pressure or vacuum on the IMS tanks to circulate cover gas through the TMS (see Section 9.1.3) or initiate reactor coolant transfers between tanks. Electrical heating and thermal insulation of the tanks, pumps, valves, and piping is provided to maintain the reactor coolant in a liquid phase for system operations. Process sensors are included for use by the plant control system (see Section 7.2) to monitor the reactor coolant inventory (e.g., load cells and coolant level sensors) and temperature in the system tanks. A process flow diagram of the IMS is provided in Figure 9.1.4-1.

The IMS reactor coolant transfer lines are constructed of stainless steel and designed per ASME B31.3 2016 (Reference 1). The IMS tanks are constructed of stainless-steel and are designed per ASME BPVC, Section VIII 2015 (Reference 2). The tanks and reactor coolant transfer lines are designed and fabricated to meet the pressure, mechanical loads, corrosion, and temperature requirements of the system.

The three IMS tank functions are the RV coolant level management tank function, the RV reactor coolant fill/drain tank function, and the PHTS fill/drain tank function. The IMS design grants the operational flexibility to combine the tank functions into a single tank or to separate the tank functions into multiple tanks. The reactor coolant can be transferred between physical IMS tanks driven by a cover gas pressure differential. The IMS tanks and their functions are described in Sections 9.1.4.1.1 through 9.1.4.1.3.

9.1.4.1.1 RV Coolant Level Management Tank

The RV coolant level management tank provides the means to maintain the level of reactor coolant in the RV through a transfer line, a dip tube, and an overflow weir. The transfer into the RV is pump driven through the dip tube. The return flow is collected by an overflow weir and the transfer is gravity driven back into the RV level management tank. The coolant from the RV level management tank is also pumped through the CCS (see Section 9.1.1) through a separate loop. The simultaneous operation of both circulation loops enables the RV coolant level management and CCS functions. If a small leak occurs in the PHTS and the RV coolant level management tank pump is operating to maintain constant level in the RV, the RV coolant level management tank will gradually lose inventory. The RV coolant level management tank has load cells to measure coolant inventory and detect changes associated with small leaks.

The RV level management tank volume is designed to hold reactor coolant inventory necessary to perform RV level management and CCS recirculation functions. However, the RV coolant level management tank is not credited for maintaining the reactor coolant level during postulated events. Additional details of the IMS vessel level monitoring will be provided with the application for an operating license.

9.1.4.1.2 RV Fill/Drain Tank

The RV fill/drain tank provides a means of filling and draining the RV through a transfer line and a dip tube. The transfer into the RV is pump driven and the transfer out of the RV is gravity driven. The RV fill/drain tank transfer line is equipped with a passive RV isolation system to prevent unintentional draining, which is discussed in Section 9.1.4.3. The RV fill/drain tank is sized to hold the RV coolant inventory.

9.1.4.1.3 PHTS Fill/Drain Tank

The PHTS fill/drain tank provides a means of filling and draining the reactor coolant from the PHTS (see Section 5.1), including the intermediate heat exchanger (IHX), through a transfer line. The PHTS drain is gravity driven and the fill is pump driven between the PHTS fill/drain tank and the PHTS.

The PHTS fill/drain tank is sized to hold the PHTS and IHX reactor coolant inventory.

9.1.4.1.4 Solid IMS

New and used reactor coolant is stored in transfer canisters used to transport reactor coolant to and from the site in solid state at ambient temperature. Within the IMS, the reactor coolant is transferred – in liquid form – through transfer lines, driven by a cover gas pressure differential. The solid IMS function is to melt new reactor coolant in the canisters prior to a transfer into the IMS or to freeze the used reactor coolant in the canisters following a transfer from the IMS. The used reactor coolant presents a potential hazard due to radiological contamination.

The transfer canisters are constructed of stainless-steel and are designed per ASME BPVC, Section VIII. The transfer canisters are designed and fabricated to meet the pressure, mechanical loads, corrosion, and temperature requirements of the system.

9.1.4.2 Design Bases

Consistent with PDC 2, safety-related SSCs located near the IMS are protected from the adverse effects of IMS failures during a design basis earthquake.

Consistent with PDC 4, safety-related SSCs located near the IMS are protected from the adverse effects of IMS failures during dynamic events.

Consistent with PDC 33, the design of the IMS includes design features to limit the loss of reactor vessel coolant inventory in the event of breaks in the system.

Consistent with PDC 70, the IMS is designed to maintain the purity of reactor coolant within specified design limits.

Consistent with 10 CFR 20.1406, the IMS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning.

9.1.4.3 System Evaluation

The IMS does not perform safety-related functions and is not credited for the mitigation of postulated events. The system is also not credited for performing safe shutdown functions. The system is not credited to maintain the integrity of the reactor coolant pressure boundary.

Portions of the IMS may be located in proximity to SSCs with safety-related functions. Those safety-related SSCs are protected from failure of the IMS during a design basis earthquake by either seismically mounting the applicable IMS components, physical separation, or barriers to preclude adverse interactions. The IMS is designed to preferentially fail in a way that does not impact the RV system. This satisfies PDC 2 for the IMS.

The IMS is designed such that safety-related systems in proximity to the IMS are protected against the dynamic effects potentially created by the failure of IMS equipment. The IMS is a low pressure system, as the reactor coolant pressures are bounded by the reactor coolant static head pressures, thus precluding pipe whip. This satisfies PDC 4 for the IMS.

The IMS is designed to preclude the inadvertent draining of the RV during normal operation and during RV fill/drain operations. During normal operation, when the reactor vessel is fueled, the RV fill/drain transfer line is equipped with passive RV isolation features such as caps, flanges, and/or a transfer line disconnect, designed to preclude inadvertent reactor coolant draining from the RV by siphoning. In the event of a leak in the RV fill/drain transfer line, while connected to the reactor vessel during fueled operation, the reactor coolant leak is detected by the plant control system, the PSP is tripped, and the RV cover gas pressure is limited to an upper bound thus precluding the ejection of reactor coolant through the transfer line dip-tube. During RV fill/drain operations, the reactor vessel is defueled, and the fill/drain line is connected, an isolation valve is used to interrupt the reactor coolant flow and a cover gas inlet is used to break the siphon in the transfer lines. These design features satisfy the requirements of PDC 33.

The RV coolant level management line short dip tube and overflow weir designs preclude inadvertent reactor coolant draining from the RV into the RV level management tank. As level drops in response to a break in the reactor coolant level management line, cover gas would fill the short dip tube and would break the siphon. Additionally, the overflow weir is designed in a way that precludes the uncovering of fuel due to thermal expansion of the reactor coolant. In the event of a leak in the RV level management tank or transfer line, the reactor coolant leak is detected by the plant control system, and the pump for the reactor level management is tripped to minimize the overflow of reactor coolant from the RV through the overflow weir. As level drops in response to a break in the reactor coolant level management line, cover gas would fill the overflow weir and would break the siphon. This design configuration satisfies the requirements of PDC 33.

The IMS encompasses a PHTS drain line, equipped with a PHTS drain valve, which interfaces with the PHTS fill/drain tank. The PHTS design contains an RV anti-siphon feature (see Section 5.1), thus precluding inadvertent reactor coolant drain from the RV, precluding the IMS from draining the RV. These design features satisfy the requirements of PDC 33.

The makeup inventory function of IMS is not relied on to mitigate the consequences of a postulated event. As described in Section 4.3, the safety-related portions of the reactor coolant boundary are limited to the reactor vessel and a failure of the reactor vessel is precluded by design. Therefore, the makeup functional requirements of PDC 33 have been addressed by design.

The CCS (see Section 9.1.1) periodically monitors the reactor coolant chemistry using offline sample analysis to ascertain whether the coolant is within the Flibe specifications in KP-TR-005-P-A, “Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor,” (Reference 3), or within the circulating activity limits in the technical specifications. If the Flibe is not within limits, the IMS may be used to remove and replace a sufficient amount of reactor coolant to restore conformance to the Flibe specification. This satisfies the requirements of PDC 70 for maintaining the purity of the reactor coolant.

The system is expected to handle reactor coolant with fission as well as activation products; therefore, the system will be designed to minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406.

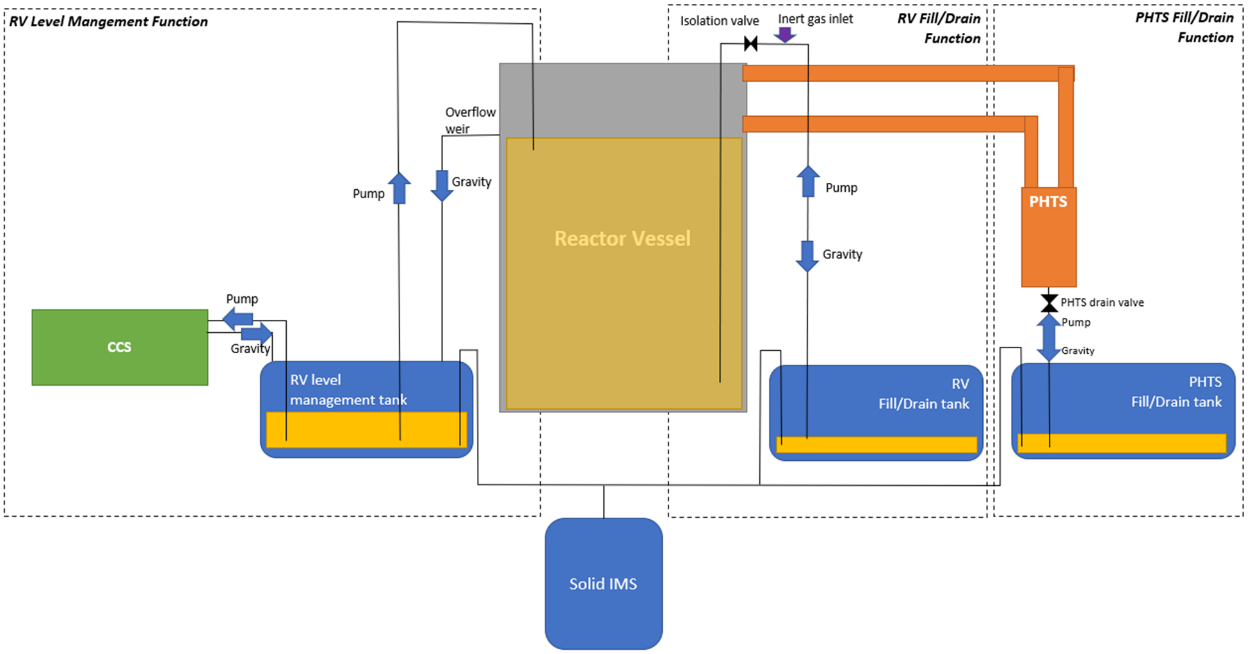
9.1.4.4 Testing and Inspection

The components of the IMS, including valves, tanks, pumps and other components, are located such that they are accessible for periodic inspection and testing.

9.1.4.5 References

1. American Society of Mechanical Engineers, “Process Piping,” ASME B31.3. 2016.
2. ASME, Boiler and Pressure Vessel Code, Section VIII, “Rules for Construction of Pressure Vessels,” New York, NY. 2015.
3. Kairos Power, LLC, “Reactor Coolant for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor,” KP-TR-005-P-A. July 2020.

Figure 9.1.4-1: Inventory Management System



9.1.5 Reactor Thermal Management System

9.1.5.1 Description

The reactor thermal management system (RTMS) consists of two primary subsystems - the equipment and structural cooling system (ESCS) and the reactor auxiliary heating system (RAHS). Neither subsystem is credited with performing a safety-related function. Each unit has its own RTMS and there are no components shared between the units.

The RTMS interfaces with the PSP (see Section 5.1), reactor vessel (see Section 4.3), reactor vessel support system (RVSS) (see Section 4.7), reactor cavity concrete and steel structure (Section 3.5), IGS (see Section 9.1.2), PHSS (see Section 9.3), IMS (see Section 9.1.4) and the RCSS (see Section 4.2) as shown in Figure 9.1.5-1.

9.1.5.1.1 Equipment and Structural Cooling System

The ESCS removes heat from selected SSCs in the reactor cavity area to maintain the operational temperature limits of those structures and components during normal operations. This is accomplished by active heat removal as well as high temperature, load-bearing, irradiation-hardened insulation on SSC surface areas.

Heat removed by the ESCS is transferred to the component cooling water system (CCWS) (see Section 9.7.3). SSCs insulated by the ESCS include the RVSS support columns, the steel liner of the concrete structure surrounding the reactor cavity, the PSP, and multiple reactor vessel head components including the IMS, IGS and PHSS penetrations. The systems that are also actively cooled by the ESCS during normal operation are the steel liner of the concrete structure, the RCSS, and the PSP. The steel liner of the concrete structure is water cooled. Heat removal from the PSP is achieved with gas cooling by active components such as fans, blowers, and pumps in a closed loop system.

9.1.5.1.2 Reactor Auxiliary Heating System

The RAHS is designed to pre-heat the reactor vessel and to ensure Flibe in the vessel is maintained above a minimum operating temperature. The system consists of electric heaters adjacent to the heated surface. The system provides the initial startup heating required to achieve and maintain operational temperatures for the reactor vessel, PSP, and reactor vessel internals as well as SSCs such as IMS, IGS, and PHSS reactor vessel head penetrations prior to the availability of nuclear heating.

The RAHS will be used upon initial commissioning to “bake” residual moisture out of the reactor vessel internal graphite structures and, the reactor vessel, to preclude corrosion upon contact with the introduction of the molten salt reactor coolant. The reactor coolant melting point is high; therefore, pre-heating of the vessel and internals prevents thermal shock from damaging the reactor system.

9.1.5.2 Design Bases

The reactor thermal management system does not perform any safety-related functions and is not credited for the mitigation of any postulated events. The system is also not credited for performing safe shutdown functions.

Consistent with PDC 2, safety-related SSCs located near the RTMS are protected from the adverse effects of RTMS failures during a design basis earthquake.

Consistent with PDC 44, the ESCS provides a means to transfer heat from SSCs to an ultimate heat sink under normal operating conditions.

Consistent with PDC 45, the ESCS is designed to permit appropriate periodic inspection.

Consistent with PDC 46, the ESCS is designed to permit appropriate periodic functional testing.

Consistent with PDC 71, the RAHS is designed to ensure that the temperature distribution and rate of change of temperature in systems and components containing reactor coolant are maintained within design limits.

Consistent with 10 CFR 20.1406, the RTMS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning.

9.1.5.3 System Evaluation

Portions of the RTMS may be located in proximity to SSCs with safety-related functions. Those safety-related SSCs are protected from failure of the RTMS during a design basis earthquake by either seismically mounting the applicable RTMS components, physical separation, or barriers to preclude adverse interactions. These features along with the seismic design discussed in Section 3.5 demonstrate conformance with PDC 2 for the RTMS.

The ESCS interfaces with the CCWS to transfer heat from SSCs to the environment under normal operating conditions. The system does not perform safety-related functions and is not credited for the mitigation of postulated events. The ESCS is also not credited with performing safe shutdown functions. The ESCS is designed to detect gas and water leaks and isolate breaches in the system via the plant control system (see Section 7.2). The system is also designed to permit appropriate periodic inspection and testing to ensure the integrity and capability of the system to cool SSCs and to adequately interface with the CCWS to transfer heat to the ultimate heat sink. This satisfies the requirements of PDC 44, 45 and 46.

The RAHS does not perform safety-related functions and is not credited for the mitigation of postulated events. The RAHS ensures sufficient heat is added to components containing reactor coolant to compensate for parasitic heat loss during periods where the reactor is not supplying fission heat (such as post reactor shutdowns and subsequent restarts) or extended maintenance periods where operational temperatures are desired to maintain the reactor coolant in a liquid phase. Surface heaters are used to heat the vessel and vessel head components uniformly to preclude thermal shock and thus damage to the reactor vessel system during normal operation. These design features satisfy the requirements of PDC 71.

Pipe leaks in interfacing systems with radioactive contaminants could cause contamination in the RTMS; therefore, the RTMS is designed to minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406.

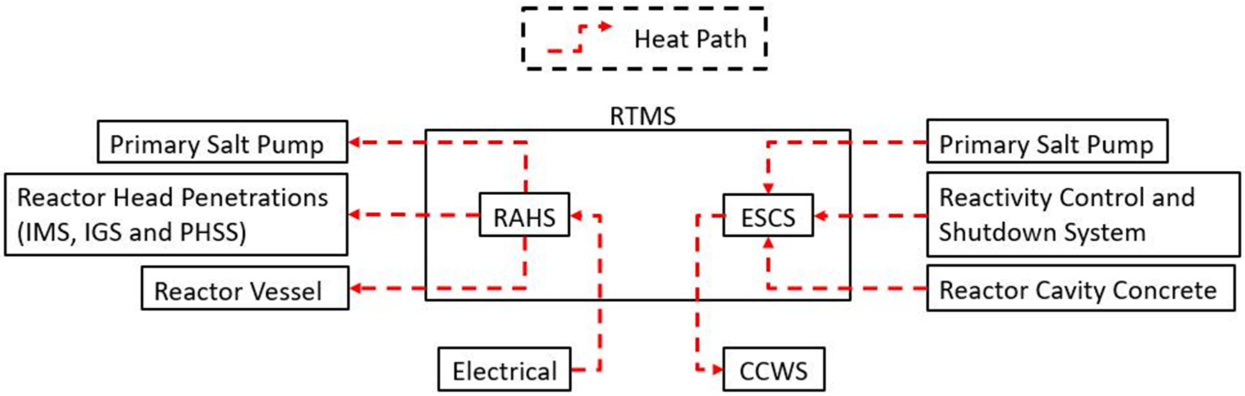
9.1.5.4 Testing and Inspection

Temperatures in and around the SSCs served by the RTMS are routinely monitored and controlled by the plant control system to maintain the desired operational limits. The components of the RTMS, including thermocouples, heaters, and other components, are located such that they are accessible for periodic inspection and testing.

9.1.5.5 References

None

Figure 9.1.5-1: Reactor Thermal Management System Interfaces



9.2 REACTOR BUILDING HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS

9.2.1 Description

The reactor building heating ventilation and air conditioning system (RBHVAC) provides independent environmental control to the Reactor Building (RB) and associated habitable spaces. In addition to directly supporting environmental control for workers in designated low hazard zones, the RBHVAC ventilation air flow and leakage is designed to be from a low hazard potential to a higher hazard potential. The RBHVAC system in the RB is independent from the ventilation systems of surrounding buildings. Each unit has its own RBHVAC and there are no components shared between the units.

The RBHVAC performs the following non-safety related functions:

- Maintain environmental conditions (air quality, temperature, humidity, pressure, and noise levels) for personnel health, habitability, and for SSC operability.
- Provide a means to control and monitor tritium, beryllium and other controlled effluents.
- Monitor exhaust air vented from the RB for controlled effluents.
- Ensure ventilation flow and leakage from areas of low hazard to areas of higher hazard potential.
- Minimize contamination of facility areas.

The system is comprised of fans, duct work, dampers, heaters, and filters that draw filtered supply air from the atmosphere and supply it to the RB. Ventilation exhaust that is discharged to the atmosphere from portions of the RB that potentially contain contaminants during normal operation is monitored and utilizes appropriate filtration, including HEPA filters.

9.2.2 Design Bases

The RBHVAC systems ensure that temperature, relative humidity and air circulation rates are within limits for personnel and equipment. The systems are also designed to ensure that normal sources of airborne radioactive material, including tritium, are controlled so that occupational doses do not exceed the requirements of 10 CFR 20. In addition, the RBHVAC system ensures that chemical hazards (such as Beryllium) are within applicable limits.

Consistent with PDC 2, Design Bases for Protection Against Natural Phenomena, safety-related SSCs located near the RBHVAC are protected from the adverse effects of RBHVAC failures during a design basis earthquake.

Consistent with PDC 60, Control of Releases of Radioactive Materials to the Environment, the RBHVAC is designed to control the release of radioactive materials in gaseous effluents during normal reactor operation.

Consistent with PDC 64, Monitoring Radioactivity Releases, the RBHVAC is designed to provide for monitoring the reactor building effluent discharge paths for radioactivity that may be released during normal operations.

Consistent with 10 CFR 20.1406, the RBHVAC is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning.

9.2.3 System Evaluation

The RBHVAC does not perform safety-related functions and is not credited for the mitigation of postulated events. The system is also not credited for performing safe shutdown functions.

Portions of the RBHVAC may be located in proximity to SSCs with safety-related functions. Those safety-related SSCs are protected from failure of the RBHVAC during a design basis earthquake by either seismically mounting the applicable RBHVAC components, physical separation, or barriers to preclude adverse interactions. Also, the RBHVAC is located in safety-related and non-safety related portions of the Reactor Building. As a result, portions of the RBHVAC may cross the isolation moat discussed in Section 3.5. SSCs that cross the base isolation moat may experience differential displacements as a result of seismic events. The RBHVAC is designed so that postulated failures of SSCs in the system from differential displacements do not preclude a safety-related SSC from performing its safety function. Design features addressing differential displacement are discussed in Section 3.5. This satisfies the requirements of PDC 2 for the RBHVAC.

The RBHVAC system is not credited for the filtration of radionuclides for minimizing dose consequences during postulated events. However, filters are provided in the discharge pathway to provide for filtration of radioactivity prior to release to the atmosphere during normal plant operations. These design features provide for the control of radioactive materials in gaseous effluents, consistent with PDC 60.

The RBHVAC discharge pathway to the atmosphere is monitored for radioactivity during normal operations and postulated events. This monitoring capability satisfies PDC 64 for the RBHVAC.

The RBHVAC system contains radiological contaminants: therefore, the system is designed to minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406 as described in Chapter 11.

9.2.4 Testing and Inspection

The system will be monitored and periodically functionally tested. RBHVAC filters will also be periodically replaced.

9.2.5 References

None

9.3 PEBBLE HANDLING AND STORAGE SYSTEM

For fuel pebbles in the PHSS the TRISO fuel particles provide a functional containment such that radionuclides are contained within the particle. The pebbles are designed to prevent damage to the TRISO fuel particles during normal operation, storage, shipping and handling thus the fuel particle is credited for confining radioactive material rather than the pebble matrix material, the handling equipment and the storage system. The fuel pebbles can experience thermal and mechanical loads while being handled, inspected, operated, and stored but such loads are within the design basis of the fuel pebble design.

9.3.1 Description

The PHSS provides for handling and storing fuel and other pebbles. The system encompasses receipt and inspection of new fuel upon delivery, core loading, sensing, inspection and sorting during downstream circulation, re-insertion, core unloading, and removal and transfer to storage.

Major components and features of the PHSS include the pebble extraction machine (PEM), debris removal, off-head conveyance line, pebble processing, pebble inspection, pebble insertion, PHSS inert gas boundary, pebble storage, and new pebble introduction. Each unit has its own PHSS and there are no components shared between the units. A process flow diagram is provided in Figure 9.3-1.

The PHSS interfaces with the reactor vessel (Section 4.3), IGS (Section 9.1.2), spent fuel cooling system (SFCS) (Section 9.8.2), and the CCWS (Section 9.7.3) as shown in Figure 9.3-2.

9.3.1.1 Pebble Extraction Machine

The PEM removes buoyant pebbles which accumulate in the reactor de-fueling chute at the top of the reactor core and routes them towards the off-head conveyance. The PEM is comprised of a single screw shaft located at the top of the reactor vessel head. As the pebbles traverse the screw, they are removed from the molten Flibe and moved into the inert gas space. The PEM also acts as a pathway for debris removal from the vessel to the debris removal portion of the system. Components in the PEM are cooled by the reactor thermal management system (see Section 9.1.5) to preclude overheating. The elevation of the PEM relative to the coolant limits coolant leaks from the reactor vessel in the event of breaks in the PEM.

9.3.1.2 Debris Removal

Pebble or graphite debris removal is accomplished by extracting Flibe primary coolant up the PEM via a pressure differential, transferring debris-carrying Flibe to a filtering tank through a debris pipe, filtering debris from the coolant in an off-head tank, and returning filtered Flibe back to the vessel through the PEM.

9.3.1.3 Off-Head Conveyance

An off-head conveyance line routes pebbles from the PEM to a buffer storage prior to the processing system, located off the reactor head as shown in Figure 9.3-2. The off-head conveyance mechanism is a downward angled chute with a diameter that is larger than the pebbles. The off-head conveyance line includes design features for removing debris or jams that could impede pebble movement. This design minimizes the risk of pebbles and debris from jamming the line, such that a geometrically safe configuration is maintained at all times.

9.3.1.4 Pebble Processing

PHSS pebble processing directs pebbles to the correct insertion channel or to a storage canister for spent fuel, based on results from the inspection system via an automated mechanism. A rotating wheel in the processing system moves pebbles from the off-head conveyance to the inspection area. After inspection, the pebbles are directed for re-insertion into the core, or to pebble storage for removal from the circulating pebble inventory, based on inspection results.

9.3.1.5 Pebble Inspection

An automated inspection system provides information to the processing portion of the PHSS for determining pebble health. This includes inspection of the physical condition of the pebble for unacceptable wear or damage, identifying moderator and fuel pebbles, as well as an evaluation of the burnup of the fuel relative to a maximum burnup limit using the burn up measurement sensor (BUMS). The burnup measurement is done by means of a gamma spectrometer. Further details pertaining to inspections for wear and damage will be provided with the application for an Operating License.

9.3.1.6 Pebble Insertion

Pebbles are received from the processing system and placed in a buffer storage until required for reinsertion. The pebble buffer storage is sized and orientated to prevent a critical configuration. Individual pebbles are fed into the step feeder insertion machine from this pebble buffer storage as shown in Figure 9.3-2. The pebbles are inserted into the top of the reactor vessel head, then pushed through the insertion line and enter the reactor core via the in-vessel fueling chute at the bottom of the core (see Section 4.3). There is a single active insertion line into the vessel and is designed with overflow protection cutouts to limit coolant loss from the reactor vessel in the event the insertion line breaks.

9.3.1.7 PHSS Inert Gas Boundary

The components of the PHSS are designed to maintain an inert gas boundary outside of the reactor vessel for pebble handling. The function of the inert gas environment is to prevent absorption of moisture and oxygen into pebbles for pebble handling during normal operations. The inert gas boundary within the PHSS (see Figure 9.3-2) is created by a mechanical structure that encloses the aforementioned components with penetrations for motor shafts, storage outlets, inspection viewport, data channels, electrical power, and pebbles from the off-head conveyance mechanism and for insertion. Portions of the inert gas boundary that are adjacent to personnel access areas have the appropriate radiation shielding.

9.3.1.8 Pebble Storage

Pebble storage is provided for pebble debris, damaged pebbles, spent fuel, and end of life moderator pebbles. The storage portion of the system is composed of a stainless steel storage canister and transporter device. Individual storage canisters are sized to hold approximately 1,900-2,100 pebbles. The dimensions of the canister and quantity of pebbles are sized to maintain a non-critical configuration. A transporter device is used to transfer canisters to either the spent fuel storage area during normal operation or the full core offload area in the event of a periodic maintenance full core offload or an emergent full core offload.

9.3.1.8.1 Spent Fuel Storage

Spent fuel is discharged from service in the core under normal operating conditions, placed in sealed storage canisters, and moved to the spent fuel storage area as shown in Figure 9.3-2. The initial storage area is a cooling pool designed to hold spent fuel canisters while the decay heat of the pebbles drops. The pool is designed to limit radiation exposure to personnel. After cooling in pool storage, the canisters are moved to a concrete storage bay with radiation shielding and forced air cooling. The pool is actively cooled by the CCWS using an in-pool heat exchanger. Water is re-circulated in the pool by the SFCS and make-up water is provided by the treated water system (see Section 9.7.2). The pool and concrete storage bay are designed to prevent a critical configuration. The storage bay sizing is sufficient to store spent fuel and moderator pebbles generated during the 11 year operating lifetime of the reactor. Criticality calculations will assume that the canister storage bay and canister interiors are flooded for conservatism. The air cooled storage bay is cooled by the SFCS. Both air-cooled and water-cooled storage areas are sized and spaced to passively cool the spent fuel and moderator pebbles under normal conditions and postulated events.

9.3.1.8.2 Fill, Sealing, and Movement

The storage canister interior is maintained in an inert environment while attached to the processing portion of the system for canister filling of pebbles and preparing the filled canisters for storage. Filling is performed by attachment of a storage canister to pebble processing via a chute and pebbles identified for storage are routed to the canister. Once the canister is filled with pebbles, the fill valve is closed and the canister is moved via an automated transfer system for sealing or welding. The canister is then moved within a canister transporter to the cooling pool for initial spent fuel storage. The canister has two seals in series to prevent accidental ingress of oxygen. The fill environment remains within the inert gas boundary in argon gas as shown in Figure 9.3-2. The sealing and welding processes are performed in a shielded and recessed concrete bay.

9.3.1.8.3 Full Core Offload

The PHSS has the capability to fully offload pebbles from the core in the event periodic maintenance requires complete removal of all the fuel within the reactor or if an emergent issue requires a full core offload. During a full core offload, the pebbles are directed to storage canisters for filling. During this process, pebbles are not sorted based on burnup level or pebble type (i.e., moderator, fuel, etc.). Once the canister is full, the fill, sealing, and movement operations are performed, and the canister fill valve is closed. The canister is not welded shut but rather sealed via a valve to allow reintroduction of the pebbles into the core.

Full core offload is functionally similar to spent fuel storage but has a different cooling demand due to the increased decay heat production rates of the removed pebbles. The canisters are stored in a pool. The pool is sized and the canister spacing is such that during a loss of power condition there is sufficient thermal mass to prevent overheating of pebbles in the storage canisters. The concrete structure surrounding the pool and storage bay as well as the support restraints in the pool holding the canisters in place are designed as seismic design category (SDC) 3 structures. The storage pool is cooled by the CCWS and is designed to ensure a subcritical configuration.

9.3.1.9 New Fuel Pebble Introduction

New fuel pebbles are received from shipment and stored in their shipping containers in a new fuel storage area until required. The new fuel pebble storage area is sized and arranged such that a subcritical geometry is maintained under all conditions.

New pebbles are moved into the preconditioning and introduction area when desired for use. Pebbles are first removed from the shipping container and placed into a new pebble canister. New pebbles are pre-conditioned by “baking” them to remove moisture and oxygen. A vacuum is also pulled to remove the contaminants from the gas space, followed by an argon purge.

The preconditioned pebbles are then inserted into the PHSS inert gas boundary, and ultimately the insertion system. The insertion point precedes the inspection system to allow for pebble inspection, if deemed necessary. This same insertion point is also used for reintroduction of pebbles after a full core offload. In the full core offload scenario, inspection and burn-up measurements are conducted to exclude pebbles that would not meet physical condition and burnup limits. The pebble introduction process is done via two sequential valves to prevent introduction of contaminants to the PHSS inert gas boundary or new pebbles. The interstitial space between the valves is purged prior to opening of either valve to limit the ingress of oxygen.

9.3.2 Design Bases

Consistent with PDC 2, the PHSS is designed to withstand the effects of natural phenomena without exceeding the offsite dose consequences of the MHA, compromising decay heat removal, or criticality as a result of a system failure or breach.

Consistent with PDC 3, the PHSS is designed and located within the facility to minimize the probability and dose consequences of fires and explosions.

Consistent with PDC 4, the PHSS is designed to accommodate environmental conditions associated with normal operation, maintenance, testing and postulated events.

Consistent with PDC 33, The PHSS is designed to limit the loss of reactor coolant from the reactor vessel due to potential breaks in the system.

Consistent with PDC 61, the PHSS is designed to permit periodic inspection and testing and is suitably shielded for radiation protection. The PHSS design includes appropriate confinement and adequately accounts for decay heat and a reduction in fuel storage cooling under postulated events.

Consistent with PDC 62, the PHSS is designed to prevent criticality.

Consistent with PDC 63, the PHSS is designed to detect conditions that may result in excessive radiation levels and initiates appropriate safety actions.

Consistent with 10 CFR 70.24(a)(1), the PHSS design includes a monitoring system capable of detecting criticality.

Consistent with 10 CFR 20, the PHSS is designed to be shielded to support worker occupational dose limits and adhere to a radiation protection program.

Consistent with 10 CFR 20.1406, the PHSS is designed, to the extent practicable, to minimize contamination of the facility and the environment, and facilitate eventual decommissioning.

9.3.3 System Evaluation

The concrete structures associated with the storage bay, pool, and support restraints in the pool are designed as SDC 3 structures to ensure the geometry of the storage area is maintained to preclude an inadvertent criticality during a design basis earthquake. The design of the support restraints and storage bay also ensures adequate spacing is maintained for air cooling between each canister. During a postulated earthquake, the fuel particles prevent radionuclide release. The particles are supported in their safety function during a postulated earthquake by the pool and by the canister transporter, both of which provide passive cooling and spacing to restrict pebble movement thereby preventing recriticality. Other portions of the PHSS that do not perform a safety function are designed to be either seismically mounted or physically separated to preclude adverse interactions with other safety-related SSCs during a design basis earthquake. These design features satisfy the requirements of PDC 2.

The PHSS is designed to minimize the probability of a fire or explosion by limiting the accumulation of potentially combustible material such as graphite dust and debris within the system. Grinding of pebbles which contribute to graphite dust generation is precluded by system design. The small amount of graphite dust that might be generated is directed through pebble motion to the storage canisters for removal from the system. The PHSS is not located near nor interfaces with pneumatic systems with the potential for air in-leakage. The system is filled with an inert gas operated at a slightly positive pressure to further prevent air ingress in the event of a PHSS breach. Locations where pebbles are not submerged in coolant, such as the PEM, will either not exceed temperatures that would induce oxidation of the graphite or are expected to cool quickly such that oxidation, if any, would be minimal and not affect the acceptability of the pebble for reuse. These design features satisfy the requirements of PDC 3 for the PHSS. Fire protection systems are further discussed in Section 9.4.

The pebble handling portion of the PHSS is protected from the effects of discharging fluids. There are no pressurized piping systems in or around the PHSS thus precluding the design from pipe whip hazards. A hypothetical water line break in the area of the storage system does not pose a criticality risk as the analyses supporting the storage system assume complete submergence and internal flooding of the storage canisters in water. The PHSS is designed in consideration of the high radiation environment where equipment will be functioning. The PHSS design also considers and accounts for the temperature within the system to preclude oxidation of graphite pebbles. The stainless steel PHSS storage canisters are designed to accommodate pressure due to the accumulation of radionuclides and thermal loads associated with the amount of spent fuel loaded in each canister during normal and postulated event conditions. The canisters are also designed to accommodate the tensile stress exerted during transfer and are compatible with handling equipment. The interior of the stainless steel canisters is also designed to account for radiolysis products from spent nuclear fuel and ensures the integrity of the canister, seal, and weld thus precluding the potential release of radionuclides from the canister. These design features demonstrate that the PHSS satisfies the environmental and dynamic effects in PDC 4.

The PHSS interfaces with the reactor vessel at the PEM and the pebble insertion line. The elevation of the PEM relative to the coolant free surface is such that coolant inventory loss from the reactor vessel is limited in the event the PEM breaks. The pebble insertion line is designed to limit inventory loss to an elevation no lower than the primary salt pump elevation, in the event of a break in the insertion line. The pebble insertion line uses overflow protection cutouts to direct any coolant in the insertion line back down into the reactor vessel. Cover gas fills the line to break the siphon. These design features of the PHSS satisfy the requirements in PDC 33.

PDC 61 requires that the safety-related portions of the PHSS that contain radioactivity be designed to ensure (1) capability to permit appropriate periodic inspection and testing of components, (2) suitable shielding for radiation protection, (3) appropriate containment, confinement, and filtering, (4) residual heat removal capability, and (5) significant reduction in fuel storage cooling under postulated event conditions is precluded. The design features which address PDC 61 for the PHSS are discussed below:

- The TRISO fuel particle provides a functional containment as described in Section 6.2. Radioactive material and fission products are contained within the particle unless the TRISO layers are compromised or defective (see Section 4.2.1). The fuel pebble, as described in Section 4.2.1, is designed to preclude physical damage or changes in geometry to the TRISO particle during anticipated loads from normal operation, storage, shipping and handling. Therefore, the TRISO particle is credited for the confinement of radioactive materials rather than the PHSS. The pebble can experience thermal and mechanical loads while being handled, inspected, operated, and stored; however, such loads do not introduce incremental failures of TRISO particles. Furthermore, the PHSS design precludes pebble damage from overheating and oxidation. Heat removal mechanisms within the system, such as thermal radiation and convection via natural circulation, are sufficient to remove the decay heat produced by individual pebbles during their transit through the PHSS. Also, oxidation associated with air or moisture ingress into the PHSS is negligible for pebbles at temperatures experienced in the system. The system also minimizes pebble wear. The limiting PHSS malfunction event, which is discussed in Section 13.1.5, does not cause temperature excursions, oxidation, or mechanical stresses on the TRISO particles. Therefore, containment and confinement of radioactivity is maintained by the TRISO particles.

- Fuel and moderator pebbles are manufactured to specifications as described in Section 4.2.1 and are “baked” prior to introduction to the reactor to remove residual moisture. After the pebbles exit the core, the inspection system, as described in Section 9.3.1.5, is used to inspect the physical condition of the pebble and measure the fuel burnup. The inspection is performed to identify abnormal wear, cracking, and missing surfaces due to pebble chipping. Gamma spectrometry is also used to determine the burnup by measuring gamma ray activity from fission products. Pebbles at or approaching the burnup limit are sent to storage in lieu of being returned to the core. Pebbles that show indications of wear, cracking, or missing surfaces are also removed from service and placed into storage.
- The PHSS is adequately shielded to limit worker dose, in accordance with 10 CFR 20 and the radiation protection program, as described in Chapter 11.
- The storage part of PHSS is designed to transfer ex-vessel decay heat to the CCWS and the SFCS from a full core offload and pebble offload due to normal operation. The PHSS is designed to ensure decay heat loads from pebbles in the spent fuel storage pool are passively cooled by the water of the pool and spacing of the storage canisters in the event of a loss of power. The canisters in the storage bay are cooled during postulated events by natural convection due to the spacing which allows sufficient air flow.

PDC 62 requires criticality in a fuel storage and handling system be prevented by physical systems or processes, preferably by use of geometrically safe configurations. The design features which address PDC 62 for the PHSS are described below:

- The PHSS is designed to preclude criticality by maintaining a subcritical geometry during handling. The PHSS removes pebbles from the core at a rate that prohibits the formation of a critical configuration of fuel pebbles outside the reactor. In the event of a PHSS line breach, the number of spilled pebbles is limited and a critical geometry is precluded by design. The off-head conveyance, processing, inspection, pebble insertion, storage areas, and inert gas boundary maintain an inert gas environment precluding moisture intrusion into those handling areas, further reducing the risk of criticality. Fuel handling equipment maintains a subcritical geometry via physical constraints and/or system interlocks.
- The spent fuel storage area consists of a water-cooled pool, an air-cooled storage bay, seismic restraints maintaining the canisters’ physical location (i.e., spacing), and the surrounding concrete structure. The preliminary criticality analysis determining the spacing requirements for each canister in the spent fuel storage area conservatively assumes the storage containers are flooded and completely submerged under water.
- The transport configuration, in which a storage canister is being moved using a canister transporter to either the storage bay or the full core off-load system (i.e., fuel pool), will be analyzed to ensure a subcritical geometry is maintained. A summary of the criticality analyses confirming the system design maintains a geometrically safe configuration will be provided with the application for an Operating License.

PDC 63 requires detection of conditions that could result in excessive radiation levels in handling areas and a means by which to initiate appropriate safety actions. The PHSS is designed to assure that mechanical and thermal loads to the fuel pebble as well as oxidation during handling, inspection, and loading into canisters do not exceed pebble design limits. Therefore, operations in the PHSS do not introduce TRISO particle failures that would result in excessive radiation levels in the handling area. The pebble inspection and sorting functions performed by the PHSS ensure that damaged pebbles removed from the reactor core are removed from use. Monitoring of the cover gas and reactor coolant radioactivity provides early indication of a potential TRISO particle failures. This satisfies the requirements of PDC 63.

The PHSS contains radiological contaminants; therefore, the system is designed to minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406.

9.3.4 Testing and Inspection

The fuel pebble inspection portion of the system is periodically calibrated to provide assurance that limits on the physical condition and burnup of the pebbles to be reinserted into the core are within specified Technical Specification limits. Temperature of the spent fuel storage pool, water level of the spent fuel pool, and air temperature and flow in the storage bay are also monitored to confirm adequate cooling of storage canisters. The plant control system (see Section 7.2) is capable of shutting down the system such that additional pebbles do not enter the PHSS line upon a PHSS line breach. Criticality monitoring alarms throughout the PHSS are tested periodically to confirm functionality.

9.3.5 References

None

Figure 9.3-1: Process Flow Diagram for the Pebble Handling and Storage System

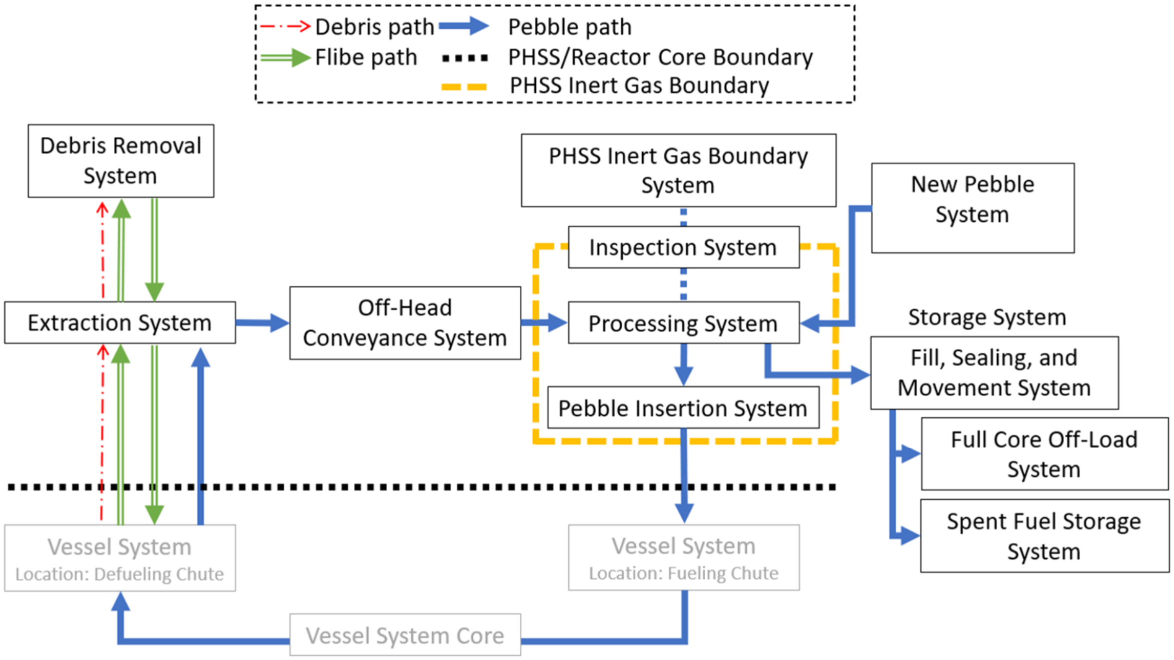
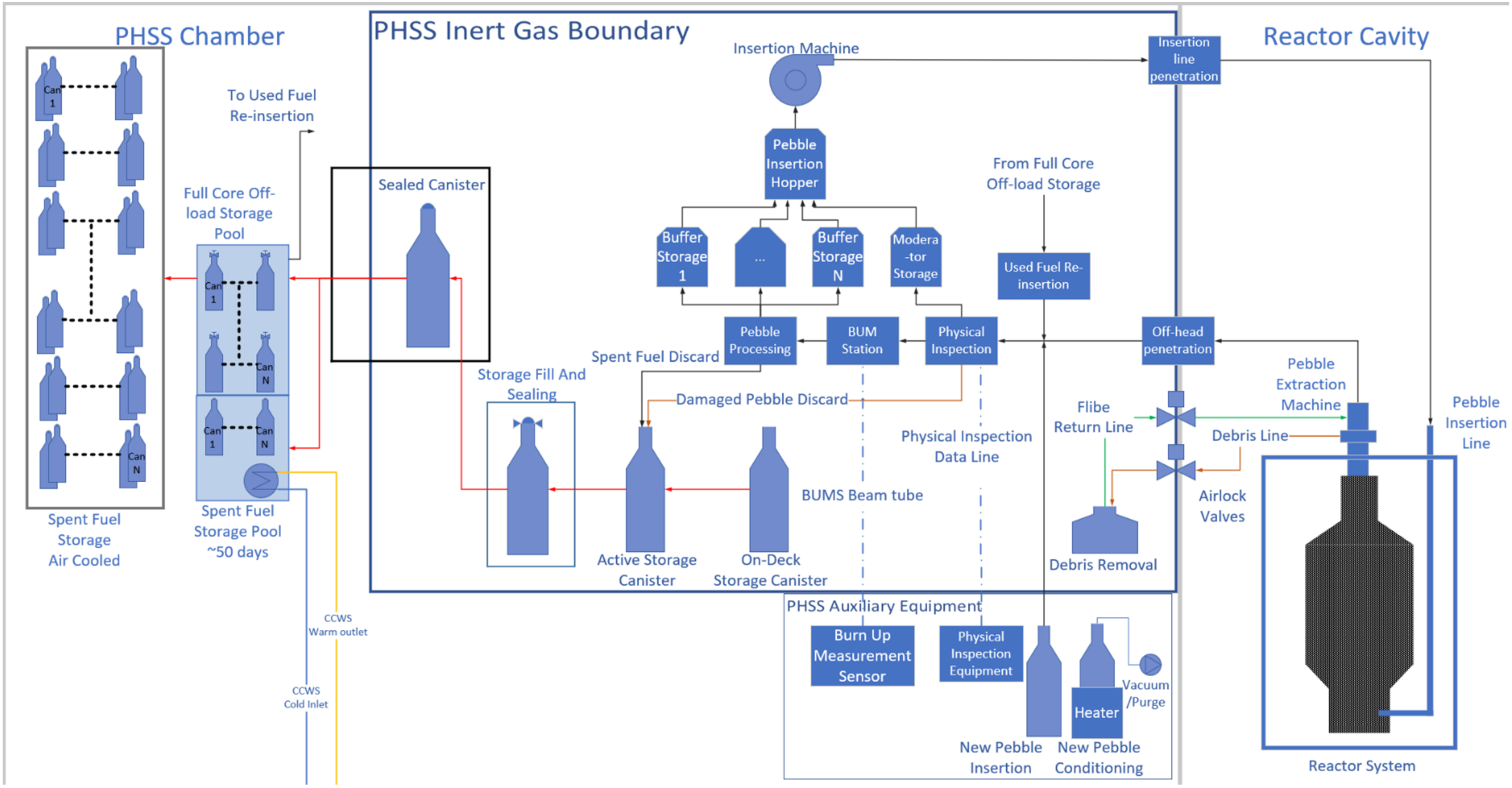


Figure 9.3-2: Pebble Handling and Storage System



9.4 FIRE PROTECTION SYSTEMS AND PROGRAMS

9.4.1 Fire Protection Program

There is one common fire protection program for the Hermes 2 site. A description of the fire protection program and a fire hazards analysis will be provided with the application for an Operating License. The fire protection program addresses those SSCs that could affect safety or the protection of licensed materials.

9.4.2 Fire Protection Systems

9.4.2.1 Description

The Hermes 2 fire protection systems consist of unit-specific systems that serve each reactor building and common systems that serve the shared turbine building and the shared main control room. The fire protection systems are 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. This is accomplished in part by limiting the types and quantities of combustible materials present. The fire protection systems do not perform safety-related functions.

The fire protection systems include fire detection and alarm systems as well as automatic and manual fire suppression systems. Design features such as fire barriers and fire area penetration protection are also included to provide for life safety provisions and to minimize the spread of fire. Fire protection system elements and design features will be identified and installed in defined fire areas based on the results of the fire hazards analysis. Where required by the fire hazards analysis, automatic fire detectors will be installed. Manual pull stations will also be installed to allow personnel to activate the fire protection system.

9.4.2.2 Design Bases

Consistent with PDC 2, safety-related SSCs located near fire protection systems are protected from the adverse effects of fire protection system failures during a design basis earthquake.

Consistent with PDC 3, noncombustible and fire-resistant materials are used whenever practical, particularly in locations with SSCs that are safety-related or required for safe shutdown. Fire detection and fighting systems of appropriate capacity and capability are provided and designed to minimize the adverse effects of fires on these SSCs, and firefighting systems are designed to ensure that their rupture or inadvertent operation does not significantly impair the safety capability of these SSCs.

9.4.2.3 System Evaluation

The regulations in 10 CFR 50.48(a)(1) require a fire protection plan to satisfy Criterion 3 of 10 CFR 50, Appendix A. While this criterion does not apply to non-light water reactors, a similar design criterion (PDC 3) has been established as described in Section 3.1. The details of the fire protection program plan will be provided with the application for an Operating License.

The fire protection systems conform to local building and fire codes. Additionally, the fire protection systems will be designed to ANSI/ANS 15.17, "Fire Protection Program Criteria for Research Reactors" (Reference 1) and NFPA 801 (Reference 2). Life safety provisions are included in the facility design in accordance with the Life Safety Code, NFPA 101 (Reference 3).

Safety-related SSCs and equipment required for safe shutdown of the reactor are located within the reactor buildings. The floors, walls, and ceilings of the reactor buildings are constructed almost entirely of reinforced concrete. Fire detection and suppression capability provided by the fire protection system minimizes the potential for adverse effects of fires on safety-related SSCs and those required for safe shutdown. Fire water piping is routed such that a rupture or inadvertent operation of the fire protection system does not significantly impair safety-related or safe shutdown functions. These design features, in conjunction with the fire protection program, satisfy PDC 3.

The fire protection systems are not safety-related but portions of these systems may cross the isolation moats discussed in Section 3.5. SSCs that cross a base isolation moat may experience differential displacements as a result of seismic events. The fire protection systems are designed so that postulated failures of SSCs in these systems from differential displacements do not preclude a safety-related SSC from performing its safety function. Design features addressing differential displacement are discussed in Section 3.5. These features address conformance with PDC 2.

9.4.2.4 Testing and Inspection

Functional tests of the fire protection systems will be performed prior to startup and periodic functional tests and inspections of the system will be performed during facility operation.

9.4.2.5 References

1. American National Standards Institute/American Nuclear Society, ANSI/ANS 15.17, "Fire Protection Program Criteria for Research Reactors," ANS, LaGrange Park, Illinois, 1981.
2. National Fire Protection Association, NFPA 801, "Standard for Fire Protection for Facilities Handling Radioactive Materials," 2020.
3. National Fire Protection Association, NFPA 101, “Life Safety Code,” 2021.

9.5 COMMUNICATION

9.5.1 Description

The communication system provides communications during normal and emergency conditions between essential areas of the facility, as well as locations remote to the facility. The communication system is not safety-related; it is not credited for mitigation of design basis events and has no safe shutdown function. The system is designed such that a failure of a subsystem does not impair the ability of the other subsystems to function. These diverse communications systems are independent of each other to provide effective communications. The system is capable of providing communications between the reactor operator and the shift supervisor and radiological protection staff on duty, during the full range of reactor operations. The system is also capable of announcing an emergency condition across the site. The communication systems are common systems shared between Unit 1 and Unit 2. The diverse technologies are described in the sections below.

9.5.2 Normal and Emergency Communication

The communication system provides normal and emergency communication capabilities and is comprised of the following subsystems:

- Plant radio
- Public address and general alarm
- Communication capability in the event of a loss of normal power
- Distributed antenna
- Security communications

The facility uses a communication system that provides for alarming and party-line-type voice communications and communications broadcasting. The communication system uses diverse voice over internet protocol and commercial land and cellular phone lines in combination, in the control room and several other locations within and outside the reactor building. The communication system provides communications between key areas of the facility. The communication system is designed so that a failure of any one station does not impact the other stations. In an emergency, the public address system is used to alert personnel. The details for the communication systems to provide provisions for summoning emergency assistance from designated personnel are discussed in the physical security and emergency plans as appropriate.

9.5.3 Off-Site Communication

Commercial land and cellular telephone lines are provided in normally occupied plant areas. These phone lines allow personnel to contact any outside telephone number in the case of an emergency. In the event of a postulated event or security event, offsite communication, including with the NRC, emergency responders, or local law enforcement, is addressed in the physical security plan and emergency plan discussed in Chapter 12.

9.5.4 Testing and Inspection

The diverse implementation of the communications systems permits routine testing and inspection without disruption to normal communications. Testing on the communication system is used to detect and correct any problems or degradation.

9.5.5 References

None

9.6 POSSESSION AND USE OF BYPRODUCT, SOURCE, AND SPECIAL NUCLEAR MATERIAL

Special nuclear material (SNM), source, and byproduct material will be present at the reactor facility. The applicable requirements in 10 CFR Part 30, 10 CFR Part 40, and 10 CFR Part 70 may 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 SNM, source, 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 as low as reasonably achievable (ALARA) objectives. Additional information on ALARA practices is discussed in Chapter 11.

Spaces in which the materials are handled and equipment used to handle the material, are subject to 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, source, and special nuclear material will be provided in the application for an Operating License.

Waste from SNM, source material, or byproduct material is handled through the radioactive waste management program described in Section 11.2.1. The radioactive waste handling system also handles drains and vents for the facility including handling of contaminated liquids collected by the drain system (see Section 11.2.2).

9.6.1 Special Nuclear Material

SNM is received and used at the facility in the form of fresh fuel particles contained in pebbles (see Section 9.3). Fuel pebbles containing SNM use high assay, low enriched uranium (less than 20% enrichment) at different enrichment levels.

SNM is handled in the fuel intake area, the PHSS, and the reactor vessel. In the intake area, SNM is managed by compliance with 10 CFR Part 70, by the use of fresh fuel canisters and by the nature of the pebble design, in which the SNM is encapsulated in a graphite substrate. Section 9.3 and Chapter 4 describe how the PHSS and the reactor vessel, respectively, prevent uncontrolled releases of radioactive material. Of the systems described in this paragraph, only the fresh fuel handling areas have the potential for direct contact with the SNM during normal operation. At this location, the activity of the fresh fuel is very low and administrative procedures that minimize contact with the fresh fuel are sufficient in support of ALARA practices. In the PHSS, spent fuel is handled in canisters and shielding is used to support ALARA practices.

9.6.2 Source Material

Source material that contains unenriched uranium is also received and used at the facility in the form of unenriched fuel particles contained in fuel pebbles. Handling of fuel pebbles containing source material is within the same systems as the pebbles that contain SNM. Source material is managed by compliance with 10 CFR Part 40, by use of fresh and spent fuel canisters, and by the nature of the pebble design, in which the source material is encapsulated in a graphite substrate.

9.6.3 Byproduct Material

Byproduct materials are both used in and generated to support operation of the KP-FHR.

Tritium is generated at the facility and is classified as byproduct material. Tritium is generated as a result of the nuclear reaction in the core. Tritium is present throughout the primary system and in the graphite core of fuel pebbles. Because the pebbles travel through the PHSS, tritium will be present in PHSS as well. The tritium management system manages the inventory of tritium in the reactor system.

Byproduct material is managed by compliance with 10 CFR Part 30, by use of spent fuel canisters, by the tritium management system, and by the radioactive waste management program (see Chapter 11). A description of how the tritium management system prevents the uncontrolled release of tritium can be found in Section 9.1.3. That section also describes how tritium is removed from the facility and discusses administrative procedures to minimize exposure to tritium from the disposal of tritium capture materials, in support of ALARA practices.

9.6.4 Laboratories

Auxiliary services, described in Section 9.8.5, include laboratories under the reactor operating license in which licensed material will be used. Offsite laboratories (if any) are not governed by this facility license. Byproduct, source, and special nuclear material may be handled in the laboratories associated with the auxiliary services for the site under the license(s) described above. Laboratory work involving byproduct, source, and special nuclear material is for research and testing purposes. Materials are handled appropriately (e.g., in glove boxes, as appropriate) in those laboratories so that 10 CFR Part 20 dose limits are not exceeded and consistent with ALARA practices. Airborne materials are handled through air exhaust systems, as applicable, and radioactive waste material is managed through the radioactive waste management program.

9.7 PLANT WATER SYSTEMS

Plant water systems that take water into the site, treat it, and distribute it for cooling and maintenance activities are discussed in this section. Figure 9.7-1 shows the plant water systems and their interfaces. The figure also shows which of the systems perform a cooling function. None of the water systems are safety-related or are credited for the mitigation of postulated events. Portions of the plant water systems are shared between Unit 1 and Unit 2 as discussed in the subsections below.

The portions of the plant water systems that directly interface with systems that contain radioactive material, could potentially become contaminated. The plant water systems that directly interface with the systems that contain radioactive material are designed to meet the requirements of 10 CFR 20.1406, “Minimization of Contamination,” to minimize to the extent practicable contamination of the facility and the environment, facilitate eventual decommissioning, and minimize to the extent practicable the generation of radioactive waste. The design of the radioactive waste handling system (see Chapter 11) is sufficient to accommodate potential leaks from those portions of the plant water systems that could become contaminated.

9.7.1 Service Water System

The service water system serves as the main supply of water for the facility. The service water system is a common system shared between Unit 1 and Unit 2. The system receives and stores water and distributes it to site services. The service water supply comes from municipal sources. A portion of the water from the service water system is provided to the treated water system discussed in Section 9.7.2. The service water system is comprised of piping from the municipal supply, storage tanks, filters, pumps, and distribution piping. The system is designed in accordance with local building codes. The service water system is a non-safety related system and is not credited for the mitigation of postulated events. The system will not be located in the proximity of safety-related SSCs.

9.7.2 Treated Water System

The treated water system provides chemistry control of water supplied from the Service Water System (Section 9.7.1) and provides make-up water to the component cooling water system (CCWS) (Section 9.7.3), the chilled water system (CWS) (Section 9.7.4), the decay heat removal system (Section 6.2), and the power generation system deaerator (Section 9.9.3). The system is comprised of supply piping, pumps, filters, storage tanks, demineralization exchangers and tanks, demineralization support components, and distribution piping. The system is designed in accordance with local building codes. Portions of the treated water system are shared between Unit 1 and Unit 2. This includes portions of the treated water system piping that distributes water to the component cooling water system and chilled water system for Unit 1 and Unit 2.

The treated water system is not a safety-related system and is not credited for the mitigation of postulated events. The treated water system is also not credited with performing safe shutdown functions.

Portions of the treated water system may be located in proximity to SSCs with safety-related functions. Those safety-related SSCs are protected from failure of the treated water system during a design basis earthquake by either seismically mounting the applicable treated water system components, physical separation, or barriers to preclude adverse interactions, consistent with PDC 2. Nearby safety-related SSCs are also protected from the effects of discharging fluid and missiles by design. There are also no pressurized piping systems in or around the treated water system, thus precluding the potential for adverse effects from pipe whip hazards, consistent with PDC 4.

The treated water system is not safety-related but portions of the system may cross the base-isolation moat discussed in Section 3.5. SSCs that cross a base-isolation moat may experience differential displacements as a result of seismic events. The treated water system is designed so that postulated failures of SSCs in the system from differential displacements do not preclude a safety-related SSC from performing its safety function. Design features addressing differential displacement are discussed in Section 3.5. These features demonstrate conformance with the requirements in PDC 2.

9.7.3 Component Cooling Water System

The component cooling water system (CCWS) provides water cooling for RBHVAC system (see Section 9.2), ESCS (see Section 9.1.5.1.1), SFCS (see Section 9.8.2) and the IGS coolers and compressors. The system consists of heat exchangers, pumps, and piping that remove heat, as needed, to maintain desired operational temperatures. Temperatures in and around the SSCs served by the CCWS are routinely monitored and controlled by the plant control system (PCS) (see Section 7.2) to maintain the desired operational limits. Heat from the CCWS is rejected to the environment. Each unit has its own CCWS and there are no components shared between the units.

The CCWS does not perform safety-related functions and is not credited for the mitigation of postulated events. The CCWS is also not credited with performing safe shutdown functions.

Portions of the CCWS may be located in proximity to SSCs with safety-related functions. Those safety-related SSCs are protected from failure of the CCWS during a design basis earthquake by either seismically mounting the applicable CCWS components, physical separation, or barriers to preclude adverse interactions, consistent with PDC 2.

The CCWS is not safety-related but portions of the system may cross the seismic base-isolation moat discussed in Section 3.5. SSCs that cross a base-isolation moat may experience differential displacements as a result of seismic events. The CCWS is designed so that postulated failures of SSCs in the system from differential displacements do not preclude a safety-related SSC from performing its safety function. Design features addressing differential displacement are discussed in Section 3.5. These features address conformance with PDC 2.

Nearby safety-related SSCs will also be protected from the effects of discharging fluid and missiles by design. There are no pressurized piping systems in or around the CCWS thus precluding the design from pipe whip hazards, consistent with PDC 4.

The CCWS is not safety-related, however, any portion of the CCWS that crosses the moat will include flexible design features to accommodate maximum design displacements from postulated seismic events. The design features’ function would be to prevent the damage from the SSCs in the CCWS from affecting a safety-related SSC's ability to perform its safety function. The design features include one or more of the following: flexible features for piping, elevation of SSCs above floor level, spray and shielding, water diversion features, drains, and isolation valves. Specific design features and the SSCs to which they are applied, will be provided in the application for an Operating License.

The CCWS interfaces with RBHVAC, ESCS, SFCS and IGS to transfer heat from SSCs to the environment under normal operating conditions. The CCWS is designed with the capability to isolate leaks to minimize the potential for an adverse effects of internal flooding. The system is also designed to permit appropriate periodic inspection and testing to ensure the integrity and capability of the system to cool SSCs and to adequately transfer heat to the ultimate heat sink. This satisfies the requirements of PDC 44, PDC 45 and PDC 46.

9.7.4 Chilled Water System

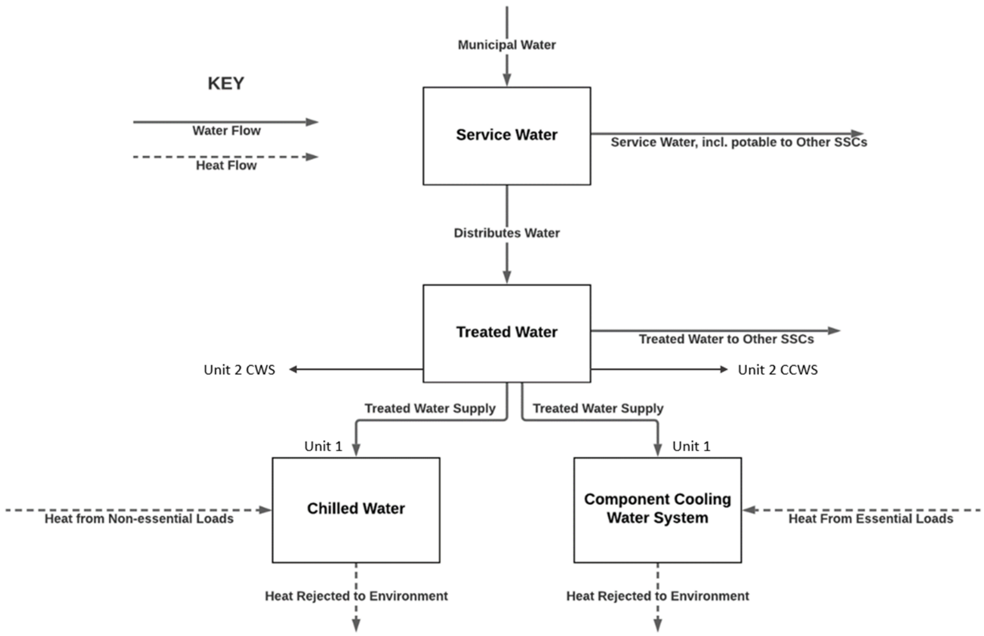
The chilled water system provides cooling water to facility SSCs that are not safety-related. The system also provides cooling water to portions of the RBHVAC (see Section 9.2). The closed-loop system receives make-up water from the treated water system (see Section 9.7.2). The system is comprised of supply piping, heat exchangers, storage tanks, and distribution piping. The system is designed in accordance with local building codes.

The chilled water system is not a safety-related system and is not credited for the mitigation of postulated events or to perform a safe shutdown function for the reactor. The chilled water system is not located in proximity of safety-related SSCs. Each unit has its own chilled water system and there are no components shared between the units.

9.7.5 References

None

Figure 9.7-1: Plant Water System Process Flow Diagram



In this figure, “essential loads” means the heat from systems that perform or support safety-related SSCs or SSCs that may contain hazardous contaminants during normal operations.

9.8 OTHER AUXILIARY SYSTEMS

The following subsections provide descriptions and functional requirements of other auxiliary systems. These other auxiliary systems include:

- Remote maintenance and inspection system
- Spent fuel cooling system
- Compressed air system
- Cranes and rigging
- Auxiliary site services

These auxiliary systems are not safety-related nor are they credited with performing a safety function.

9.8.1 Remote Maintenance and Inspection System

The remote maintenance and inspection system (RMIS) provides the capability to remotely handle components in the reactor system, PHTS, and PHSS. The system also provides the capability to conduct inspections of hazardous equipment. Components of the RMIS include remote manipulators, tooling, cameras, monitors, cranes and rigging. Each unit has its own RMIS and there are no components shared between the units. The system is located in the reactor building and contains tooling to support the following maintenance activities:

- Disassemble flanges and subassemblies
- Remove subassemblies
- Clear fuel and residual coolant before removal of SSCs for maintenance
- Transport of equipment to hot maintenance cells (via use of shielded casks)
- Activities performed in standalone hot cells
- Use of through-wall electro-mechanical manipulators for hot cells
- Use of cranes for hot cell and post-irradiation examination facilities.

The system is designed in accordance with local building codes. The system does not perform safety-related functions and is designed so that it cannot interfere with a safety system's ability to perform a safety function. The remote manipulation capabilities provided by the system facilitate limiting personnel occupational exposures to below 10 CFR Part 20 limits during maintenance of the reactor system, PHTS, and PHSS.

Consistent with 10 CFR 20.1406, the remote maintenance and inspection system is designed, to the extent practicable, to minimize contamination of the facility and the environment, and to facilitate eventual decommissioning.

Portions of the RMIS that may cross the isolation moat include flexible design features to accommodate maximum design displacements from postulated seismic events. The design features' function would be to prevent the damage from the SSCs in the RMIS from affecting a safety-related SSC's ability to perform a safety function. Specific design features and the SSCs to which they are applied, will be provided in the operating license application.

9.8.2 Spent Fuel Cooling System

The SFCS provides forced air cooling for spent fuel storage canisters in the storage bay of the PHSS (see Section 9.3) and recirculates water in the spent fuel pool. The system is sized to cool stored spent fuel and moderator pebbles generated during the 11 year lifetime of the reactor. The SFCS consists of fans and piping that remove heat during normal operation, to maintain desired operational temperatures in the storage bay. Temperatures in and around the SSCs served by the SFCS, including the storage canisters, will be routinely monitored and controlled by the PCS (see Section 7.2) to maintain the desired operational limits. Heat from the SFCS is rejected to the environment. In the event that normal power is not available, the SFCS is capable of passively cooling spent fuel storage canisters. Each unit has its own SFCS and there are no components shared between the units.

The SFCS does not perform safety-related functions and is not credited for the mitigation of postulated events. The SFCS is also not credited with performing safe shutdown functions.

Portions of the SFCS may be located in proximity to SSCs with safety-related functions. Those safety-related SSCs are protected from failure of the SFCS during a design basis earthquake by either seismically mounting the applicable SFCS components, physical separation, or barriers to preclude adverse interactions, consistent with PDC 2. Nearby safety-related SSCs are also protected from the effects of missiles by design. There are also no pressurized piping systems in or around the SFCS thus precluding the design from pipe whip hazards, consistent with PDC 4.

The system has the potential to become contaminated based on its location and system interfaces. Therefore, the system is designed to meet the requirements of 10 CFR 20.1406 to minimize to the extent practicable contamination of the facility and the environment, facilitate eventual decommission and minimize to the extent practicable, the generation of radioactive waste.

9.8.3 Compressed Air System

The compressed air system provides and distributes compressed air for maintenance and use in valve operation. The system includes distribution piping, valves, compressors, coolers, moisture separators, filters, and receivers. The system does not provide compressed air that is credited to perform safety-related functions. The system is designed so that a failure of the system does not interfere or preclude the ability of a safety-related system to perform its safety function. The system does not directly interface with systems that contain or have the potential to contain radioactive materials. Each unit has its own compressed air system and there are no components shared between the units.

9.8.4 Cranes and Rigging

9.8.4.1 Description

A crane and rigging are provided to lift and move equipment within the reactor building and to facilitate equipment and material receiving and shipping. The crane and rigging are also provided to support maintenance activities, including lifting activities with the potential to damage safety-related SSCs in the event of a load drop. The crane and rigging equipment do not perform a safety-related function and are not safety-related. The crane is a gantry crane located in the high bay of the reactor building. Each unit has its own reactor building cranes and rigging and there are no components shared between the units.

9.8.4.2 Design Bases

Consistent with PDC 2, safety-related SSCs located near crane and rigging are protected from the adverse effects of crane and rigging failures during a design basis earthquake.

Consistent with PDC 4, the crane and rigging are designed to protect against the dynamic effects potentially created by the failure of the crane and rigging equipment.

9.8.4.3 System Evaluation

Portions of the crane and rigging may be located in proximity to SSCs with safety-related functions. Those safety-related SSCs will be protected from failure of the crane and rigging during a design basis earthquake by either seismically mounting the applicable crane and rigging components, physical separation, or barriers to preclude adverse interactions. This satisfies the requirements of PDC 2 for the crane and rigging.

The crane and rigging will be designed so that a failure of the lifting device does not interfere or preclude the ability of a safety-related system to perform a safety function. The crane design implements ASME B30.2-2016 (Reference 1). When the crane is used to move spent fuel transportation casks, administrative controls and interlocks maintain cask lift elevations within allowable areas to preclude impacts to safety-related SSCs. Also, administrative controls and interlocks prevent the crane and rigging from moving heavy loads over safety-related SSCs except when the reactor is shut down, or the consequences of a load drop have been evaluated to ensure that it could neither damage stored irradiated fuel to the extent that a significant off-site release would occur, nor preclude operation of sufficient equipment to achieve safe shutdown. These administrative controls ensure that a dropped load does not interfere with or preclude a safety-related SSC's ability to perform its function during operation, which addresses the potential for dynamic effects under PDC 4.

The crane superstructure is designed to remain standing during and after a fire so that failure of the superstructure does not interfere or preclude the ability of a safety-related system to perform its safety function. Further information about the design of the superstructure in the event of a fire will be provided in the operating license application.

The crane is supported by the non-safety portion of the reactor building, which is designed for seismic loads in accordance with local building codes as described in Section 3.5. No parts of the crane supports are on the portion of the reactor building that uses base isolation, i.e., the safety-related portion of the building.

9.8.4.4 Testing and Inspection

Cranes and rigging will be periodically inspected prior to use.

9.8.5 Auxiliary Site Services

Auxiliary site services encompass supportive non-safety related SSCs that provide additional functions necessary to maintain and operate the facility. The services are not credited to mitigate any postulated events. These services include:

- Machine shop(s), which include radioactive and non-radioactive machining capabilities
- Chemistry laboratory
- Post-irradiation examination laboratory
- Materials testing laboratory
- Vents, drains for non-potentially contaminated facility compartments
- Warehouse(s) for storage of spare equipment
- Storage of contaminated equipment
- Facility lighting, including emergency lighting
- Non-hazardous waste management services
- Firewater storage systems
- Storm and sanitary sewers
- Groundwater monitoring wells

These auxiliary site services are designed in accordance with local building code and relevant permits. The services are designed so that they do not interfere with a safety-related SSC's ability to perform its safety function. Portions of the auxiliary site services may be located in proximity to SSCs with safety-related functions. Those safety-related SSCs are protected from failure of the auxiliary site services 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.

Services that involve handling of radioactive material may include remote manipulation capabilities, as appropriate, to facilitate limiting personnel occupational exposures to below 10 CFR Part 20 limits.

Some site services are shared between Unit 1 and Unit 2. Warehouses, fire water storage, facility lighting, storm and sanitary sewers, and ground water monitoring wells are expected to be shared across the site.

9.8.6 References

1. ASME B30.2-2016, Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist), New York, NY. 2016.

9.9 POWER GENERATION SYSTEM

Power generation includes the following systems:

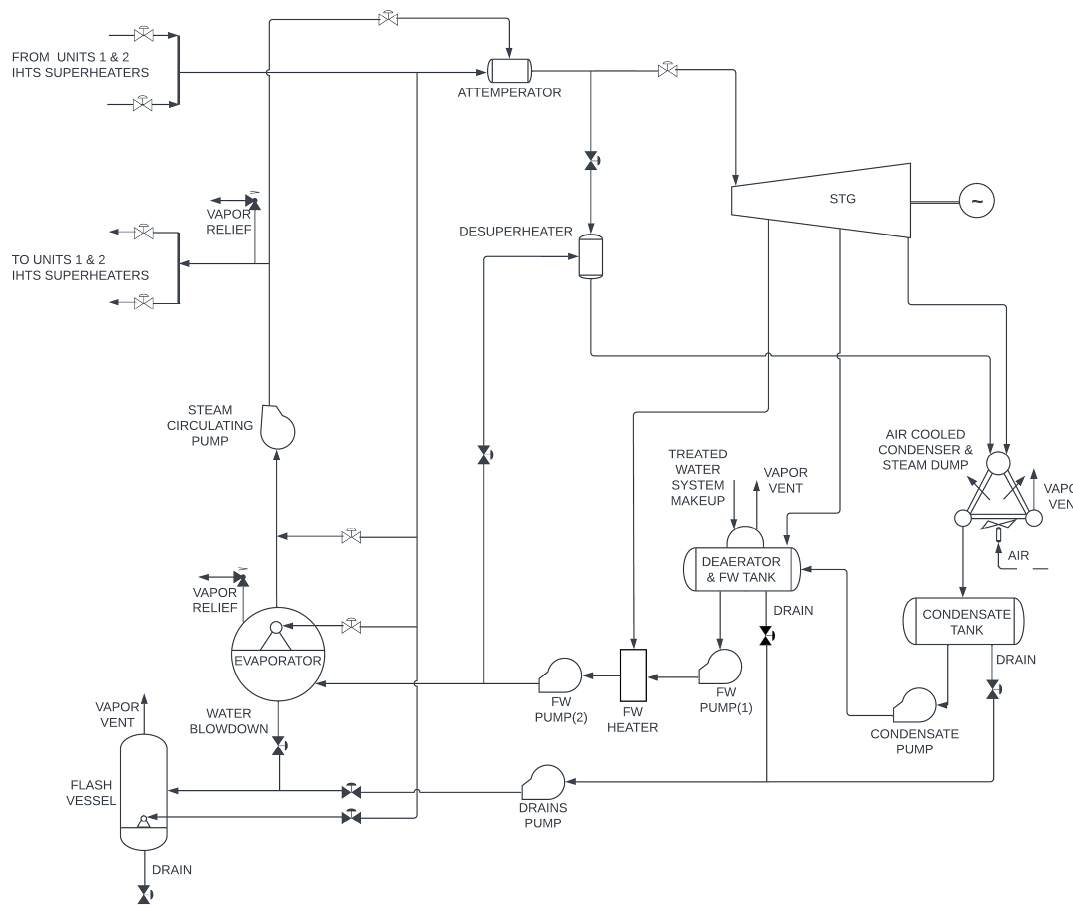
- Steam system
- Turbine generator system
- Feedwater and condensate system

The steam system receives heat energy from the intermediate heat transport system. The superheated steam is directed to the steam turbine generator and converted to electrical power. Steam from the turbine exhaust is condensed by ambient air in the condenser and returned to the evaporator by the feedwater system. The power generation system does not perform any safety-related functions. The majority of the power generation system are shared between Unit 1 and Unit 2. Key parameters for the power generation system at nominal full power are provided in Table 9.9-1. A process flow diagram of the power generation system is provided in Figure 9.9-1.

Table 9.9-1 Key Design Power Generation System Parameters

Parameter	Value
Reactor thermal power (two units)	70 MWth
Superheater steam flow	88 kg/s
Superheater steam pressure	14 MPa
Superheater inlet steam temperature	337°C
Superheater outlet steam temperature	540°C
Feedwater temperature	69-224°C
Evaporator pressure	14 MPa
Generator output	~20 MW

Figure 9.9-1 Power Generation System



9.9.1 Steam System

The primary function of the steam system is to utilize the heat from the intermediate heat transport system to generate steam for the turbine generator system, as shown in Figure 9.9-1. The steam system utilizes a regenerative direct-contact evaporator to produce saturated steam by mixing feedwater with a portion of recirculated superheated steam. A small portion of the recirculated superheated steam is injected prior to the steam circulating pump to reduce moisture content to the superheater inlet header. The steam pump directs the saturated steam to the superheaters in each unit's IHTS.

The superheater uses heat from the intermediate heat transport system to superheat the saturated steam, as discussed in Section 5.2. The unit specific superheater outlet steam lines combine the output of the superheaters from each unit into a common header. From the superheater outlet header, the steam system directs about thirty percent of the superheated steam to the turbine generator system and recirculates about seventy percent of the superheated steam. The recirculated superheated steam is used to evaporate feedwater, reduce moisture carry-over from the evaporator, and evaporate excess liquid discharge in the flash vessel.

The turbine generator includes steam extraction lines which are directed to the deaerator and feedwater heater in the feedwater system to raise feedwater temperature prior to entry into the evaporator.

The evaporator is equipped with a blowdown line to remove liquid contaminants. The water blowdown from the evaporator, along with the drains of the feedwater and condensate system, is directed to a flash vessel. The flash vessel uses recirculated superheated steam to evaporate the remaining liquid discharge.

The steam system includes piping, pumps, valves, vessels, drains, and traps and is designed to industry codes and standards, as noted in Table 3.6-2. Unit-specific main steam isolation valves are provided upstream of the superheater outlet header to isolate the steam supply from each unit, as needed, to support single unit operation. The steam system is designed to handle a turbine trip without a corresponding reactor trip via the turbine bypass line and the condenser (which is sized to handle 100% steam load), and the steam relief valves (which have the capability to reject 100% load to the atmosphere). These features are not safety-related.

9.9.1.1 Design Bases

Consistent with PDC 4, nearby safety-related systems are protected against dynamic effects associated with high-pressure steam system pipe leaks and breaks.

Consistent with PDC 60, the design of the steam system supports the control of radioactive materials release during normal reactor operations.

Consistent with PDC 64, the steam system is designed to monitor radioactive releases.

Consistent with 10 CFR 20.1406, the steam system is designed to minimize contamination of the facility and the environment and facilitate eventual decommissioning.

9.9.1.2 System Evaluation

Portions of the steam line piping are located in the non-safety related portion of the reactor building and in the turbine building. Piping is not located near safety-related SSCs, such that postulated steam line leaks or breaks do not adversely affect the ability to perform safety functions. These design features satisfy the requirements of PDC 4.

Tritium produced in the reactor migrates through the PHTS and IHTS to the steam system. The release of tritium is controlled by collecting liquid discharge to an appropriately sized flash vessel. The flash vessel will evaporate the liquid discharge via a vapor vent to the atmosphere. These design features satisfy the requirements of PDC 60.

The steam system includes radiation monitors on the evaporator, the inlet header prior to the superheater, and the flash vessel vapor vents which monitor tritium releases to the atmosphere during normal operations. These design features satisfy the requirements of PDC 64.

The steam system contains radiological contaminants. Therefore, the design of the system will minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406, as described in Chapter 11.

9.9.1.3 Testing and Inspection

The steam system does not perform any safety-related functions or support safe shutdown of the reactor. The steam system piping and components are periodically inspected for investment protection.

9.9.2 Turbine Generator System

The primary function of the turbine generator system is to convert steam into electricity. One turbine generator building and one turbine generator system are shared between the two reactor units. The turbine generator system interfaces with both the steam system and the condensate and feedwater system. Figure 9.9-1 depicts the turbine generator system.

The commercially available steam turbine generator system converts thermal energy into mechanical energy. The turbine includes rotors, stop valves, control valves, drains, vents, and turning gear. Prior to entering the turbine generator, the superheated steam is attemperated with a portion of saturated steam. The attemperator controls the turbine inlet steam temperature according to turbine manufacturer guidelines.

Turbine stop valves and control valves are the primary methods of steam flow control for normal operations and postulated events. The turbine control valves throttle steam through the turbine in normal operations. A turbine bypass line and associated valve is provided for the superheated steam to directly route to the air-cooled condenser. The turbine bypass steam passes through a desuperheater to lower the temperature prior to entering the air-cooled condenser.

The steam turbine is coupled with the generator. The generator converts the rotational mechanical energy from the turbine into electricity by rotating a magnetic field. The frequency is synchronized between the generator and offsite transmission to transfer electricity to the grid. A portion of the output of the turbine generator is also connected (via a stepdown transformer) to the normal power system and provides power to plant electrical loads, as described in Section 8.2. The turbine control system interfaces with the power generation control system, as discussed in Section 7.2, and provides control of the turbine generator system from the main control room. The turbine generator is equipped with protective monitoring, which will trip the turbine for investment protection in the event of off-normal turbine generator conditions. The turbine generator system is designed to handle a reactor trip on either unit without a corresponding turbine trip. In the event of a reactor trip on one unit, the turbine control system partially closes the turbine control valves to accommodate the reduced steam flow while generating less than 50% electrical load. Residual heat from the tripped reactor is removed by the decay heat removal system, as discussed in Section 6.3. In the event of a reactor trip on both units, the turbine is automatically tripped, and residual heat is removed by the decay heat removal system, as discussed in Section 6.3.

9.9.2.1 Design Bases

Consistent with PDC 4, nearby safety-related systems are protected against dynamic affects associated with postulated turbine missiles.

Consistent with 10 CFR 20.1406, the turbine generator system is designed to minimize contamination of the facility and the environment and facilitate eventual decommissioning.

9.9.2.2 System Evaluation

The turbine generator is located in the turbine building away from safety-related SSCs. The turbine generator is favorably oriented with respect to the reactor building and safety-related SSCs such that in the event of a postulated turbine missile, it would not be able to strike a safety-related SSC or impair its safety functions. The turbine generator is designed with protective monitoring from overspeed events to minimize the probability for turbine missile generation. When the shaft speed indicator exceeds acceptable limits, the turbine control system signals the control valve to throttle and preclude overspeed. These design features satisfy the requirements of PDC 4.The turbine generator system contains radiological contaminants. Therefore, the design of the system will minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406, as described in Chapter 11.

9.9.2.3 Testing and Inspection

The turbine generator system does not perform any safety-related functions or support safe shutdown of the reactor. The turbine generator system is tested and inspected as recommended by the manufacturer.

9.9.3 Condensate and Feedwater System

The condensate and feedwater system returns condensed steam from the air-cooled condenser to the condensate tank, deaerates and reheats the water to feedwater temperature and pressure, and supplies feedwater to the evaporator, as shown on Figure 9.9-1.

The air-cooled condenser uses ambient air to cool and condense the turbine exhaust. The steam ejector draws a partial vacuum on the turbine exhaust through the air-cooled condenser.

The condensate is pumped from the condensate tank to the deaerator and feedwater tank. Steam extraction from the steam system is used for deaeration and feedwater heating. Chemically treated water is added to the deaerator.

The feedwater is pumped through a direct contact regenerative feedwater heater prior to the evaporator. The feedwater heater uses turbine steam extraction to heat the feedwater. The feedwater pumps increase the pressure to meet the individual equipment pressure.

The condensate and feedwater system is designed for the efficiency of the steam cycle and is not safety-related. The condensate and feedwater system is of conventional design, including piping, isolation valves, regulation valves, deaerator, tanks, pumps, heaters, drains, and vents and is designed to industry codes and standards, as noted in Table 3.6-2.The condensate and feedwater system is designed to support a turbine trip without a corresponding reactor trip. This is accomplished via the turbine bypass lines and the air-cooled condenser (which is sized to handle 100% load from the turbine bypass line), and the feedwater control valves (which can control the amount water sent to the steam generators to match steam production). A portion of the feedwater is mixed with the turbine bypass superheated steam in the desuperheater to reduce steam temperature prior to the condenser. These features are not safety-related.

9.9.3.1 Design Bases

Consistent with PDC 4, nearby safety-related systems are protected against dynamic affects associated with postulated high-pressure leaks and breaks in the condensate and feedwater system.

Consistent with PDC 60, the design of the steam system supports the control of radioactive materials during normal reactor operations.

Consistent with PDC 64, the condensate and feedwater system is designed to monitor radioactive releases.

Consistent with 10 CFR 20.1406, the condensate and feedwater system is designed to minimize contamination of the facility and the environment and facilitate eventual decommissioning.

9.9.3.2 System Evaluation

The condensate and feedwater system piping is located in the non-safety related turbine building away from safety-related SSCs such that postulated line leaks or breaks do not adversely affect the ability to perform safety functions. These design features satisfy the requirements of PDC 4.

Tritium produced in the reactor migrates through the PHTS and IHTS to the condensate and feedwater system. The release of tritium is controlled by collecting liquid discharges from the condensate and feedwater drains to the flash vessel in the steam system. These design features satisfy the requirements of PDC 60.

The condensate and feedwater system includes radiation monitors on the air-cooled condenser and deaerator vapor vents which monitor tritium releases to the atmosphere during normal operations. These design features satisfy the requirements of PDC 64.

The condensate and feedwater system contains radiological contaminants. Therefore, the design of the system will minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406, as described in Chapter 11.

9.9.3.3 Testing and Inspection

The condensate and feedwater system does not perform any safety-related functions or support safe shutdown of the reactor. The feedwater and condensate system piping and components are periodically inspected.

9.9.4 References

None

Chapter 10

Experimental Facilities and Utilization

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

None

List of Figures

None

CHAPTER 10 EXPERIMENTAL FACILITIES AND UTILIZATION

10.1 SUMMARY DESCRIPTION

The test reactor is constructed to demonstrate the application of fluoride salt-cooled, high-temperature technology for a two-unit facility using nuclear heat for electrical power production. The facility does not include special facilities dedicated to the conduct of reactor experiments or experimental programs. The design allows for performance of startup physics testing and to conduct maneuvering operations at various power levels to assess plant performance and capabilities. The startup testing plan is described in Section 12.11. The design of the plant systems provides process monitoring capability as described in prior sections of this report. Specifically, the design of the pebble handling and storage system includes process monitoring capability that is used to assess the fuel pebble performance during normal operations and transients. These features are described in Section 9.3. Similarly, the reactor vessel is equipped with a material surveillance system to insert and remove material specimens to assess long term material performance. These features are described in Section 4.3.

Kairos Power

Chapter 11

Radiation Protection Program and Waste Management

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 11 RADIATION PROTECTION AND WASTE MANAGEMENT

11.1 RADIATION PROTECTION

The radiation protection programs described in the following subsections identify the sources of radiation and describe the program elements for radiation protection. The program descriptions included with this preliminary safety analysis report (PSAR) are appropriately at a high level and identify the use of existing approved NRC regulatory guides (RGs) for demonstrating conformance to the requirements in 10 CFR 20. The identification of the expected radiation sources along with the commitments to existing approved guidance provide reasonable assurance, consistent with 10 CFR 50.40(a), that the regulations in 10 CFR 20 will be met, and that the health and safety of the public will not be endangered. Additional details of these programs will be provided with the application of an Operating License as noted in the subsections below.

11.1.1 Radiation Sources

The sources of radiation that present a potential hazard to workers and the public in the facility result from fission in the fuel (fission products and decay products) and neutron activation products (including tritium) generated as a result of exposure to neutrons. Fission products generated in the TRISO fuel may leak into the Flibe as a result of manufacturing defects in the TRISO layers (see Section 4.2.1). Fission products also may be generated from potential uranium impurity in the reactor coolant. Activation products are located in the coolant, cover gas, and structures, and are the result of neutron activation of various isotopes, and corrosion and wear products.

Table 11.1-1 lists the radiation sources.

Additional details of the radiation sources, including activity and external radiation fields for the facility which demonstrate compliance with 10 CFR 20, Subparts C and D, will be provided with the application for an Operating License consistent with 10 CFR 50.34(b)(3).

11.1.2 Radiation Protection Program

A radiation protection program is required by 10 CFR 20.1101. The radiation protection program will comply with the regulatory requirements in 10 CFR 19 and 10 CFR 20, and will be developed, documented, and implemented commensurate with the scope and extent of licensed activities for a test reactor facility.

As required by 10 CFR 20.1101(c), the program content and implementation will be reviewed periodically. Procedures and engineering controls will be employed, to the extent practical, to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA), as required by 10 CFR 20.1101(b).

In accordance with 10 CFR 20.1101(d), there will be a constraint on air emissions of radioactive material to the environment, with consideration of the guidance provided in RG 4.20, “Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees Other than Power Reactors,” Revision 1. In addition, dose rates in unrestricted areas will be controlled to remain below the limits set forth in 10 CFR 20.1302 for individual members of the public.

The radiation protection program will be designed and implemented consistent with the following guidance:

- RG 8.2, “Administrative Practices in Radiation Surveys and Monitoring,” Revision 1
- RG 8.13, “Instruction Concerning Prenatal Radiation Exposure,” Revision 3
- RG 8.29, “Instruction Concerning Risks from Occupational Radiation Exposure,” Revision 1

The preliminary organizational structure for the facility is described in Section 12.1 and includes provisions for Radiation Protection functions. The radiation protection training program will be designed and implemented in accordance with the requirements of 10 CFR 19.12. Recordkeeping will be conducted in accordance with 10 CFR 20, Subpart L.

Additional details of the radiation protection program for the facility, including organization and staffing levels, authorities and responsibilities, position qualifications, personnel training requirements, and document control and recordkeeping procedures, will be provided with the application for an Operating License consistent with 10 CFR 50.34(b)(6).

11.1.3 ALARA Program

A program to ensure occupational doses and doses to members of the public are ALARA is required by 10 CFR 20.1101. An ALARA program will be implemented consistent with the guidance in RG 8.10, “Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable,” Revision 2. The Radiation Protection function described in Section 12.1 will be responsible for the ALARA program.

Additional details of the ALARA program for the facility will be provided with the application for an Operating License consistent with 10 CFR 50.34(b)(3).

11.1.4 Radiation Monitoring and Surveying

The requirements for radiation monitoring and surveys are outlined in 10 CFR 20, Subpart F. The purpose of radiation monitoring and surveys is to (1) determine radiation levels, concentrations of radioactive materials, and potential radiological hazards that could be present in the facility, and (2) detect releases of radioactive material from facility equipment and operations. Radiation surveys will focus on those areas of the facility where the occupational radiation dose limits could potentially be exceeded. Measurements of airborne radioactive material and/or bioassays will be used to determine that internal occupational exposures to radiation do not exceed the dose limits specified in 10 CFR 20, Subpart C, "Occupational Dose Limits." Written procedures will be established to ensure compliance with the requirements of 10 CFR 20, Subpart F, "Surveys and Monitoring."

The radiation survey and monitoring programs will consider the guidance provided in the following regulatory guides:

- 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
- RG 8.34, “Monitoring Criteria and Methods to Calculate Occupational Radiation Doses,” Draft Revision 1

Additional details of radiation monitoring and surveying, including a description of the equipment, methods, and procedures will be provided with the application for an Operating License consistent with 10 CFR 50.34(b)(3).

11.1.5 Radiation Exposure Control and Dosimetry

A summary of the controls for exposure and access control are provided below. Additional details of the dosimetry and radiation exposure control for the facility, including the locations of radiological control areas, access controls, shielding, remote handling equipment, and expected annual radiation exposures, will be provided with the application for an Operating License consistent with 10 CFR 50.34(b)(3).

Effluent Monitoring

Facility effluents are monitored for radioactivity during normal operations and postulated events, and structures, systems, and components (SSCs) are designed to limit uncontrolled liquid or gaseous effluent releases to work areas or the environment, consistent with the goal of maintaining radiation exposures ALARA. Releases during postulated events are evaluated in Chapter 13.

During normal operations, liquid radioactive waste is expected to be packaged and disposed of using a licensed and qualified low-level radioactive waste disposal vendor.

The Reactor Building heating, ventilation, and air conditioning (RBHVAC) system (see Section 9.2) provides for gaseous effluent monitoring and filtration, after which gaseous effluents are generally released to the atmosphere. Other potential gaseous effluent release points include the heat rejection stack (see Section 5.1), the power generation system evaporator, flash vessel, deaerator, and condenser vent pipes (see Section 9.9), and the spent fuel cooling system stack (see Section 9.3).

A screening analysis of the radionuclide emissions from the reactor was performed for the Hermes facility using the NRC's XOQDOQ and GASPAR II models (Reference 1). XOQDOQ is designed to calculate the annual relative effluent concentrations and deposition due to routine releases. XOQDOQ evaluates the impacts at radial downwind distances as well as at sensitive locations specified by the user. GASPAR II is an air release radiation dose code that models the gaseous effluent pathway using the release model described in Regulatory Guide 1.109. GASPAR II requires input of released source terms (curies per year), atmospheric dispersion from the XOQDOQ model and surrounding demographics. The code was developed to analyze airborne effluents from light-water-cooled reactors during routine operations. GASPAR II considers such pathways as inhalation, plume-immersion, ground-shine, and ingestion of various contaminated media (meat, milk, vegetation, etc.). Dose calculations can be applied to a defined population or an individual using dose conversation factors from the International Commission on Radiological Protection (ICRP). Each calculation considers multiple organs (including but not limited to bone, gastrointestinal tract, kidney, liver, lung, skin, and thyroid) as well as the whole-body dose.

Site-specific, validated meteorological data covering a 5-year period of record from January 1, 2016 through December 31, 2020 from Tower L was used to quantitatively evaluate routine-releases at the facility. The meteorological data needed for the X/Q and D/Q calculations in XOQDOQ included wind speed, wind direction, and atmospheric stability as joint frequency distributions.

Tritium is expected to be the dominant routine radionuclide release. The gaseous effluent release was modeled under normal operations from a stack with a high-energy plume (>0.4MW) including a bounding tritium emissions rate conservatively modeled as the tritium generation rate of 62,500 Curies per year per unit or 125,000 Curies per year for the facility as well as a bounding spectrum of other radionuclides (Reference 2). This bounding tritium emissions rate does not evaluate the anticipated retention of tritium from the reactor units and engineered systems. These systems will reduce the effective tritium effluent rate. The power generation system deaerator and condenser vent pipes are expected to release only a small fraction of the total tritium generated. The heat rejection stack, Reactor Building heating, ventilation, and air conditioning system (RBHVAC), and power generation system flash vessel and evaporator, are considered radionuclide release pathways and would each have vent heat emission rates greater than 0.4 MW. Stacks with high-energy plumes will have 100-foot release heights. A high energy single stack model bounds these release pathways. The TEDEs from gaseous effluents for both units are summed with the TEDEs of the Hermes facility (Reference 1) and reported as a site total.

TEDEs from gaseous effluents from the facility and the combined TEDEs from gaseous effluents from all reactors on the entire site were calculated for two locations: the location of the maximally exposed individual (MEI) in an unrestricted area and an analytical nearest resident. The MEI in an unrestricted area dose location was calculated to be a location accessible to the public 0.5 miles to the south-southeast of the reactor with a combined site TEDE of 4.2 mrem/yr and a Hermes 2 facility TEDE of 2.8 mrem/yr. A combined site total analytical nearest residence dose, including ingestion pathways, of 3.6 mrem/yr was calculated located 1.1 miles east of the reactor. The Hermes 2 facility analytical nearest resident dose, including ingestion pathways, was calculated to be 2.4 mrem/yr. These two dose calculations are conservative because the radioactivity per cubic meter per radioactivity released per second (χ/Q) estimates for each reactor will not coincide due to differing locations for each release point. The calculation of analytical nearest resident dose is additionally conservative for two reasons: a) the direction analyzed (east) is different than the direction of the actual nearest resident (north-northwest), and b) the analyzed location exists inside of an industrial park, the East Tennessee Technology Park (ETTP), where future residences are not expected to be located. The analytical resident dose also included the ingestion pathway assuming consumption of meat and vegetables cultivated at the analyzed location. The milk ingestion dose pathway was not incorporated as no dairy production was identified in the area. Incorporating the ingestion dose pathways for this distance and direction is conservative because the analytical nearest resident is located inside an industrial park where there is also no identified garden or livestock production. The MEI location doses did not include an ingestion pathway because this location is within the ETTP and are not evaluated as residences. Effluent analysis corresponding to the detailed design will be discussed in the application for an Operating License.

Access Control and Shielding

Radiological control areas will be established to protect against undue risks from exposure to radiation and radioactive materials, and access to high and very high radiation areas will be controlled as required by 10 CFR 20, Subpart G. Precautionary procedures will be employed in the facility consistent with the requirements in 10 CFR 20, Subpart J.

Shielding and/or remote handling equipment is provided for worker protection from high radiation areas.

11.1.6 Contamination Control

SSCs with the potential to contain/handle radiological materials include design considerations to limit leakage and control the spread of contamination and to facilitate eventual decommissioning consistent with the requirements in with 10 CFR 20.1406. Such design features consider the guidance in RG 4.21, “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning,” Revision 0 and include the following considerations and administrative controls:

- Minimize the potential for leaks and spills to prevent the spread of contamination
- Leakage detection capabilities
- Minimize the potential of release of contamination from undetected leaks
- Measures to reduce the need to decontaminate SSCs

- Periodic review of operational practices

A description of the design features for the control of radioactive contamination for the facility, including consideration of RG 4.21, will be provided with the application for an Operating License consistent with 10 CFR 50.34(b)(3).

11.1.7 Environmental Monitoring

The regulations in 10 CFR 20.1302 requires surveys of radiation levels in unrestricted areas and radioactive materials in effluents to demonstrate compliance with the dose limits for individual members of the public. To meet these requirements, an operational radiological environmental monitoring program (REMP) will be established. The facility is located on a prior U.S. Department of Energy (DOE) nuclear facility site and the radiological conditions in the area are well characterized and establish a baseline prior to facility operation. The operational REMP will consider the guidance in RG 4.1, "Radiological Environmental Monitoring for Nuclear Power Plants," Revision 2, Sections C.2 and C.3, for establishing and conducting the environmental monitoring program used during plant operations. RG 4.1 refers to NUREG-1301, "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors" for additional guidance on the effluent and environmental monitoring. Both RG 4.1 and NUREG-1301 are written for nuclear power plants rather than test reactors. However, there are similarities in airborne releases of radioactivity such that the guidance in RG 4.1 and NUREG-1301 are considered generally relevant (with consideration of the KP-FHR technology and as appropriate for a test reactor) for developing the operational REMP .

The REMP will be implemented coincident with the start of plant operational activities. As such, the description of the environmental monitoring program, including consideration of RG 4.1 and NUREG 1301, will be provided with the application for an Operating License consistent with 10 CFR 50.34(b)(3).

11.1.8 References

1. Kairos Power LLC, "Preliminary Safety Analysis Report for the Kairos Power Fluoride Salt-Cooled, High Temperature Non-Power Reactor (Hermes)," Revision 3, May 2023.
2. Clinch River Nuclear Site, Early Site Permit Application, Revision 2. March 2019.

Table 11.1-1: Radiation Sources

Description	Contents
Reactor Vessel and Internals	Flibe, Fuel and Moderator Pebbles, Startup Source, Circulating Activity, Tritium, Activated Structures and Components: Graphite Reflector; Stainless Steel Vessel, Internals, and Head Components
Primary Heat Transport System (PHTS)	Flibe, Activated Structures and Components inside the Reactor Cavity, Circulating Activity, Tritium, Fluorine Activation Products
Pebble Handling and Storage System (PHSS)	Fuel and Moderator Pebbles, Pebble Wear Products, Pebble Fragments, Activated Structures and Components inside the Reactor Cavity
Intermediate Heat Transport System (IHTS)	Intermediate Coolant, Tritium
Power Generation Systems (PGS)	Water, Tritium
Inert Gas System (IGS)	Circulating Activity, Tritium, Filters, Activated Solid Deposits
Inventory Management System (IMS)	Flibe, Circulating Activity, Tritium, Fluorine Activation Products
Tritium Management System (TMS)	Tritium
Chemistry Control System (CCS)	Flibe, Circulating Activity, Tritium, Fluorine Activation Products
Decay Heat Removal System (DHRS)	Activated Structures and Components inside the Reactor Cavity
Liquid Radioactive Waste Handling	Liquid waste and Residual Solids
Solid Radioactive Waste Handling System	Filters from RBHVAC, IGS, and CCS; and dry active waste (DAW)
RBHVAC System	Filters, Tritium
Maintenance Hot Shop	DAW, Contaminated/Activated Components Undergoing Maintenance

11.2 RADIOACTIVE WASTE MANAGEMENT

11.2.1 Radioactive Waste Management Program

A description of the radioactive waste management program for the facility, including organization and staffing levels, authorities and responsibilities, position qualifications, personnel training requirements, and document control and recordkeeping procedures, will be provided with the application for an Operating License consistent with 10 CFR 50.34(b)(6). The preliminary organizational structure for the facility is described in Section 12.1.

11.2.2 Radioactive Waste Handling Systems and Controls

11.2.2.1 Description

The radioactive waste handling systems provide for the collection, packaging, storing, and dispositioning of low-level radioactive wastes in solid, and liquid forms. The relevant functions include:

- Decontamination
- Vent and drain
- Liquid radioactive waste handling
- Solid radioactive waste handling

There is no anticipated need for a gaseous radioactive waste system. Gaseous radioactive wastes are discharged to the Reactor Building heating, ventilation, and air conditioning (RBHVAC) system, described in Section 9.2, where they pass through a high efficiency particulate air (HEPA) filter and are monitored prior to release. See Section 11.1.1 for sources of gaseous radioactive waste.

Components removed or replaced during maintenance are radiologically and chemically decontaminated. Cleaning materials are used to remove component contamination. Contaminated liquids are collected by drains and directed to a liquid radioactive waste holdup tank.

Vents and drains provide for the collection of liquid wastes from decontamination and system leakage, and for venting systems during some filling and draining operations. Waste collected by vents and drains is held up in the radioactive waste holdup tanks.

Liquid radioactive waste handling includes components such as piping, pumps, tanks, filters, and valves to provide for the collection, storage, monitoring, and processing of liquid radioactive waste produced from normal reactor operations and maintenance. Liquid radioactive waste sources handled during operations and maintenance include those from vents, drains, and decontamination. Liquid radioactive waste may be recycled or released in accordance with applicable regulations. A portion of liquid radioactive waste is expected to be packaged and disposed of using a licensed and qualified low-level radioactive waste disposal vendor.

The solid radioactive waste system provides for the collection, processing, packaging, and storage of wet and dry solid radioactive waste produced from normal reactor operations and maintenance. Solid wastes include filters from the RBHVAC, the inert gas system (IGS), and the chemistry control system (CCS); IGS oxygen and moisture absorbers; and dry active waste (DAW). A solid waste compactor may be used to increase the density of some solid waste for ultimate disposal. Solid waste is disposed of using a licensed and qualified low-level radioactive waste disposal vendor. This system is not responsible for managing spent fuel waste.

11.2.2.2 Design Bases

Consistent with principal design criterion (PDC) 2, the radioactive waste handling systems are designed to prevent damage to SSCs during seismic and other external events.

Consistent with PDC 60, the radioactive waste handling systems are designed to control the release of radioactive materials in gaseous and liquid effluents with sufficient holdup capacity, and to handle radioactive solid wastes produced during normal reactor operation.

Consistent with PDC 63, the radioactive waste handling systems are equipped with appropriate systems to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions.

The radioactive waste handling systems are designed to meet the requirements of 10 CFR 20.1406 as it relates to the minimization of contamination and eventual decommissioning of the facility.

11.2.2.3 System Evaluation

The radioactive waste handling systems are not credited to perform a safety function to mitigate postulated events and are not relied on to achieve safe shutdown of the reactor. Releases of radioactive materials to the environment from the radioactive waste handling systems are controlled such that they do not exceed the limits of 10 CFR 20. A description of the radioactive waste handling systems design to conform to PDC 60 will be provided with the application for an Operating License.

The radioactive waste handling systems are not safety-related, but portions of these systems may cross the isolation moat discussed in Section 3.5. SSCs that cross a base-isolation moat may experience differential displacements as a result of seismic events. The radioactive waste handling systems are designed so that postulated failures of SSCs in the system from differential displacements do not preclude a safety-related SSC from performing its safety function. Design features addressing differential displacement are discussed in Section 3.5. These features address conformance with PDC 2.

The radiation monitoring system provides for monitoring of the radioactive waste handling systems to satisfy PDC 63.

The radioactive waste handling systems contains radiological contaminants; therefore, the systems are designed to minimize contamination and support eventual decommissioning, consistent with the requirements of 10 CFR 20.1406 as described above.

11.2.2.4 Testing and Inspection

The radioactive waste handling systems are not safety-related and will be periodically tested for functionality to support facility operations.

11.2.3 Release of Radioactive Waste

As discussed in Section 11.2.2, gaseous radioactive wastes are filtered and monitored prior to release. Liquid radioactive waste may be recycled or released in accordance with applicable regulations. Some liquid and solid radioactive waste is expected to be packaged and disposed of using a licensed and qualified low-level radioactive waste disposal vendor.

A description of the radioactive effluents from the facility, including points of effluent release and effluent monitoring equipment, will be provided with the application for an Operating License consistent with 10 CFR 50.34(b)(3).

11.2.4 References

None

Chapter 12

Conduct of Operations

Hermes 2 Test Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 12 CONDUCT OF OPERATIONS

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 (RP) program, 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 plant operations is shown in Figure 12.1-1.

12.1.2 Responsibility

Kairos Power LLC (Kairos Power) is the entity with legal responsibility for holding the Construction Permit and the facility Operating License.

The responsibilities for the key functional positions in the organizational structure are described in the following subsections, consistent with Figure 12.1-1.

12.1.2.1 Chief Executive Officer

The Chief Executive Officer (CEO) is responsible for the overall management and leadership of the company. The CEO provides direction to the Site Executive regarding company business and strategic objectives.

12.1.2.2 Site Executive

The Site Executive is responsible for compliance with the Operating License and overall management and leadership of the facility. The Site Executive provides direction to the Plant Manager regarding plant business and plant strategic testing and performance objectives.

12.1.2.3 Plant Manager

The Plant Manager (PM) is responsible for all aspects of the facility operations, including the protection of personnel from radiation exposure resulting from site operations and materials, and for compliance with applicable NRC regulations and the facility license. The PM is also responsible for establishing and managing the required training programs to support the operations organization. The PM is the final certification authority for individuals qualifying for Senior Operator or Operator status. The PM reports to the Site Executive.

12.1.2.4 Technical Services Manager

The Technical Services Manager is responsible for aspects of the facility services, including pre-op/startup testing, radiation protection, chemistry, security, emergency planning, regulatory affairs, supply chain, and document management.

12.1.2.5 Shift Supervisors

The Shift Supervisor (SS) is responsible for the safe operation of the reactor and maintains a Senior Operator license. The SS authorizes day-to-day site activities, including: control of access to the facility, work within the facility, decisions to start or shutdown equipment, and directing abnormal or emergency actions, including notifications. After facility commissioning, and until facility decommissioning, an SS is stationed at the site. The SS reports to the PM or designated alternate.

The SS authorizes work in several ways, which may include approving daily plans, work permits, and execution of specific operations procedures. Activities are approved based on the site's readiness to safely execute those activities.

12.1.2.6 Senior Operators and Operators

Senior Operators and Operators are responsible for conforming to applicable rules, regulations, and procedures for operation of the facility. Senior Operators accept responsibility for safe and efficient operation of a portion of the facility when designated by the SS. Senior Operators and Operators are responsible for maintaining Senior Operator and Operator status, respectively.

12.1.2.7 Quality Manager

The Quality Manager (QM) reports to the Site Executive and has responsibilities as described below. The QM is responsible for auditing for compliance with regulatory requirements and procedures through assessments and technical reviews, monitoring organizational processes to ensure conformance to commitments, and licensing document requirements. The QM has sufficient independence from other priorities to bring forward issues affecting safety and quality. The QM has the ability and responsibility to report to the CEO any quality issues that cannot be resolved at the Site Executive or PM level.

12.1.2.8 Radiation Protection

Radiation Protection reports to the Technical Services Manager and is responsible for establishing and implementing the RP program and the as low as reasonably achievable (ALARA) program, monitoring worker doses, and calibration of health physics instrumentation. Radiation Protection has the authority to terminate unsafe activities. Management could subsequently overrule following appropriate analysis and consideration the Radiation Protection termination of an activity.

12.1.3 Staffing

Staffing may be shared to support each of the licensed reactors on the site. Sufficient resources are provided in personnel and materials to safely conduct plant operations. Specific staffing considerations, minimum staffing levels, allocation of control functions, overtime restrictions, shift turnover, procedures, training, and availability of Senior Operators during routine operations will be provided in the application for an Operating License, consistent with 10 CFR 50.34(b)(6)(i).

12.1.4 Selection and Training of Personnel

An indoctrination and training program is maintained for personnel performing, verifying, or managing facility operation activities. ANSI/ANS 15.4-2016, "American National Standard for the Selection and Training of Personnel for Research Reactors" (Reference 1) is used in the selection and training of personnel as applicable. Records of personnel training and qualification are maintained.

A description of the training program and the required minimum qualifications for facility staff will be provided in the application for an Operating License, consistent with 10 CFR 50.34(b)(6)(i).

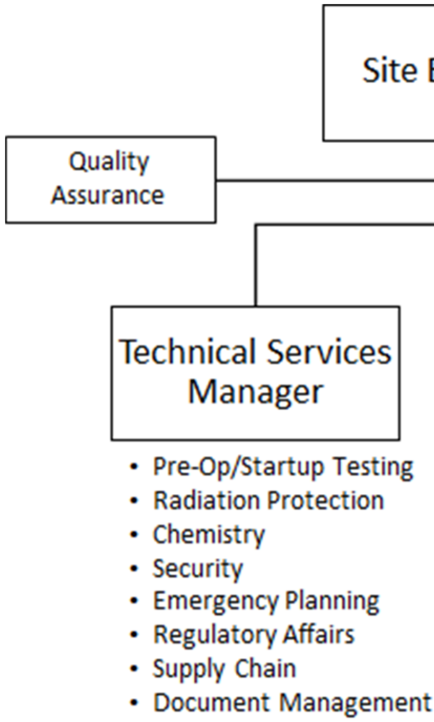
The licensed operator training program, including the requalification training program, is addressed in Section 12.10.

12.1.5 Radiation Safety

Sufficient resources in terms of staffing and equipment are provided to implement an effective RP program. Further details related to the authority of the RP program staff with respect to facility operations will be provided in the application for an Operating License application, consistent with 10 CFR 50.34(b)(6)(i).

The RP program is described in Section 11.1.2.

Figure 12.1-1: Organizational Structure



12.2 REVIEW AND AUDIT ACTIVITIES

The Site Executive establishes the Review and Audit Committee and ensure that the appropriate technical expertise will be available for review and audit activities. Committee activities are summarized and reported to the Site Executive. The details of review and audit activities and who holds the approval authority and how it communicates and interacts with facility and corporate management will be provided in the application for an Operating License, consistent with 10 CFR 50.34(b)(6)(ii).

12.3 PROCEDURES

Operating procedures provide appropriate direction to ensure that the facility is operated normally and within the design basis and technical specification limits. Activities affecting safety are performed in accordance with approved implementing 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.

Technical specifications require procedures for the following topics consistent with Section 6.4 of ANSI/ANS 15.1-2007, “The Development of Technical Specifications for Research Reactors” (Reference 2):

- Startup, operation, and shutdown of 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 regulatory guidance; procedures include management commitment and programs to maintain exposures and releases ALARA in accordance with applicable guidance
- Administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect nuclear safety
- Implementation of required plans (e.g., emergency, security)

A description of the facility procedures, including the review, approval, and changes processes, will be provided with the application for an Operating License, consistent with 10 CFR 50.34(b)(6)(vi).

12.4 REQUIRED ACTIONS

Technical specifications specify the actions be taken when a Safety Limit is exceeded; or a Limiting Condition for Operation (LCO) or its associated Surveillance Requirement (SR) is not met. Technical specifications are described in Chapter 14 and will be provided with the application for an Operating License, consistent with 10 CFR 50.34(b)(6)(vi).

12.5 REPORTS

Technical specifications specify the required routine operating reports and reporting requirements for changes to the facility or facility organization to be provided to the NRC. Technical specifications are described in Chapter 14 and will be provided with the application for an Operating License, consistent with 10 CFR 50.34(b)(6)(vi).

12.6 RECORDS

The records management program defines the process for managing test reactor facility records. The records management program includes the identification, generation, authentication, maintenance, and disposition of records. The records management program is implemented as part of the Quality Assurance Program described in Section 12.9.

The technical specifications will specify the required records to be maintained, where and how they are maintained and the length of retention for the facility. Technical specifications are described in Chapter 14 and will be provided with the application for an Operating License, consistent with 10 CFR 50.34(b)(6)(vi).

12.7 EMERGENCY PLANNING

In accordance with 10 CFR 50.34(a)(10), the specific information required of a PSAR in Appendix E.II, a description of the plans for addressing emergencies is provided in Appendix A of this chapter. The emergency plan will be updated with the application for an Operating License, consistent with the requirements in 10 CFR 50.34(b)(6)(v). The emergency plan will consider the guidance provided in ANSI/ANS 15.16-2015, "Emergency Planning for Research Reactors" (Reference 3), RG 2.6, "Emergency Planning for Research and Test Reactors", Revision 2, and NUREG-0849, "Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors".

12.8 SECURITY

A description of the security plan for the facility will be provided with the application for an Operating License consistent with 10 CFR 50.34(c) and will consider the guidance provided in RG 5.59, "Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance", Revision 1.

12.9 QUALITY ASSURANCE

The Quality Assurance Program Description (QAPD) for the design, construction, and operation of the Hermes reactor is based on ANSI/ANS 15.8–1995 (R2005), "Quality Assurance Program Requirements for Research Reactors" (Reference 4) and considers the guidance from RG 2.5, "Quality Assurance Program Requirements for Research and Test Reactors", Revision 1. The QAPD is provided as Appendix B to this Chapter. Note that the QAPD provided in Appendix B, written for the Hermes Test Reactor, is also applicable to Hermes 2.

12.10 REACTOR OPERATOR TRAINING AND REQUALIFICATION

The operating training and requalification plan is developed and implemented in accordance with 10 CFR 55 as it pertains to non-power facilities. Kairos Power complies with the requirements of 10 CFR 55 as it pertains to non-power facilities (e.g., 10 CFR 55.53(j), 10 CFR 55.53(k), 10 CFR 55.61(b)(5)). The operating training and requalification plan will be provided with the application for the Operating License, consistent with the requirements in 10 CFR 50.34(b)(8). The qualification process will include passing a comprehensive written exam and an operating test as required by 10 CFR 55.

12.11 STARTUP PLAN

The startup plan will be provided with the application for the Operating License, consistent with the requirements in 10 CFR 50.34(b)(6)(iii).

12.12 REFERENCES

1. American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.4-2016, "American National Standard for the Selection and Training of Personnel for Research Reactors." 2016.
2. American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.1-2007, "The Development of Technical Specifications for Research Reactors." 2007.
3. American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.16-2015, "Emergency Planning for Research Reactors." 2015.

4. American National Standards Institute/American Nuclear Society (ANSI/ANS) 15.8–1995 (R2005), "Quality Assurance Program Requirements for Research Reactors." 1995.

APPENDIX A

DESCRIPTION OF THE EMERGENCY PLAN

Appendix 12A. Emergency Planning

Introduction

Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," establishes requirements for emergency plans 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 (Appendix E.II).

A. Facility Emergency Organization

Appendix E.II.A requires information regarding 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.

A.1 Facility Organization

The minimum staff required to conduct routine and immediate emergency operations is maintained at the station on a continuous basis. Staffing is described in Section 12.1 of the Preliminary Safety Analysis Report. Station administrative procedures will provide the details of the station organization, including reporting relationships. There is no offsite emergency organization required to cope with emergencies because the Emergency Planning Zone (EPZ) is coincident with the site boundary and no offsite emergency plan actions are required (see Section 2.1).

A.2 Authorities and Responsibilities of Facility Emergency Personnel

The senior individual on-shift is responsible for assessing and declaring an emergency, and assuming command and control responsibilities following an emergency declaration. Upon declaration of an emergency, designated members of the 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 plant assessment and technical support for operations; and operations personnel focus on plant operations.

Additional personnel may be designated by station management as emergency responders providing special expertise deemed beneficial, but not mandatory, to the planned response. The individuals assigned as emergency response personnel are designated by station management based on the technical 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 application for an Operating License.

Emergency Director

In the event of an emergency, the senior individual on-shift will be the Emergency Director (ED). This individual will fulfill this role until duties are transitioned to the dedicated replacement. The emergency classification levels and the associated protective actions will be provided in the application for the Operating License.

The responsibilities of the ED are as follows.

- Declare and classify the emergency
- Direct emergency operation and ensure proper implementation of the emergency response plan
- Ensure that any necessary NRC notifications are made in accordance with the applicable requirements
- Authorize emergency workers to incur radiation exposures in excess of normal occupational limits, with the concurrence of the Radiation Safety Officer (RSO), if available. This function cannot be delegated.
- Terminate the emergency and initiate recovery operations
- Assess conditions in the facility after termination of the emergency to determine the proper course of further recovery actions
- Authorize an evacuation of all or part of the site.
- Authorize reentry into the facility (or portion thereof) that required evacuation during the emergency
- Establish and coordinate recovery/re-entry efforts
- Evaluate the causes of the emergency and recommend corrective actions before returning the facility to a normal operating status
- Coordinate emergency response actions with the off-site emergency support services
- Request augmented support as appropriate

Radiation Safety Officer

In the event of an emergency, the senior health physics person on-shift will be responsible for the radiological health physics aspects of the emergency.

The responsibilities for the RSO are to:

- Evaluate personnel doses received during the incident
- Assess subsequent potential doses and recommend protective actions, as appropriate
- Assist the ED and help determine the course of further action

A.3 Means for Notifications

Kairos Power will provide the capability for 24-hour notification to onsite and offsite organizations including a primary and backup means to accomplish the required notifications.

B. Authorities and Responsibilities of Governmental Agencies

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.

The arrangements with the City of Oak Ridge and Oak Ridge Central Fire Station, Oak Ridge Police Department, Oak Ridge Methodist Medical Center, and the State of Tennessee, will be obtained and documented and included in the application for an Operating License, to ensure a clear understanding of the emergency support responsibilities of each organization.

B.1 Federal Agencies

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.

Department of Energy – Oak Ridge Office

The Radiation Emergency Assistance Center/Training Site is a Department of Energy asset operated by Oak Ridge Associated Universities in cooperation with the Oak Ridge Methodist Medical Center in Oak Ridge, Tennessee. This organization can provide 24-hour availability to Kairos Power for medical/radiological emergencies which exceed in-house capabilities.

B.2 State Agencies

Tennessee Emergency Management Agency

Notification procedures (e.g., telephone, electronic messaging, written reports, etc.) will be maintained and the Tennessee Emergency Management Agency (TEMA) will be notified of an emergency declaration. The methods used to notify TEMA, and the information provided to TEMA, will be established in coordination with TEMA.

B.3 County Agencies

Roane County Office of Emergency Management

The Roane County Office of Emergency Management will assist by providing emergency support mainly in the form of transportation, communications, and equipment, when such assistance is sought by local emergency support agencies.

Roane County Sheriff's Department

The Roane County Sheriff's Department will assist in law enforcement activities responding to the facility as requested by the City of Oak Ridge Police Department.

B.4 Local Agencies

Anderson County Ambulance Service

The Anderson County Ambulance Service operates a local ambulance service and can provide transportation for injured and/or contaminated personnel. The decision as to the need to transport injured and/or contaminated personnel to a hospital will be made by attending medical personnel with advice from the RSO.

Oak Ridge Fire Department

The Oak Ridge Fire Department will provide assistance during emergencies involving actual or potential fire, explosions, or injuries.

Oak Ridge Police Department

The responsibilities of the Oak Ridge Police Department during an emergency are to:

- Respond to emergencies arising from a threat or a threatened or actual breach in physical security. The standard operating procedure (SOP) response capabilities will be detailed in the Physical Security Plan.
- Monitor and maintain the security of the facility after any emergency evacuation.
- Coordinate with other law enforcement agencies, as needed.

C. Protective Measures

Appendix E.II.C requires information with respect to 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; and the expected response of offsite agencies in the event of an emergency. The EPZ is coincident with the site boundary (see Section 2.1), therefore only protective measures within the site boundary are discussed.

The steps for taking protective action within the EPZ are:

1. The individual who initially confirms an emergency situation will immediately contact the Control Room and describe the emergency.
2. The ED is responsible for assessing and declaring an emergency and will classify the emergency.
3. The NRC will be notified of the class of emergency by the ED when required by applicable licenses and regulations.
4. The ED will mobilize that part of the facility organization appropriate for the emergency.

5. The 24-hr per day emergency call list for emergency response personnel is posted in the facility, including the (Control Room).
6. Required off-site support agencies will then be mobilized (normally by telephone) by the ED.
7. Corrective and Protective actions will be implemented in accordance with site procedures, as required for the situation, at the discretion of the ED.

Protective measures to be taken within the site boundary during an emergency may include the following:

- Performing first aid
- Moving personnel away from hazardous areas
- Contamination control measures including moving personnel away from contaminated areas
- Establishing restricted areas
- Site evacuation of non-essential personnel

The public address system and action specific alarms (e.g. site evacuation) can be used to communicate appropriate protective actions.

D. First Aid, Decontamination, and Emergency Transportation

Appendix E.II.D requires discussion of features of the facility to be provided for onsite emergency first aid and decontamination and for emergency transportation of onsite individuals to offsite treatment facilities.

D.1 Contamination Control and Personnel Decontamination

The RSO will coordinate necessary contamination control and decontamination of personnel.

- If there are a number of people involved in an emergency where there is a possibility for contamination, injured personnel will be monitored first.
- Contaminated personnel will be kept in an area isolated from other personnel activities, to avoid the spread of contamination.
- Injured personnel will be decontaminated if possible, and then dispatched to the either Oak Ridge Methodist Medical Center or the University of Tennessee Hospital.
- Monitoring and decontamination may occur in route or after arrival, depending on the nature of the injury.
- After injured persons are cared for, uninjured personnel will be checked for contamination, and necessary action taken to remove whatever contamination is detected.

D.2 First Aid, Decontamination Facilities and Equipment

The following describes first aid and decontamination facilities and equipment:

- Personnel first aid and decontamination kits are available throughout the plant.
- Showers are available that can be used for personnel decontamination.

- In the event on-site showers are not accessible or available, there are personnel decontamination facilities at Oak Ridge Methodist Medical Center and the University of Tennessee Hospital.

D.3 Medical Transportation

The responding medical staff will decide where injured persons are taken, based on the:

- Nature and severity of their injuries
- Level of radioactive contamination
- Personnel with serious injuries, with contamination, will be transported by ambulance directly to the emergency room of Oak Ridge Methodist Medical Center or University of Tennessee Hospital.

E. Offsite Treatment

Appendix E.II.E requires information regarding provisions to be made for emergency treatment at offsite facilities of individuals injured as a result of licensed activities.

The Oak Ridge Methodist Medical Center and the University of Tennessee Medical Center have standard operating procedures for dealing with radiological emergencies, including contaminated patients.

F. Training

Appendix E.II.F requires discussion of training for employees, including those assigned specific authority and responsibility in the event of an emergency, and for other persons who are not employees of the licensee but whose assistance may be needed in the event of a radiological emergency.

An initial training and periodic retraining program will be conducted to maintain the ability of emergency response personnel to perform their assigned functions. The personnel involved in the training program would include facility personnel responsible for decision-making and transmitting emergency information.

In addition, offsite personnel and agencies whose assistance is needed in responding to an emergency will be provided training as appropriate, such as briefings or site orientation visits.

The content of the training program will include the overall Emergency Plan and the relevant implementing procedures. Details of the training program will be provided in the application for an Operating License.

G. Evacuation

Appendix E.II.G requires discussion of preliminary analyses projecting the time and means to be employed in the notification of State and local governments and the public in the event of an emergency. This requirement is for a nuclear power reactor, however, the facility is licensed as a test reactor, and therefore this requirement is not applicable. Furthermore, there are no transient or permanent populations within the EPZ, therefore no analysis of evacuation is required. State and local governments and the public will be notified as appropriate.

H. Emergency Equipment and Facilities

Appendix E.II.H requires discussion of preliminary analyses reflecting the need to include facilities, systems, and methods for identifying the degree of seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including capabilities for dose projection using real-time meteorological information and for dispatch of radiological monitoring teams within the EPZs; and a preliminary analysis reflecting the role of the onsite technical support center (TSC) and the emergency operations facility (EOF) in assessing information, recommending protective action, and disseminating information to the public.

Preliminary analysis indicates the EPZ is coincident with the site boundary. Facilities, systems, and methods for identifying the seriousness of the radiological consequences of emergency situations will be available. Because there is no offsite release above the Environmental Protection Agency Protective Action Guides (EPA PAGs), there is no need for offsite monitoring teams, TSC, or EOF.

Timely notification will be made to the public for a declared emergency.

H.1 Emergency Support Center

The control room serves as the Emergency Support Center (ESC). The ESC will be the central point from which emergency control directions will be given. Additional space is available to support the emergency response if needed.

H.2 Assessment Facilities and Equipment

A listing of the current locations for emergency equipment cabinets and other emergency equipment storage areas, plus representative equipment inventories for these storage locations, will be provided in the application for an Operating License.

H.3 Portable and Fixed Radiological Monitors

Portable radiation monitoring instruments are available for use during an emergency. Some of these monitors are kept in emergency equipment cabinets and others are routinely used for normal operations. A representative listing of these instruments includes:

- High-range gamma ion chamber survey meters
- Medium-range beta/gamma ion chamber survey meters
- Beta/gamma Geiger-Mueller survey meters
- Neutron survey meters
- Alpha survey meters

H.4 Sampling Equipment

There are portable air samplers available for use in an emergency. A representative listing includes:

- Continuous particulate air monitors (on carts)
- High-volume particulate air samplers
- Medium-volume particulate and halogen air samplers
- Low-volume, battery-operated (lapel) particulate air samplers

H.5 Instrumentation for Specific Radionuclide Identification and Analysis

The following systems are representative of available equipment:

- Multi-channel analyzer
- Liquid scintillation counter
- Gas flow proportional counter

The actual equipment in the facility will be specified in the application for an Operating License.

APPENDIX B

QUALITY ASSURANCE PROGRAM

HER-PQ-PRG-000-001

Kairos Power LLC
707 W. Tower Ave
Alameda, CA 94501

Quality Assurance Program for the Kairos Power Hermes Reactor Facility

Revision No. 1
Document Date: 09/2022

Non-Proprietary

Rev	Description of Change	Date
0	Initial Issuance	08/2021
1	Changes to Sec 1.2 – definition of term "safety related item" aligned with Hermes Design. Changes to Sec 2.4 revised item to read SSC/services Changes to Sec 3.7 revised item to read SSC	See Document Date

EXECUTIVE SUMMARY

This document provides a description of the Quality Assurance Program (QAP) for the Kairos Power LLC (Kairos Power) Hermes Reactor Facility (Hermes). The Hermes reactor is a non-power reactor as described in 10 CFR 50.21, "Class 104 licenses; for medical therapy and research and development facilities." NRC NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content," directs applicants to ANSI/ANS-15.8-1995, "Quality Assurance Program Requirements for Research Reactors," (ANSI/ANS-15.8) when preparing an application for non-power reactors. This standard is endorsed by NRC Regulatory Guide 2.5, "Quality Assurance Program Requirements for Research and Test Reactors" (RG 2.5).

The Hermes Quality Assurance Program Description (H-QAPD) provides the methods and establishes quality assurance and administrative control requirements that meet 10 CFR 50.34 based on the criteria of ANSI/ANS-15.8-1995 as endorsed by RG 2.5, Revision 1.

The scope of this QAP includes design, construction, and operation phase activities for Hermes. Consistent with common licensing practice, text is written in the present tense, active voice, including discussions of activities and processes associated with a phased implementation of design, construction, and operation.

The document is divided into three parts:

1. Introduction,
2. Design, Construction, and Modifications, and
3. Facility Operations

Kairos Power is implementing this program to satisfy quality assurance requirements for use in the Hermes application submitted in accordance with 10 CFR 50 (as applicable):

- Construction Permit (CP) Applications pursuant to 10 CFR 50.34(a)(7)
- Operating License (OL) Applications pursuant to 10 CFR 50.34(b)(6)(ii)

Note: The H-QAPD is distinct from the Kairos Power Quality Assurance Program for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor that is the subject of a separate Topical Report which, at this writing, is under review by the NRC staff. That document describes the quality assurance program for the Kairos Power commercial power reactor(s). The H-QAPD is specific and unique to the non-power Hermes reactor facility.

Abbreviations

Term	Abbreviation
ANS	American Nuclear Society
ANSI	American National Standards Institute
CEO	Chief Executive Officer
CFR	Code of Federal Regulations
CP	Construction Permit
FSAR	Final Safety Analysis Report
H-QAPD	Quality Assurance Program Description for the Hermes Reactor Facility
KP	Kairos Power
LLC	Limited Liability Company
M&TE	Measurement & Test Equipment
NRC	Nuclear Regulatory Commission
OL	Operating License
QA/QC	Quality Assurance/Quality Control
QAP	Quality Assurance Program
RG	Regulatory Guide
SAR	Safety Analysis Report
SSC	Structures, Systems, and Components

Table of Contents

POLICY STATEMENT

Kairos Power shall design, procure, deliver, construct, and operate the Hermes Reactor Facility (Hermes) in a manner that ensures the health and safety of the public and workers. These activities shall be performed in compliance with the requirements of the Code of Federal Regulations and applicable laws and regulations of the state and local governments.

The Quality Assurance Program (QAP) for Hermes is described in this document and associated implementing documents. Together they provide for control of Kairos Power activities that affect the quality of safety-related structures, systems, and components (SSCs) and include all planned and systematic activities necessary to provide adequate confidence that such SSCs perform satisfactorily in service. This Quality Assurance Program Description for the Hermes Reactor Facility (H-QAPD) may also be applied to certain equipment and activities that are not safety-related, but support safe plant operations, or where other NRC guidance establishes program requirements.

The H-QAPD is the top-level program document that establishes the manner in which quality is to be achieved and presents Kairos Power's overall philosophy regarding achievement and assurance of quality for Hermes. Implementing documents assign more detailed responsibilities and requirements and define the organizational interfaces involved in conducting activities within the scope of the QAP. Senior management establishes overall expectations for effective implementation of the QAP and is responsible for obtaining the desired end result. Compliance with the H-QAPD and implementing documents is mandatory for personnel directly or indirectly associated with implementation of the Hermes QAP.

Michael Laufer, Chief Executive Officer	8-31-2021 Date

1 INTRODUCTION

This document provides the description of the Kairos Power LLC Quality Assurance Program (QAP) for the site selection, design, construction, and operation of the Kairos Power Hermes Reactor Facility (Hermes).

The Quality Assurance Program Description for the Hermes Reactor Facility (H-QAPD) is the top-level program document that establishes the quality assurance policy and assigns major functional responsibilities for all quality-related activities conducted by or for Hermes.

The Hermes reactor is a non-power reactor as described in 10 CFR 50.21, "Class 104 licenses; for medical therapy and research and development facilities." NRC NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content," directs applicants to ANSI/ANS-15.8-1995, "Quality Assurance Program Requirements for Research Reactors," (ANSI/ANS-15.8) when preparing an application for non-power reactors. This standard is endorsed by NRC Regulatory Guide 2.5, "Quality Assurance Program Requirements for Research and Test Reactors" (RG 2.5).

The H-QAPD describes the methods and establishes quality assurance (QA) and administrative control requirements that meet 10 CFR 50. 34 based on the criteria of ANSI/ANS-15.8 as endorsed by RG 2.5, Revision 1.

The Hermes QAP comprises the document that describes the QA elements (i.e., the H-QAPD), along with the associated implementing documents. Procedures and instructions that prescribe quality-related activities are developed prior to commencement of those activities. Policies that establish high-level responsibilities and authority for carrying out important administrative functions are outside the scope of the H-QAPD.

1.1 SCOPE AND APPLICABILITY

The H-QAPD applies to design-phase, construction-phase, and operations-phase activities, including those in support of Construction Permit (CP) and Operating License (OL) applications affecting the quality and performance of safety-related structures, systems, and components (SSCs), including, but not limited to:

Designing	Shipping	Inspecting
Siting	Receiving	Testing
Procuring	Storing	Operating
Fabricating	Constructing	Maintaining
Cleaning	Erecting	Repairing
Handling	Installing	Modifying

Safety-related SSCs within the scope of the H-QAPD are identified by design documents. The technical aspects of these items are considered when determining program applicability, including, as appropriate, the item's design safety function. The H-QAPD may be applied to certain activities where regulations other than 10 CFR 50 establish QA requirements for activities within their scope. Implementing documents establish program element applicability.

1.2 DEFINITIONS

The definitions provided in ANSI/ANS-15.8, Section 1.3, apply to this document with the following exception:

- The term "safety-related items" defined in ANSI/ANS-15.8, Section 1.3 will be replaced with the term "safety-related SSCs" and will be defined as:

  Those SSCs that are relied upon to remain functional during normal operating conditions and during and following design basis events to assure:

  - The integrity of the portions of the reactor coolant boundary relied upon to maintain coolant level above the active core;
  - The capability to shut down the reactor and maintain it in a safe shutdown condition; or
  - The capability to prevent or mitigate the consequences of accidents which could result in potential exposures exceeding the limits set forth in 10 CFR 100.11.

2 DESIGN, CONSTRUCTION, AND MODIFICATIONS

This section provides the requirements for establishing, managing, conducting, and assessing the program of controls over the design, construction, and modification of Hermes. This section is implemented as applicable to the specific scope of work activities.

2.1 ORGANIZATION

This section describes the Kairos Power organizational structure supporting Hermes including functional responsibilities, levels of authority, and interfaces for establishing, executing, and verifying H-QAPD implementation during design, construction, and operations phases.

The organizational structure and assignment of responsibilities is defined and documented such that: (a) quality is achieved and maintained by those who have been assigned responsibility for performing work; and (b) quality achievement is verified by persons not directly performing the work. Persons responsible for ensuring that appropriate controls have been established, and for verifying that activities have been correctly performed, have sufficient authority, access to work areas, and independence to: (a) identify problems; (b) initiate, recommend, or provide corrective action; and (c) ensure corrective action implementation. It is recognized that for Hermes, the organization is small, and personnel may perform multiple functions.

The organizational structure includes corporate/support/off-site and on-site functions for Hermes including interface responsibilities for multiple organizations that perform quality-related functions. Implementing documents assign more specific responsibilities and duties, and define the organizational interfaces involved in conducting activities and duties within the scope of the H-QAPD. Management considers the timing, extent, and effects of organizational structure changes.

During design, Safety Assurance and Quality Management is responsible to size the Quality Assurance staff commensurate with the duties and responsibilities assigned. During construction and operations, this responsibility transitions to the Site Executive.

The responsibility for quality-related activities during design, construction, and operations phases are shown below:

Design Phase	Construction Phase	Operations Phase
• Technology Development • Engineering • Fabrication • Supply Chain • Safety Assurance and Quality	• Construction • Engineering • Fabrication • Supply Chain • Construction Testing • Document Control and Other Support Services • QA/QC	• Operations • Maintenance • Engineering • Supply Chain • Startup/Preop Testing • Document Control and Other Support Services • QA/QC

Design, engineering, environmental, and construction services may be provided to Kairos Power for Hermes by contractors in accordance with their quality programs.

The following sections describe the reporting relationships, functional responsibilities, and authorities for functional organizations implementing and supporting the Hermes QA Program. The organization for Hermes is shown in Figure 2.1-1.

2.1.1 Chief Executive Officer

The Chief Executive Officer (CEO) is responsible for all aspects of design, construction, and operations. The CEO is also responsible for all technical and administrative support activities provided by Kairos Power and contractors. The CEO directs Technology Development, Engineering, Supply Chain, Safety Assurance & Quality (during the design phase and for corporate support during construction and operations), and the Site Executive (during construction and operations) in fulfillment of their responsibilities

2.1.2 Design Phase and Corporate Support

The following functions report to the CEO during the design phase and during corporate/off-site support of construction and operations.

2.1.2.1 Technology Development

Reports to the CEO and is responsible for fuel, coolant, and materials qualification and testing and associated design analysis, including modeling.

2.1.2.2 Engineering Design

Reports to the CEO and is responsible for engineering design and support services.

2.1.2.3 Fabrication

Reports to the CEO and is responsible for fabrication of components.

2.1.2.4 Supply Chain

Reports to the CEO and is responsible for supply chain management (including supplier evaluation) and procurement.

2.1.2.5 Safety Assurance and Quality

Reports to the CEO and is responsible for nuclear safety assurance, document control and records management, and the establishment and implementation of the Hermes H-QAPD.

The Quality Assurance function reports to the Safety Assurance and Quality function and is responsible for planning and performing activities to verify development and effective implementation. Effective implementation includes, but is not limited to, developing and maintaining the H-QAPD, evaluating conformance to QA Program requirements through assessments and technical reviews, independent oversight of the implementation of quality activities, and ensuring that suppliers providing quality services, parts, and materials for Hermes are conforming with the applicable QA requirements through Kairos Power supplier audits, and managing Quality Assurance organization resources.

The QA function has sufficient independence from other Kairos Power priorities to bring forward issues affecting safety and quality and makes judgments regarding quality in all areas regarding Hermes design activities as appropriate. QA may make recommendations to management regarding improving the quality of work processes. If QA disagrees with any actions taken by the organization and is unable to obtain resolution, QA shall inform Safety Assurance and Quality management and bring the matter to the attention of the CEO, who determines the final disposition.

Figure 2.1-1 reflects the QA function (within the Safety Assurance and Quality function) but with a “dotted line” relationship directly to the CEO, irrespective of specific organizational structure.

2.1.3 Site Executive

During construction and operations phases, the Site Executive reports to the CEO and is responsible for site related construction and operation activities. Transition from design phase to construction and operations phases occurs such that those positions required to support quality-related activities retain their applicable responsibilities until it is deemed that they are no longer necessary.

2.1.3.1 Construction Phase Management

Construction Phase Management reports to the Site Executive and is responsible for construction activities, including construction, fabrication, engineering, supply chain, construction testing, document control and other support services, and QA/QC.

Construction Phase Management is staffed and has the appropriate authority required to perform quality-related construction activities. Interfaces between site/construction phase management and corporate support is defined in implementing procedures.

The Kairos Power Quality Assurance organization is responsible for independent oversight of the implementation of activities at Hermes including but not limited to construction; engineering; procurement; and construction testing. QA is responsible for assuring conformance with regulatory requirements and procedures through assessments and technical reviews; and ensuring that suppliers providing quality services, parts, and materials to Hermes are meeting quality requirements through third-party audits, Kairos Power supplier audits and/or other acceptable means.

QA has sufficient independence from other Hermes construction priorities to bring forward issues affecting safety and quality and makes judgments regarding quality in all areas regarding Hermes construction activities as appropriate. QA may make recommendations to management regarding improving the quality of work processes. If QA disagrees with any actions taken by the organization and is unable to obtain resolution, QA shall inform Construction Phase Management, and bring the matter to the attention of the CEO, who determines the final disposition.

2.1.3.2 Operations Phase Management

Reports to the Site Executive and is responsible for plant operation activities, including operations, maintenance, engineering, supply chain, startup/preop testing, document control and other support services, and QA/QC.

Operations Phase Management is staffed and has the appropriate authority required to perform quality-related operations activities. Those positions required to support activities after fuel load retain their applicable construction/pre-operation responsibilities until it is deemed that they are no longer necessary. As the construction of systems (or portions thereof) is completed, control and authority (including oversight, configuration, and operations) is transferred from Construction Phase Management to the cognizant departments in the operational phase. During the transition, responsibilities are clearly defined in instructions and procedures to ensure appropriate authority is maintained for each SSC.

The Quality Assurance organization is responsible for independent oversight of the implementation of activities including but not limited to operations; maintenance; engineering; startup/preop testing; and procurement.

QA is responsible for assuring conformance with quality requirements and procedures through assessments and technical reviews; monitoring organizational processes to ensure conformance to commitments and licensing document requirements; and ensuring that suppliers providing quality services, parts, and materials to Hermes are conforming to applicable QA requirements through third-party audits, Kairos Power supplier audits and/or other acceptable means.

QA has sufficient independence from other Hermes operational priorities to bring forward issues affecting safety and quality and makes judgments regarding quality in areas regarding Hermes operations activities as appropriate. QA may make recommendations to management regarding improving the quality of work processes. If QA disagrees with any actions taken by the organization and is unable to obtain resolution, QA shall inform Operations Phase Management, and bring the matter to the attention of the CEO, who determines the final disposition.

2.1.4 Authority to Stop Work

Quality Assurance and Quality Control Inspection personnel have the authority, and the responsibility, to stop work in progress which is not being done in accordance with approved procedures or where safety or SSC integrity may be jeopardized. This authority extends to off-site work performed by suppliers that furnish safety-related materials and services to Hermes.

2.1.5 Quality Assurance Organizational Independence

Independence shall be maintained between the organization(s) performing the checking (quality assurance and control) functions and the organizations performing the functions. This provision is not applicable to design review/verification.

2.2 QUALITY ASSURANCE PROGRAM

Kairos Power establishes the necessary measures and governing procedures to implement the Hermes QAP as described in the H-QAPD at the earliest time consistent with the schedule for accomplishing quality-related activities. Kairos Power is committed to implementing this QAP in all aspects of work that are important to the safety of the nuclear plants as described and to the extent delineated in the H-QAPD. This QAP shall include monitoring activities against acceptance criteria in a manner sufficient to provide assurance that the activities important to safety are performed satisfactorily. Further, Kairos Power ensures through the systematic process described herein that its suppliers of safety-related equipment or services meet applicable quality requirements. Senior management is regularly apprised of the adequacy of implementation of the QAP through the assessment functions described in Section 2.18.

The objective of the Hermes QAP is to assure that Hermes Reactor Facility is designed, constructed, and operated in accordance with governing regulations and license requirements. The program is based on the criteria of ANSI/ANS-15.8, as further described in this document. This QAP applies to those quality-related activities that involve the functions of safety-related structures, systems, and components (SSCs) associated with the design, fabrication, construction, and testing of the SSCs of the facility and to the managerial and administrative controls to be used to assure safe operations. Examples of CP/OL program safety-related activities include, but are not limited to, engineering related to safety-related SSCs, site geotechnical investigations, site engineering analysis, seismic analysis, and meteorological analysis. A list or system that identifies SSCs and activities to which this program applies is maintained at the appropriate facility. Design documents are used as the basis for this list. Cost and scheduling challenges must be addressed; however, they do not prevent proper implementation of the QAP. This includes the managerial and administrative aspects of internal and external activities that affect quality of Hermes and programs.

This QAP provides for the use of a graded approach to quality. The measures applied to a particular engineered or administrative control or control system may be graded commensurate with the reduction of the risk attributable to that control or control system. This approach to achieving levels of quality is described in the H-QAPD and related implementing documents.

In general, the program requirements specified herein are detailed in implementing procedures that are either Kairos Power implementing procedures, or supplier implementing procedures governed by a supplier quality assurance program.

Delegated responsibilities may be performed under a supplier’s quality program, provided that it has been approved in accordance with the H-QAPD. Periodic assessments are conducted to assure compliance with the supplier's or principal contractor's quality program and implementing procedures. In addition, routine interfaces with their personnel provide added assurance that quality expectations are met. Assessments may be planned and performed by Kairos Power qualified assessors or independent contractors or consultants as determined by Quality Management.

Personnel assigned to implement elements of the H-QAPD shall be capable of performing their assigned tasks. Kairos Power establishes and maintains formal indoctrination and training programs for personnel performing, verifying, or managing activities within the scope of the H-QAPD to ensure that suitable proficiency is achieved and maintained. Sufficient managerial depth is provided to cover absences of incumbents. When required by code, regulation, or standard, specific qualification and selection of personnel is conducted in accordance with those requirements as established in applicable Kairos Power procedures. Indoctrination includes the administrative and technical objectives and requirements of the applicable codes and standards and QAP requirements as necessary. Records of personnel training and qualification are maintained.

2.3 DESIGN CONTROL

Kairos Power has established and implements a process to control the design, design changes, and temporary modifications of items that are subject to the provisions of the QAP. The design process includes provisions for the development, verification, approval, release, status, distribution, and revision of design inputs and outputs.

2.3.1 Design Requirements

Applicable design inputs, such as design bases, performance requirements, regulatory requirements, codes, and standards shall be identified and documented.

2.3.2 Design Process

Design interfaces shall be identified and controlled, and the design efforts shall be coordinated among the participating organizations.

The applicability of standardized or previously proven designs, with respect to meeting pertinent design inputs, shall be verified for each application. Known problems affecting the standardized or previously proven designs, and their effects on other features, shall be considered. Deviations from the established and documented design inputs, including the reasons for the changes, shall be documented and controlled.

The design organization is responsible to ensure that the final design shall:

1. be relatable to design input by documentation in sufficient detail to permit design traceability and verification, and
2. identify assemblies and/or components that are part of the item being designed

When a computer design program is used to develop portions of the facility design or to analyze a design for acceptability, that program shall be fully documented, validated, and controlled to ensure the correctness of its output. When a design program must be developed, the program shall be controlled to ensure that it is fully documented and validated. Where changes to previously valid computer programs are made, documented revalidation shall be required for the change. Verification of design-unique computer programs shall include appropriate benchmark testing.

2.3.3 Design Verification

Independent design verifications shall be used to verify the adequacy of design by one or more of the following:

1. performance of design reviews,
2. use of alternate calculations,
3. performance of qualification tests, or
4. comparison of similar proven systems.

The responsible design organization shall identify and document the design verification method or methods used. Design verification is performed by competent individuals or groups other than those who performed the design, but who may be from the same organization. In all cases the design verification shall be completed prior to reliance upon the component, system, structure, or computer program to perform its function in operations.

In the event that qualification testing is needed to verify design, the use of qualification tests is defined in a formal test plan that shall include appropriate acceptance criteria and shall demonstrate the adequacy of performance under conditions that simulate the most adverse design conditions. Test results are documented and evaluated by the responsible design organization to ensure that test requirements have been met.

2.3.4 Design Documents and Records

Design documents and records, which provide evidence that the design and design verification process were performed, shall be collected, stored, and maintained for the life of the safety-related item.

2.3.5 Commercial Grade Items

The use of commercial-grade equipment in safety-related applications shall be reviewed to ensure that it can adequately perform its intended function. When a commercial grade item, prior to its installation, is modified or selected by special inspection and/or testing to requirements that are more restrictive than the supplier's published product description, the component part shall be represented as different from the commercial grade item in a manner traceable to a documented definition of the difference.

2.3.6 Change Control

Modifications to safety-related structures, systems, and components, or computer codes shall be based on a defined "as-exists" design. Changes to verified designs shall be documented, justified, and subject to design control measures commensurate with those applied to the original design. The control measures shall include assurance that the design analyses for the structure, system, component, or computer code are still valid. Where a significant design change is necessary because of an incorrect design, the design process and verification procedure should be reviewed and modified as necessary.

2.4 PROCUREMENT DOCUMENT CONTROL

Procedures shall be established to ensure that procurement documents contain sufficient technical and quality requirements to ensure that the items or services satisfy the needs of Kairos Power. Procurement documents at all procurement levels shall identify the documentation required to be submitted for information, review, or approval by Kairos Power. At each level of procurement, the procurement documents shall provide for access to the supplier's plant facilities and records, for inspection or audit by Kairos Power, a designated representative, or other parties authorized by Kairos Power.

Hermes procurement documents shall include Kairos Power's requirements for reporting and approving disposition of supplier's non-conformances associated with the items or services being procured. The procurement documents for safety-related SSCs/services should prohibit the supply/use of sub-standard or counterfeit parts or materials.

2.5 PROCEDURES, INSTRUCTIONS, AND DRAWINGS

Activities affecting quality shall be performed in accordance with documented instructions, procedures, or drawings appropriate to the circumstances. These documents shall include or reference appropriate quantitative or qualitative acceptance criteria for determining that activities have been satisfactorily accomplished.

2.6 DOCUMENT CONTROL

The preparation, issue, and change of documents which specify requirements that affect quality or prescribe activities affecting quality shall be controlled to ensure that correct documents are used. The document control system shall be documented and provide for:

1. identification of documents to be controlled and their specified distribution;
2. identification of assignment of responsibility for preparing, reviewing, approving, and issuing documents; and
3. review of documents for adequacy, completeness, and correctness prior to approval and issuance.

Major changes to controlled documents shall be reviewed and approved by the same organizations that performed the original review and approval unless other organizations are specifically designated.

2.7 CONTROL OF PURCHASED ITEMS AND SERVICES

The procurement of items and services shall be controlled to ensure appropriate procurement planning, source evaluation and selection, evaluation of objective evidence of quality furnished by the supplier, source inspection, audit, and examination of items or services for acceptance upon delivery or completion.

2.7.1 Supplier Selection

The selection of suppliers shall be based on evaluation of their capability to provide items or services in accordance with requirements of the procurement documents.

2.7.2 Work Control

Kairos Power shall establish measures to control the supplier’s performance to ensure that purchased items and services meet Hermes quality requirements. Controls may include test plans, review of supplier’s submitted documents, arrangements for source surveillance or inspection, and other technical and administrative interfaces with the supplier in accordance with procurement documents.

2.7.3 Verification Activities

The supplier shall be responsible for the quality of its product and shall verify and provide evidence of that quality. Supplier-generated documents shall be controlled, handled, and approved in accordance with established methods. Means shall be implemented to provide for the acquisition, processing, and recorded evaluation of technical, inspection, and test data against acceptance criteria. Based on complexity of the product and importance to safety, Kairos Power should independently verify the quality of a supplier’s product through source surveillances, inspections, audits, or review of the supplier’s non-conformances, dispositions, waivers, and corrective actions.

2.7.4 Item or Service Acceptance

Kairos Power shall establish a system to provide assurances that purchased items and services conform to procurement specifications. Methods used to accept an item or related service from a supplier shall be a supplier Certificate of Conformance, source verification, receiving inspection, post-installation test, or a combination thereof. Receiving inspection shall be performed in accordance with established procedures and instructions, to verify by objective evidence such features as proper configuration, identification, and cleanliness, and to determine any shipping damage, fraud, or counterfeit.

2.8 IDENTIFICATION AND CONTROL OF ITEMS

When specified by codes, standards, or specifications that include identification or traceability requirements, the item identification and control process shall be capable of providing identification and traceability control. Items' identification shall be maintained from the initial receipt or fabrication of the items up to and including installation and use. Where physical identification on the item is either impractical or insufficient, physical separation, procedural control, or other appropriate means shall be employed. Identification markings shall be applied using materials and methods which provide clear and legible identification and do not detrimentally affect the function or service life of the item. Markings shall be transferred to each part of an identified item when the item is subdivided and shall not be obliterated or hidden by surface treatment or coatings unless substitute means are provided. Where specified, items having limited calendar or operating life shall be identified and controlled to preclude use of items whose shelf life or operating life is expired.

2.9 CONTROL OF SPECIAL PROCESSES

Special processes include those in which the results are highly dependent on the control of the process or skill of the personnel. These are also those processes in which the specified quality cannot be readily determined by inspection or non-destructive testing of the product. Kairos Power implements the necessary measures and governing procedures to assure that special processes that require interim process controls to assure quality, such as welding, heat treating, and nondestructive examination, are controlled. These processes shall be controlled by instructions, procedures, drawings, checklists, travelers, or other appropriate means. Kairos Power and its suppliers are responsible to adhere to the approved procedures and processes when performing special processes for Hermes. The requirements of applicable codes and standards, including acceptance criteria for each process, shall be specified or referenced in the procedures or instructions that control the process. Records shall be maintained as appropriate for the currently qualified personnel, processes, and equipment associated with special processes.

2.10 INSPECTIONS

Inspections to verify conformance of an item or activity to requirements shall be planned, documented, and performed. The inspection program shall apply to procurement, construction, modification, and maintenance. Inspection of items in-process or under construction shall be performed for work activities where product quality cannot be determined by inspection of the completed product. The final inspection shall be planned to arrive at a conclusion regarding conformance of the item to specified requirements. Completed items shall be inspected for completeness, markings, calibration, adjustments, protection from damage, or other characteristics as required to verify the quality and conformance of the item to specified requirements. Associated quality records shall be examined for adequacy and completeness. Only items that have passed the required inspections and tests shall be used, installed, or operated. Measuring and Test Equipment (M&TE) used to perform inspections shall be identified in inspection documentation for traceability of inspection results.

Inspection results shall be documented. Acceptance of items shall be documented and approved by authorized personnel. Inspection shall be performed by persons other than those who performed the work being inspected but they may be from the same organization. Each person who verifies conformance of work activities for purposes of acceptance shall be qualified to perform the assigned inspection task. The need for formal training shall be determined and training activities conducted as required to qualify personnel who perform inspections and tests. On-the-job training shall be included, with emphasis on firsthand experience gained through actual performance of inspections. Records of inspection personnel's qualification shall be established and maintained by their employer.

2.11 TEST CONTROL

Formal testing shall be required to verify conformance of designated structures, systems, or components to specified requirements and demonstrate satisfactory performance for service or to collect data in support of design or fabrication. Testing shall include prototype qualification tests, proof tests prior to installation, and functional tests. Test results shall be documented and evaluated by a responsible authority to ensure that test requirements have been satisfied.

Computer programs used for operational control shall be tested in accordance with an approved verification and validation plan and shall demonstrate required performance over the range of operation of the controlled function or process.

2.12 CONTROL OF MEASURING AND TEST EQUIPMENT

Tools, gauges, instruments, and other M&TE used for activities affecting quality shall be controlled and calibrated or adjusted, at specified periods to maintain accuracy within specified limits. Out-of-calibration devices shall be tagged or segregated, and not used until they have been recalibrated. Records shall be maintained of calibration data traceable to the individual piece of M&TE. Calibration and control measures are not required when normal commercial equipment provides adequate accuracy.

2.13 HANDLING, STORAGE, AND SHIPPING

Handling, storage, and shipping of items shall be in accordance with work and inspection instructions, drawings, specifications, shipping instructions, or other pertinent documents or procedures for conducting the activity.

2.14 INSPECTION, TEST, AND OPERATING STATUS

The status of inspection and test activities shall be identified on the items or in documents traceable to the items in order to ensure that required inspections and tests are performed and to ensure that items which have not passed the required inspections and tests are not inadvertently installed or operated.

2.15 CONTROL OF NON-CONFORMING ITEMS AND SERVICES

Items that do not conform to requirements shall be controlled to prevent inadvertent installation or use. Controls on non-conforming items shall provide for identification, documentation, evaluation, segregation from like conforming items when practical, and disposition of non-conforming items. Non-conforming conditions shall be evaluated for further reporting to appropriate regulatory agencies. Non-conforming characteristics shall be reviewed, and recommended dispositions of non-conforming items proposed and approved, in accordance with documented procedures.

The disposition (use-as-is, reject, repair, or rework) of non-conforming items shall be identified and documented. Technical justification for the acceptability of a non-conforming item disposition "repair" or "use-as-is" shall be documented. Non-conformance to design requirements of items dispositioned "use-as-is" or "repair" shall be subject to design control measures commensurate with those applied to the original design. The as-built records shall reflect the accepted deviation. Repaired or reworked items shall be re-examined in accordance with applicable procedures and with the original acceptance criteria unless the non-conforming item disposition has established alternate acceptance criteria.

2.16 CORRECTIVE ACTIONS

Conditions adverse to quality shall be identified promptly and corrected as soon as practical. The corrective actions shall be in accordance with the design requirements unless those requirements were faulty. In the case of a significant condition adverse to quality, the cause of the condition shall be investigated, and corrective action taken to preclude recurrence.

2.17 QUALITY RECORDS

A records system or systems shall be established at the earliest practical time consistent with the schedule for accomplishing work activities. The records system or systems shall be defined, implemented, and enforced in accordance with written procedures, instructions, or other documentation. The records shall include as a minimum: inspection and test results, results of quality assurance reviews, quality assurance procedures, and engineering reviews and analyses in support of designs or changes and modifications.

Some records shall be maintained by or for the plant owner for the life of the particular item while it is installed in the plant or stored for future use. Such records shall include those meeting the following criteria:

1. those that are of value in demonstrating capability for safe operation;
2. those that are of value in maintaining, reworking, repairing, replacing, or modifying an item;
3. those that are of value in determining the cause or results of an accident or malfunction of a safety-related item;
4. those that provide required baseline data for in-service inspections; or
5. those that are of value in planning for facility decommissioning.

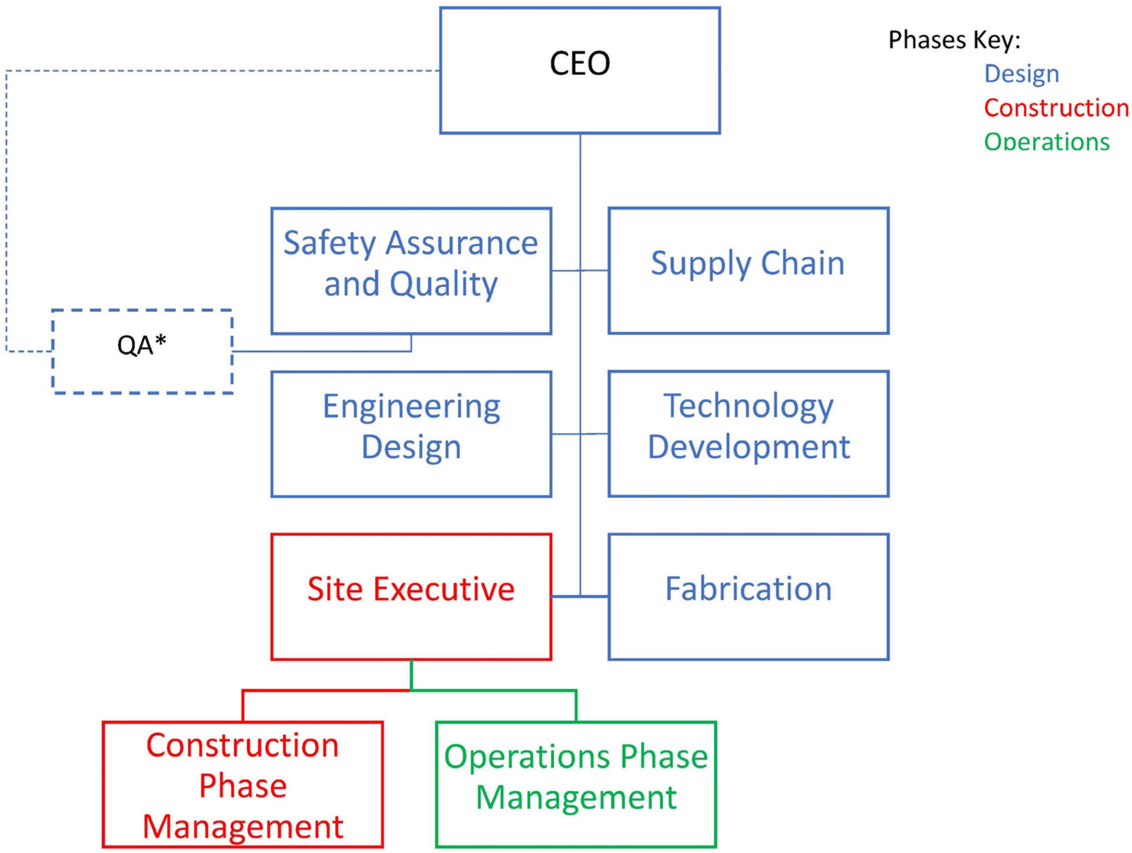
Other records shall be retained for a shorter period as determined by Kairos Power. The records shall be stored in a location or locations that prevent damage from moisture, temperature, and pestilence. Additional provisions shall be made for special processed records such as radiographs, photographs, negatives, microfilm, and magnetic media, to prevent damage from excessive light, stacking, electromagnetic fields, temperature, and humidity. Records maintained by a supplier shall be accessible to Kairos Power.

2.18 ASSESSMENTS

Kairos Power conducts periodic assessments of quality-related activities during design, construction, modification, and operations to evaluate the effectiveness of the as-implemented quality program. Assessments shall be performed in accordance with written procedures or checklists. Assessment results shall be documented and should be reviewed by management personnel who have responsibility for the area assessed. Conditions requiring prompt corrective action shall be reported immediately to the appropriate management of the assessed organization.

Management of the assessed organization or activity shall investigate adverse findings, schedule corrective action (including measures to prevent recurrence) and notify the appropriate assessing organization in writing of action taken or planned. The adequacy of the responses shall be evaluated by the assessing organization. Assessment records include assessment plans, reports, written replies, and the record of completion of corrective action. Personnel selected for assessment assignments shall have experience or training commensurate with the scope, complexity, or special nature of the activities to be assessed. The assessor shall have the capability to communicate effectively, both in writing and orally.

Figure 2.1-1. Kairos Power Organization for the Hermes Reactor Facility



- Construction
- Fabrication
- Engineering
- Supply chain
- Construction testing
- Document control and other support services
- QA/QC*

- Operations
- Maintenance
- Engineering
- Supply chain
- Startup/preop testing
- Document control and other support services
- QA/QC*

** QA function has direct access to levels of management necessary to assure effective execution of the QA program irrespective of organizational structure*

3 FACILITY OPERATIONS

This section provides the elements of a quality assurance program for conduct of operation at Hermes. The requirements shall be applied to equipment or operations as appropriate and consistent with its potential safety impact or program goals. Many of the program requirements are satisfied by existing documentation, or by procedures and activities required by other standards and requirements of the applicable permit(s) or license(s). Some requirements of the quality assurance program for operations may also be found in other documents, such as the Training Program, Emergency Plan, Security Plan, Technical Specifications, and the Radiation Protection Program. Such requirements do not need to be duplicated in the quality assurance program.

3.1 ORGANIZATION

Kairos Power shall provide sufficient resources in personnel and materials to safely conduct operations at Hermes. Planning should anticipate needs as appropriate for associated tasks. The organization structure shall be defined as required by Technical Specifications. Section 2.1.3.2, “Operations Phase Management,” contains additional detail.

3.2 QUALITY ASSURANCE PROGRAM

Kairos Power shall establish a quality assurance program for Hermes by implementing a policy for the conduct of operations. The policy should assign personnel to implement the policy and identify the goals for operating Hermes. Personnel assignments and progress toward achieving goals should be documented.

3.3 PERFORMANCE MONITORING

Kairos Power shall monitor facility performance relative to the goals used as performance indicators for Hermes. Kairos Power shall document periodic observations of operations and identify deficiencies. Kairos Power should assess deficiencies to ensure the execution of corrective actions that prevent recurrence. If appropriate, trend analysis should be performed to indicate where improvements or lessons learned could be implemented. Violations of operating practices should be addressed and documented as appropriate.

3.4 OPERATOR EXPERIENCE

Kairos Power shall document the methods for maintaining operator experience for Hermes. Operators should be responsible for maintaining experience in operating Hermes. This may be achieved by routine operation of Hermes and documentation of the activity. A method should be provided to make operators aware of important current information that is related to facility operations and individual job assignments. Operator training is addressed by the Kairos Power training program.

3.5 OPERATING CONDITIONS

Pre-operations checklists shall be used to determine or verify required pre-operational conditions and readiness to operate. Operating equipment shall be periodically monitored to detect abnormal conditions or adverse trends. Operating conditions should be documented in an operations logbook or other record. The operator should notify the appropriate level of management of abnormal situations.

3.6 OPERATIONAL AUTHORITY

Kairos Power shall establish the method for conducting operations and the responsibility for each shift for Hermes. Operating personnel shall conduct a comprehensive review of appropriate records and equipment before assuming responsibility for the facility. Operational authority may be transferred through a documented turnover briefing and facility walk-through procedures. These procedures should include checklists to record items important to facility status.

3.7 CONFIGURATION CONTROL

Equipment shall be identified that requires configuration control. Kairos Power is responsible for establishing and maintaining proper configuration for Hermes and should authorize any changes to safety-related SSCs. Configuration changes to safety-related SSCs should be documented. Before placing equipment into operation, the system shall be properly calibrated or checked, as appropriate, and any deficiencies in the equipment or the current configuration of the system documented. This should also address methods for temporary modifications. Maintenance that requires a change in the system shall be documented.

3.8 LOCKOUTS AND TAGOUTS

Locks and tags shall be placed on equipment when, for safety or other special administrative reasons, controls must be established. If there is potential for equipment damage or personnel injury during equipment operation, maintenance, inspection, or modification activities, or from inadvertent activation of equipment, a facility lockout/tagout procedure shall be implemented.

3.9 TEST AND INSPECTION

Tests shall be performed following system maintenance, design changes, or inspection that involves dismantlement of components or systems. A documented test plan shall be used to demonstrate that the component or system is capable of performing its intended function. The results of the test should be documented and retained in facility records as appropriate.

3.10 OPERATING PROCEDURES

Operating procedures shall provide appropriate direction to ensure that the facility is operated normally within its design basis, and in compliance with technical specifications. Operating procedures shall be written, reviewed, approved by appropriate management, controlled, and monitored to ensure that the content is technically correct and the wording and format are clear and concise. The facility policy on use of procedures should be documented and clearly understood by all operators. The extent of detail in a procedure should depend on the complexity of the task; the experience, education, and training of the users; and the potential significance of the consequences of error. The process for making changes and revisions to procedures should be documented. A controlled copy of all operations procedures should be maintained in the control room or equivalent area.

3.11 OPERATOR AID POSTINGS

Posted information that aids operators in performing their duties should be current and correct. Management should review operator aids to determine that they are necessary and correct before approving their posting. Postings should be checked periodically for continued applicability.

3.12 EQUIPMENT LABELING

Equipment shall be labeled to help facility personnel positively identify equipment they operate and maintain. Information on labels should be consistent with information found in facility procedures, valve lineup sheets, piping and instrument diagrams, or other documents. Labels should be permanent, securely attached, readable, and have appropriate information.

4 REFERENCES

4.1 American National Standard for Quality Assurance Program Requirements for Research Reactors, ANSI/ANS-15.8-1995.

4.2 NUREG 1537, Part 1 "Guidelines for Preparing and Reviewing Applications for The Licensing of Non-Power Reactors, Format and Content."

4.3 Regulatory Guide 2.5 Rev.1, "Quality Assurance Requirements for Research and Test Reactors."

Kairos Power

Chapter 13

Accident Analysis

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 13 ACCIDENT ANALYSIS

This chapter provides information and analyses that show that the health and safety of the public are protected. The analyses consider the potential radiological consequences in the event of malfunctions and the capability of the facility to accommodate such disturbances. The health and safety of the workers will be demonstrated in the application for an Operating License. This chapter demonstrates that the facility design features and bounding initial values for parameters expected to be controlled by technical specifications have been selected to ensure that no postulated event in the design basis leads to unacceptable radiological consequences to people or the environment.

The reactor design relies on a functional containment approach, described in Section 6.2, that results in the retention of the vast majority of the radioactive material available for release during a postulated event. The accident analysis presented in this chapter bounds all potential accident source terms by evaluating the dose consequences of a Maximum Hypothetical Accident (MHA). The MHA, described in Section 13.1.1, is a hypothetical scenario conservatively defined to bound the potential dose consequences of other events that are postulated for the test reactor design basis. The postulated event groups are described in Section 13.1.2 through Section 13.1.9.

The dose consequences of the MHA demonstrate the acceptability of the design when compared to regulatory dose limits. There are no dose limits defined in 10 CFR 50 for a non-power reactor; 10 CFR 100 defines dose limits applicable to the siting of a non-power reactor. The dose limits in 10 CFR 100.11 require that an applicant for a non-power reactor evaluate dose at the exclusion area boundary (EAB) and the low population zone (LPZ) as follows:

- EAB: An individual located on the EAB for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
- LPZ: An individual located on the outer boundary of the LPZ who is exposed to the radioactive cloud resulting from the MHA (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

Consistent with the requirements in 10 CFR 100.11(b), the size of the exclusion area, low population zone, and population center distance for multiple reactor facilities shall be fulfilled with respect to each reactor individually. The reactors are independent to the extent that a postulated event for one unit does not affect the safety of operation of the other unit. Therefore, the MHA and postulated events presented in this chapter apply to each unit and are evaluated against the siting criteria for each unit separately. The locations of the EAB and LPZ are provided in Section 2.1. The MHA analysis presented in this chapter demonstrates bounding dose consequences that are significantly lower than those specified in 10 CFR 100.11.

13.1 INITIATING EVENTS AND SCENARIOS

This section provides the events postulated for the reactor design basis. The events are grouped according to type and characteristics of the events. The event categories are:

- MHA
- Insertion of Excess Reactivity
- Salt Spills
- Loss of Forced Circulation (includes a loss of normal electric power)
- Mishandling or Malfunction of Pebble Handling and Storage System
- Radioactive Release from a Subsystem or Component
- General Challenges to Normal Operation
- Internal and External Hazard Events

The MHA is a scenario that bounds other postulated event groups. The analysis of the MHA in Section 13.2 demonstrates the safety margins of the design.

For postulated events, figures of merit for each event category provide surrogate metrics which demonstrate that the resulting dose is bounded by the dose consequences of the MHA analysis as described in KP-TR-022-P, "Postulated Event Methodology Technical Report" (Reference 2). Acceptance criteria for these figures of merit represent design limits that ensure the MHA is bounding. The acceptance criteria for the postulated event figures of merit are provided in Table 13.1-1.

The consequences of postulated events presented in this chapter would normally be mitigated by non-safety related SSCs for reactivity control and heat removal (and the building for confinement if radioactive material is released). However, consistent with the guidance in NUREG 1537, only the performance of the safety-related structures, systems, and components (SSCs) are credited in the postulated events. The performance of the SSCs also assume the worst single failure of any active components. This conservative approach to safety analysis provides additional confidence that the postulated events are bounded by the MHA. The safety classifications of SSCs are provided in Section 3.6.

The discussion on preventing certain events by design is provided in Section 13.1.10.

13.1.1 Maximum Hypothetical Accident

The MHA is an event where hypothesized conditions result in a conservatively analyzed release of radionuclides that bounds a potential release from other postulated events. The MHA analysis is consistent with the fission product release accident analysis required for the 10 CFR 100.11 determination of exclusion area, low population zone, and population center distances. The MHA is a bounding event with conservative radionuclide transport assumptions that challenge the important radioactive retention features of the functional containment. This section describes the key assumptions and non-physical conditions that are hypothesized to ensure that the dose consequences from the MHA analysis bounds the dose consequences from postulated events in the design basis. The details associated with these assumptions, as well as the methods used to calculate the dose consequences of the MHA are provided in Section 13.2.1.

13.1.1.1 Initial Conditions Assumptions

Normal operating parameters are discussed in Section 4.1. Conservative initial values are assumed for each operating parameter to maximize the release of radionuclides in the MHA.

The radioactive material that is at risk for release for the MHA includes radionuclides contained in the fuel, the radionuclides circulating in the Flibe, and the radioactive material at risk for release (MAR) distributed within the primary system (i.e., steel structures and graphite). Although radionuclides could have diffused away from the tri-structural isotropic (TRISO) fuel particles, the initial inventory of the small fraction of fuel that is defective at the initiation of the transient assumes that no diffusion has occurred. This hypothetical condition adds a bounding conservatism to the radionuclide release from the fuel and Flibe.

The TRISO fuel form and the basis for its radionuclide retention performance is discussed in Section 4.2.1. The methodology for determining the radionuclide behavior and retention properties of the fuel is provided in Section 3 of KP-TR-012, "KP-FHR Mechanistic Source Term Methodology Topical Report," (Reference 1). Fuel manufacturing and in-service performance specifications are discussed in Section 4.2.1.

The Flibe design is discussed in Section 5.1. The methodology for determining the radionuclide behavior and retention properties of the Flibe is provided in Section 4 of Reference 1. A bounding value for Flibe circulating activity is assumed as the initial condition.

A bounding value of retained tritium and activated argon available for release is assumed to encompass available volume and geometry of tritium-absorbing materials in the system.

13.1.1.2 Structures, Systems and Components Mitigation Assumptions

This section describes the structures, systems, and components (SSCs) that perform a function to mitigate the dose consequences of the MHA.

The reactor protection system (RPS) is credited with detecting the system disturbance and initiating a reactor trip, primary salt pump (PSP) trip, heat rejection blower trip, and a pebble extraction and insertion trip. The RPS initiates a reactor trip to shut down the reactor to limit the addition of heat to the system. The pebble extraction and insertion trip stops pebbles from moving into, out of, and through the core following the reactor trip to preclude any damage to pebbles from extraction faults during the event. The PSP trip facilitates the transition to decay heat removal through the decay heat removal system (DHRS) and precludes the potential for continuous entrainment of cover gas in the Flibe during the MHA. The DHRS continued operation ensures that an adequate amount of decay heat is removed from the system. The design bases of the RPS are discussed in Section 7.3. The RPS detection and actuation capabilities are automatic and do not rely on manual action to perform these functions.

The shutdown elements in the reactivity control and shutdown system (RCSS) are credited with shutting down the reactor upon receiving the reactor trip signal. The shutdown elements have sufficient worth to shut down the reactor and maintain long-term shutdown. The design bases of the RCSS shutdown function are provided in Section 4.2.2.

The DHRS is credited with removing an adequate amount of decay heat from the reactor to ensure that material design temperatures are not exceeded and no incremental fuel failures occur due to elevated temperatures. The DHRS does not rely on electrical power or manual actions to operate. The DHRS rejects heat to the ultimate heat sink passively. The design bases of the DHRS heat removal function are provided in Section 6.3.

The TRISO fuel layers of fuel in the reactor core and the Flibe are credited with the radionuclide retention properties described in Reference 1. For the Flibe to maintain the retention properties described in Reference 1, the integrity of the portion of the reactor vessel that ensures the pebbles in the reactor core remain covered by Flibe is credited with maintaining integrity under MHA conditions.

13.1.1.3 Transient Assumptions

This section describes the assumptions associated with the transient and its effects on radioactive MAR for release for the MHA.

The heating of the system is conservatively modeled with hypothetical temperature histories. The hypothetical temperature histories are selected to drive radionuclide movement and bound the system response to other postulated events. The hypothetical temperature history bounds the thermal impacts of conservative trip and actuation delays to account for uncertainty in the signal time associated with the RPS.

The TRISO particles have sufficient margin to prevent incremental barrier failures of TRISO layers of fuel in the reactor core following the temperature loads caused by the bounding MHA conditions. This includes the temperature excursion of the fuel and mechanical stresses due to insertion of the shutdown elements.

A conservative accumulation of tritium is assumed to be released using a hypothetical temperature profile to determine a conservative rate released during the MHA that bounds the radiological impact of tritium that could desorb from the system.

The amount of radioactive material released in the transient is maximized in the analysis by modeling bounding radionuclide evaporation characteristics. Radionuclide evaporation is maximized by assuming bounding mass transfer characteristics, including a bounding natural circulation gas flow rate above the Flibe-cover gas interface. No make-up or cleaning of the cover gas is modeled. Salt soluble fluoride concentrations in the Flibe remain well below solubility limits, consistent with Reference 1. Bounding thermodynamic properties of representative radionuclide species maximize radionuclide release from the Flibe, consistent with Reference 1. Partial pressures are adjusted by the concentration of soluble radionuclides in Flibe, consistent with Reference 1.

The atmospheric dispersion characteristics in the MHA analysis are based on site-specific meteorology values. The distance to the EAB is assumed to be 250 meters and the distance to the LPZ is provided in Chapter 2. As described in Reference 1, the building and reactor vessel head-space are not credited as confinement barriers and are modeled with artificially increased leakage rates. Henry’s Correlation and conservative building leakage rates are used to conservatively model aerosol deposition for Flibe aerosols, consistent with Reference 1. The methods in Reference 1 are used to calculate conservative near field atmospheric dispersion, resulting in conservative dose consequence values at the EAB and LPZ.

The methodology, inputs, and results of the MHA analysis are presented in Section 13.2.1.

13.1.2 Insertion of Excess Reactivity

There are various initiators that are postulated to result in an insertion of excess reactivity. These postulated events are bounded by the MHA, ensuring no insertion of excess reactivity results in unacceptable dose consequences. The limiting insertion of reactivity event is initiated by a control system error or an operator error that causes a continuous withdrawal of the highest worth control element at maximum control element drive speed. The reactivity insertion is detected by the RPS due to a high flux or a high coolant temperature, initiating control and shutdown elements insertion, fulfilling the reactivity control function. The decay heat removal system is already running because the limiting insertion of reactivity event occurs at an initial power above the DHRS threshold power discussed in Section 6.3, limiting reactor temperature and fulfilling the heat removal function.

This postulated insertion of excess reactivity bounds other insertion of reactivity events, including:

- Reactivity insertion events caused by fuel loading error (e.g., errors in rate of fresh fuel injection, incorrect order of fuel insertion)
- Reactivity insertion events with concurrent pump trip
- Reactivity insertion events with normal heat removal available
- Local phenomena leading to ramp insertion of reactivity
- Change in reactivity due to shifting of graphite reflector blocks
- Venting of gas bubbles accumulated in the active core
- Local phenomena leading to step insertion of reactivity
- Local negative reactivity anomaly (e.g., inadvertent single element insertion, cover gas injection)
- Reactivity insertion events during startup
- Increase in heat removal events (e.g., PSP overspeed, ISP overspeed, and heat rejection blower overspeed, spurious opening of a turbine bypass valve or steam safety valve, superheater shell leak, steam line break, spurious actuation of PHTS normal decay heat removal heat rejection radiator)

The following sections describe the key assumptions associated with the limiting postulated insertion of excess reactivity event. The quantitative values associated with these assumptions, as well as the methods used to evaluate the surrogate figures of merit that ensure the event consequences are bounded by the MHA, are provided in Reference 2.

13.1.2.1 Initial Conditions Assumptions

Normal operating parameters are provided in Section 4.1. Conservative initial values are assumed for each operating parameter to ensure a bounding result for the figures of merit that demonstrate the reactivity insertion event is bounded by the MHA.

The control element is assumed to be fully inserted as the initial condition for the event initiator.

13.1.2.2 Structures, Systems, and Components Mitigation Assumptions

This section describes the SSCs performing a function to mitigate the consequences of the event.

The RPS is credited with detecting the reactivity insertion and initiating a reactor trip after sensing a high neutron flux or a high coolant temperature. The DHRS is operating when the reactor is above a threshold power, as discussed in Section 6.3, and remains in an “always on” mode. The RPS initiates a reactor trip to shut down the reactor and limit the addition of heat to the system. The pebble handling and storage system (PHSS) trip stops pebble extraction and insertion following the reactor trip to preclude damage to pebbles from faults during the event. The DHRS remains active to ensure that an adequate amount of decay heat is removed from the system. The design bases of the RPS are discussed in Section 7.3. The RPS detection and actuation capabilities are automatic and do not rely on manual operator action to perform these functions.

The RCSS is credited with shutting down the reactor upon receiving the reactor trip signal. The shutdown and control elements are assumed to have sufficient worth to shut down the reactor and maintain long term shutdown. The design bases of the RCSS shutdown function are provided in Section 4.2.2.

Normal heat removal is expected to be available during this transient because those systems are not affected by the event initiator. However, normal heat removal is conservatively assumed to not be available during the transient. The DHRS and natural circulation within the reactor vessel are credited with removing an adequate amount of decay heat from the reactor to ensure that material design temperatures are not exceeded and no incremental fuel failures occur due to elevated temperatures. The DHRS does not rely on electrical power or manual operator actions to operate. Natural circulation within the core transfers heat from the fuel to the reactor vessel shell. Energy is transferred from the vessel shell to the DHRS and the DHRS rejects the heat to the ultimate heat sink passively. The design bases of natural circulation in the vessel and the DHRS heat removal function are provided in Section 4.3 and Section 6.3, respectively.

The TRISO fuel layers and the Flibe are credited with the radionuclide retention properties described in Reference 1. For the Flibe to maintain the retention properties described in Reference 1, the portion of the reactor vessel that ensures the pebbles in the core remain covered by Flibe is credited with maintaining integrity under the postulated event conditions.

13.1.2.3 Transient Assumptions

This section describes the assumptions associated with the transient and its effects on the surrogate figures of merit.

The postulated event analysis assumes conservative trip and actuation delays to account for uncertainty in the signal time associated with the RPS.

The amount of heat in the system is conservatively modeled in the postulated event by assuming bounding conditions for heat addition and heat removal. The transient initiator is a ramp insertion of reactivity that bounds the possible withdrawal speed and worth of a control element. Conservative values for reactivity feedback are assumed to limit the feedback available to reduce the severity of the reactivity insertion transient. Heat addition in the core during the transient is maximized by assuming a limiting element worth vs position curve that assumes the highest worth element is stuck out. The heat removal rate assumes a single failure in the DHRS by neglecting the heat removal capability provided by one of the four trains.

The key figures of merit for this event and the acceptance criteria are provided in Table 13.1-1.

A safe 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, where figure of merit temperatures are steadily decreasing during the mission time of DHRS.
- Flibe temperature inside the reactor vessel remains above the Flibe freezing temperature.

13.1.3 Salt Spills

There are various initiators that can result in a salt spill event. These postulated events are bounded by the MHA, ensuring no salt spill results in unacceptable dose consequences. The limiting salt spill postulated event initiates when a hypothetical double-ended guillotine break in the PHTS piping during normal operation causes a Flibe spill. The salt spill is detected by the RPS due to low reactor coolant level, which initiates control and shutdown elements insertion, fulfilling the reactivity control function. The decay heat removal system is already running because the limiting salt spill event occurs at an initial power above the DHRS threshold power discussed in Section 6.3, limiting reactor temperature and fulfilling the heat removal function. The RPS trips the PSP to limit the amount of spilled Flibe. The RPS trips the heat rejection blower to limit the amount of air ingress following postulated heat rejection radiator (HRR) tube breaks during low power operations. The postulated break causes negative pressure difference and allows air to enter the reactor system. In the reactor vessel head space, air reacts with Flibe to form volatile products and oxidizes portions of the structural graphite above the surface of the Flibe and the carbon matrix for pebbles in transit above the surface of the Flibe. Radionuclides from the coolant circulating activity in the broken pipe are released into the facility air when aerosols are generated from the coolant that exits the pipe. All the floor surfaces where Flibe may be spilled will have design features such as steel liners to prevent Flibe-concrete interaction, as described in Section 3.5. The spilled Flibe spreads on top of the liner and forms a Flibe pool. Radionuclides in the spilled Flibe are released through evaporation until the top surface of the Flibe pool is solidified.

The limiting salt spill postulated event bounds other salt spill events, including:

- Spurious draining and smaller leaks from the primary heat transport system
- Leaks from other Flibe containing systems and components (e.g., IMS fill/drain tank, IMS piping, chemistry control system piping)
- Leaks up to the hypothetical double-ended guillotine primary salt piping break size
- Mechanical impact or collision events involving Flibe-containing SSCs (except the vessel)
- Leaks from the intermediate heat transport system that contains a non-Flibe coolant, which may contain a non-zero amount of radionuclides
- Intermediate heat exchanger tube break or leak
- Single or multiple HRR tube(s) break

These following sections describe key assumptions associated with the limiting salt spill event. The quantitative values associated with these assumptions, as well as the methods used to evaluate the surrogate figures of merit that ensure the event consequences are bounded by the MHA are provided in Reference 2.

13.1.3.1 Initial Conditions Assumptions

Normal operating parameters are provided in Section 4.1. Conservative initial values are assumed for each operating parameter to ensure a bounding result for the figures of merit that demonstrate the event is bounded by the MHA.

A hypothetical double-ended guillotine break in the PHTS hot leg piping is assumed as the event initiator. The initial Flibe conditions are discussed in Section 5.1.

13.1.3.2 Structures Systems and Components Mitigation Assumptions

This section describes the SSCs performing a function to mitigate the consequences of the event.

The RPS is credited with detecting the break on low reactor coolant level and initiating a reactor trip, PSP trip, heat rejection blower trip, and the PHSS trip. The DHRS is operating when the reactor is above a threshold power, as discussed in Section 6.3, and remains in an “always on” mode. The RPS initiates a reactor trip to shut down the reactor and limits the addition of heat to the system. The RPS trips the PSP limit the amount of spilled Flibe. The heat rejection blower is tripped to limit the amount of air ingress following postulated HRR tube breaks. The PHSS trip stops pebble extraction and insertion following the reactor trip to preclude any damage to pebbles from faults during the event. The DHRS remains active to ensure that an adequate amount of decay heat is removed from the system. The design bases of the RPS are discussed in Section 7.3. The RPS detection and actuation capabilities are automatic and do not rely on manual operator action to perform these functions.

The RCSS is credited with shutting down the reactor upon receiving the reactor trip signal. The shutdown elements are assumed to have sufficient worth to shut down the reactor and maintain long term shutdown. The design bases of the RCSS shutdown function are provided in Section 4.2.2.

The anti-siphon design features of the PHTS (see Section 5.1) are credited with limiting the amount of Flibe available to spill out of the break. The design features below the Flibe piping (such as steel liners, catch pans, or troughs) are credited with preventing Flibe concrete interaction, as discussed in Section 3.5.

The DHRS and natural circulation within the reactor vessel are credited with removing an adequate amount of decay heat from the reactor to ensure that material design temperatures are not exceeded and no incremental fuel failures occur due to elevated temperatures. The DHRS does not rely on electrical power or manual operator actions to operate. Natural circulation within the core transfers heat from the fuel to the reactor vessel shell. Energy is transferred from the vessel shell to the DHRS, and the DHRS rejects the heat to the ultimate heat sink passively. The design bases of natural circulation in the vessel and the DHRS heat removal function are provided in Section 4.3 and Section 6.3, respectively.

The TRISO fuel layers and the Flibe are credited with the radionuclide retention properties for fuel in the core as described in Reference 1. For the Flibe to maintain the retention properties described in Reference 1, the integrity of the portion of the reactor vessel that ensures the pebbles in the core remain covered by Flibe is credited with maintaining integrity under the postulated event conditions.

13.1.3.3 Transient Assumptions

This section describes the assumptions associated with the transient and its effects on the surrogate figures of merit.

The postulated event analysis assumes conservative trip and actuation delays to account for uncertainty in the signal time associated with the RPS.

The amount of heat in the system is conservatively modeled in the postulated event by assuming bounding conditions for heat addition and heat removal. Conservative values for reactivity feedback are assumed to limit the feedback available to reduce the severity of the event prior to reactor trip. Heat addition in the core during the transient is maximized by assuming a limiting element worth versus position curve that assumes the highest worth element is stuck out. The heat removal rate assumes a single failure in the DHRS by neglecting the heat removal capability provided by one of four trains.

The analysis of the spilled Flibe uses the methodology described in Reference 2, which assumes a conservative aerosolization rate of the Flibe as it spills onto the floor.

The key figures of merit for this event and the acceptance criteria are provided in Table 13.1-1.

A safe 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, where figure of merit temperatures are steadily decreasing during the mission time of the decay heat removal system.
- Flibe temperature inside the reactor vessel remains above the Flibe freezing temperature.
- Flibe stops spilling out of the break and Flibe pool solidifies

13.1.4 Loss of Forced Circulation

There are various initiators for a postulated event involving a loss of forced circulation. The limiting postulated loss of forced circulation event initiates with the seizure of the PSP. The reduced flow is detected directly or indirectly by the RPS, which initiates control and shutdown elements insertion, fulfilling the reactivity control function. The decay heat removal system is already running because the limiting loss of forced circulation event occurs at an initial power above the DHRS threshold power discussed in Section 6.3, limiting reactor temperature and fulfilling the heat removal function.

The limiting loss of circulation postulated event bounds other loss of circulation events, including:

- Blockage of flow path external to the reactor vessel in the primary heat transport system
- Spurious pump trip signal

- Shaft fracture
- Bearing failure
- Pump control system errors
- Supply breaker spurious opening
- Loss of net-positive suction head (e.g., pump overspeed, low level)
- Loss of normal electrical power
- Flibe freezing inside HRR
- Loss of normal heat sink (e.g., turbine trip, ISP failure, superheater tube rupture)

The following sections describe the key assumptions associated with the limiting loss of forced circulation. The quantitative values associated with these assumptions, as well as the methods used to evaluate the surrogate figures of merit that ensure the event consequences are bounded by the MHA are provided in Reference 2.

13.1.4.1 Initial Conditions Assumptions

Normal operating parameters are provided in Section 4.1. Conservative initial values are assumed for each operating parameter to ensure a bounding result for the figures of merit that demonstrate the event is bounded by the MHA.

The loss of forced circulation event initiator is assumed to be a pump seizure, which disables the PSP.

13.1.4.2 Structures Systems and Components Mitigation Assumptions

This section describes the SSCs performing a function to mitigate the consequences of the event.

The RPS is credited with initiating a reactor trip. The PHSS is tripped to prevent damage to fuel in transit. The DHRS is operating when the reactor is above a threshold power, as discussed in Section 6.3, and remains in an “always on” mode. The RPS initiates a reactor trip to shut down the reactor and limits the addition of heat to the system. The PHSS trip stops pebble extraction and insertion following the reactor trip to preclude any damage to pebbles from faults during the event. The DHRS remains active to ensure that an adequate amount of decay heat is removed from the system. The design bases of the RPS are discussed in Section 7.3. The RPS detection and actuation capabilities are automatic and do not rely on manual operator action to perform these functions.

The shutdown elements in the RCSS are credited with shutting down the reactor upon receiving the reactor trip signal. The shutdown elements are assumed to have sufficient worth to shut down the reactor and maintain long term shutdown. The design bases of the RCSS shutdown function are provided in Section 4.2.2.

The DHRS and natural circulation within the reactor vessel are credited with removing an adequate amount of decay heat from the reactor to ensure that material design temperatures are not exceeded and no incremental fuel failures occur due to elevated temperatures. The DHRS does not rely on electrical power or manual operator actions to operate. Natural circulation within the core transfers heat from the fuel to the reactor vessel shell. Energy is transferred from the vessel shell to the DHRS, and the DHRS rejects the heat to the ultimate heat sink passively. The design bases of natural circulation in the vessel shell and the DHRS heat removal function are provided in Section 4.3 and Section 6.3, respectively.

The TRISO fuel layers and the Flibe are credited with the radionuclide retention properties for fuel in the reactor core, as described in Reference 1. For the Flibe to maintain the retention properties described in Reference 1, the integrity of the portion of the reactor vessel that ensures the pebbles in the core remain covered by Flibe is credited with maintaining integrity under the postulated event conditions.

13.1.4.3 Transient Assumptions

This section describes the assumptions associated with the transient and its effects on the surrogate figures of merit.

The postulated event analysis assumes conservative trip and actuation delays to account for uncertainty in the signal time associated with the RPS.

The amount of heat in the system is conservatively modeled in the postulated event by assuming bounding conditions for heat addition and heat removal. Conservative values for reactivity feedback are assumed to limit the feedback available prior to reactor trip. Heat addition in the core during the transient is maximized by assuming a limiting element worth versus position curve that assumes the highest worth element is stuck out. The heat removal rate assumes a single failure in the DHRS by neglecting the heat removal capability provided by one of four trains.

The key figures of merit for this event and the acceptance criteria are provided in Table 13.1-1.

A safe 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, where figure of merit temperatures are steadily decreasing during the mission time of the decay heat removal system.
- Flibe temperature inside the reactor vessel remains above the Flibe freezing temperature.

13.1.5 Mishandling or Malfunction of Pebble Handling and Storage System

There are various initiators for a postulated event involving a PHSS malfunction. The limiting postulated event for a PHSS malfunction is a break in a transfer line when pebbles are removed from the core, resulting in a spill of pebbles within the transfer line to the room. This condition is detected by the RPS, which trips the PHSS to stop pebble movement. For the spilled pebbles, the reactivity control function is fulfilled by the low fissile inventory of the pebbles, which precludes a criticality concern, while heat transfer mechanisms within the room fulfills the heat removal function. The structural integrity of the pebbles maintains the confinement function. For the pebbles remaining in the PHSS, the reactivity control, heat removal, and confinement functions continue to be fulfilled by the system design resulting in a safe and stable state. Air ingress into the PHSS and reactor cover gas region occurs through the break. The heat up of the pebbles in the PHSS system mobilizes the Flibe accumulated on the piping.

The limiting PHSS malfunction event bounds other PHSS malfunctions, including:

- Transfer line break when pebbles are inserted into empty core
- Transfer line break when pebbles are inserted into the core at power
- Transfer line break when pebbles are transferred to storage canisters
- Mishandling of fuel outside the reactor (e.g., containment box, at the material balance areas and key measure points)

The following sections describe the key assumptions associated with the limiting PHSS malfunction event. The quantitative values associated with these assumptions, as well as the methods used to evaluate the surrogate figures of merit that ensure the event consequences are bounded by the MHA are provided in Reference 2.

13.1.5.1 Initial Conditions Assumptions

Conservative initial values are assumed for the amounts of Flibe, tritium, and graphite dust available to be mobilized within the PHSS.

The event initiator is assumed to be a break in a fuel transfer line during extraction, allowing pebbles to spill out of the system and onto the floor.

13.1.5.2 Structures Systems and Components Mitigation Assumptions

This section describes the SSCs performing a function to mitigate the consequences of the event.

The RPS is credited with initiating a PHSS trip. The PHSS trip stops pebble extraction and insertion following the reactor trip to prevent additional pebbles spilling out of the break and to preclude any damage to pebbles from faults during the event. The design bases of the RPS are discussed in Section 7.3. The RPS detection and actuation capabilities are automatic and do not rely on manual action to perform these functions.

The TRISO fuel layers and the Flibe are credited with the radionuclide retention properties described in Reference 1. The structural integrity of the fuel pebbles is credited when the spilled pebbles hit the floor to maintain the TRISO confinement function. The low fissile inventory of the pebbles precludes criticality concerns of the spilled pebbles.

13.1.5.3 Transient Assumptions

This section describes the assumptions associated with the transient and its effects on the surrogate figures of merit.

The postulated event analysis assumes conservative trip and actuation delays to account for uncertainty in the signal time associated with the RPS.

The amount of heat in the pebbles is conservatively modeled.

The key figures of merit for this event and the acceptance criteria are provided in Table 13.1-1.

A safe state is established when:

- The movement of pebbles outside of the core has stopped and criticality safety is assured.
- Decay heat is being removed from pebbles outside of the core and long-term cooling is assured, where figure of merit temperatures are steadily decreasing.

13.1.6 Radioactive Release from a Subsystem or Component

A radioactive release from a subsystem or component could result from the failure of a system or component containing radioactive material. However, the limiting event for this category is assumed to be a seismic event that results in the failure of all systems for a single unit (including shared systems) containing radioactive material that are not qualified to maintain structural integrity in a design basis earthquake. The only figure of merit for this event is the amount of radioactive material contained in subsystems and components. To ensure that this event group is bounded by the MHA, there is a design requirement on the amount of MAR for release in subsystems and components to remain below the amount of MAR for release assumed in the MHA. The systems expected to accumulate radionuclides as a function of operation include:

- Tritium management system
- Inert gas system
- Chemistry control system (including filters)
- Inventory management system
- Intermediate heat transport system
- Power generation systems

The tritium storage strategy discussed in Section 9.1.3 ensures that the amount of MAR accumulated by this system remains below the amount of tritium assumed to be released in the MHA. The amount of MAR in subsystems and components is limited to an upper bound limit such that the total amount of materials at risk released is bounded by the amount released during the MHA.

13.1.7 Not Used

13.1.8 General Challenges to Normal Operation

This category of events includes challenges to normal operation not covered by another event category that requires an automatic or manual shutdown of the plant. Disturbances, including an inadvertent operator action, are detected directly or indirectly by the RPS, which initiates control and shutdown elements insertion, fulfilling the reactivity control function. The highest worth element is assumed to be stuck out and does not insert. The DHRS performs its function to limit reactor temperature and fulfill the heat removal function.

Grouped events include spurious trips due to control system anomalies, operator errors, and equipment failures. This event group also includes scenarios where operators choose to manually shutdown the plant. Also included are faults in the reactivity control and shutdown system, electrical system, intermediate heat transport system, and other plant systems that would challenge normal operations.

This group also contains inert gas system disturbances, and instrumentation and control system faults. This event group relies upon the reactor protection system and is bounded by the loss of forced circulation postulated event.

13.1.9 Internal and External Hazard Events

The portions of the design relied upon to perform safety functions are protected from the internal and external hazard levels defined in Chapter 2. Events in this category are bounded by or considered as initiators in other event categories. The internal hazard events in the design basis include:

- Internal fire
- Internal water flood
- Turbine missile
- High energy steam line break

The external hazard events in the design basis include:

- Seismic event
- High wind event
- Toxic release
- Mechanical impact or collision with SSCs
- External flood

Engineered safety features contained within areas protected from or able to withstand the intensity of the hazard loading for hazard events initiated outside those areas (e.g., fire) maintain their capability to bring the plant to a safe state following a postulated event. The SSCs within those areas are designed to withstand an upper bound hazard loading intensity associated with the area (e.g., SSCs can withstand an upper bound heat load and the associated area is equipped with fire detection and suppression systems to limit the heat load). Chapter 3 discusses the civil and structural design considerations of the reactor building that protect SSCs associated with engineered safety features from internal and external hazard levels.

A turbine missile could be generated due to a postulated turbine generator failure. Due to the favorable orientation of the turbine generator with respect to the reactor building, SSCs associated with engineered safety features are not affected by a potential turbine missile to the extent that they could not perform their safety functions.

For SSCs not protected with such an area, the amount of materials at risk are assumed to be limited to an upper bound limit such that the amount of radioactive material released is bounded by the amount released during the MHA. Releases from these SSCs are considered in Section 13.1.6.

During the seismic event, the packing fraction of the pebble bed would increase due to shaking of the pebble bed, and the graphite reflector blocks would shift. This results in an increase in reactivity, causing an increase in fuel temperature. The increase in reactivity due to increase in packing fraction of the pebble bed and maximum displacement of graphite reflector blocks during a seismic event is bounded by the reactivity insertion event where the control element is inadvertently withdrawn. Insertion of excess reactivity events are described in Section 13.1.2.

Mechanical aerosols could also be generated due to splashing of Flibe in the reactor during a seismic event. The amount of aerosols generated during a seismic event is bounded by the amount of aerosols generated by the salt spill event where a pipe breaks.

A break in a high energy steam line or superheater could occur due to a failure of the steam system. Physical separation of the power generation systems from safety-related SSCs and the design of the safety-related portion of the reactor building ensures that a high energy break will not prevent safety-related SSCs from performing their safety functions. The potential reactivity insertion caused by an increase in heat removal due to a steam line break is considered in Section 13.1.2.

13.1.10 Prevented Events

This section describes the events prevented by design. The justification for excluding these events from the design basis is provided with references to the relevant design information.

13.1.10.1 Recriticality or Reactor Shutdown System Failure

In postulated events that require a reactor trip, the reactor shutdown system (the safety-related portion of the RCSS), is relied upon to shut down the reactor and maintain shutdown margin. Reactor shutdown system (RSS) failure events are excluded from the design basis. Events that would result in a recriticality event are also excluded from the design basis. The RCSS is designed (described in Section 4.2.2) with sufficient independence, diversity, and redundancy from detection and actuation to element insertion to ensure reactor shutdown when necessary. The shutdown margin is maintained for all postulated event conditions to ensure there is no recriticality after the RCSS has initiated shutdown, as described in Section 4.5. Additionally, the graphite reflector blocks are designed to maintain structural integrity and ensure misalignments do not prevent the insertion path of the shutdown elements, as discussed in Section 4.3.

13.1.10.2 Degraded Heat Removal or Uncooled Events

In postulated events where the normal heat removal is not available, natural circulation in the reactor vessel and the heat removal function of the DHRS are relied upon to remove heat from the reactor core. Degraded heat removal or uncooled events are excluded from the design basis. The initiation of natural circulation is completely passive, and the design features, including the structural integrity of the reactor vessel internals, that ensure a continued natural circulation flow path are discussed in Section 4.6. The DHRS is aligned and operating when the reactor power is above a threshold power and remains in this state as described in Section 6.3, precluding the need for an actuation to occur for the DHRS to remove heat during a postulated event. The DHRS design includes sufficient redundancy to perform its safety function assuming the loss of a single train, as discussed in Section 6.3.

13.1.10.3 Flibe Spill Beyond Maximum Volume Assumed in Postulated Salt Spills

In the salt spill postulated event category, an upper bound volume of Flibe is assumed to spill out of the PHTS onto the floor. A volume of Flibe spilling out of the system beyond the amount assumed in the bounding salt spill event is excluded from the design basis. There are several design features ensuring the amount of Flibe available to spill is limited to an upper bound value. The PHTS is designed with anti-siphon features discussed in Section 5.1. These features are designed to passively break the siphon in the event of a break. The PSP also trips to allow the primary system to depressurize. The reliability of the RPS, which trips the PSP, ISP, and heat rejection blower in the event of a salt spill, is discussed in Section 7.3. The reactor vessel shell also maintains integrity in postulated events to ensure the fuel in the core remains covered with Flibe. The reactor vessel shell design features that prevent leakage are discussed in Section 4.3.

13.1.10.4 In-Service TRISO Failure Rates and Burnups Above Assumptions in Postulated Events

The in-service fuel failure rates and the burnup of pebbles assumed in the postulated events are based on the fuel qualification specifications in Section 4.2.1. In-service TRISO failure rates above the rate assumed in postulated events are excluded from the design basis. The insertion of pebbles with a burnup higher than the fuel qualification envelope is excluded from the design basis. As described in Section 7.3, the RPS includes a function to stop the pebble insertion and extraction functions to ensure pebbles are not damaged in faults occurring after an event initiation. The fuel qualification program includes testing, inspection, and surveillance to ensure the fuel operating envelope is within the fuel qualification envelope. Inspection and surveillance of the fuel in service is performed in the PHSS as discussed in Section 9.3.

13.1.10.5 Significant Air Ingress Into PHTS

Events where significant quantities of forced air are entrained in the PHTS coolant during normal operation are excluded from the design basis. Operational controls are expected to monitor the quantity of air within the PHTS to prevent accumulating significant quantities. Chapter 14 discusses the expected coolant systems technical specifications that monitor significant air ingress.

Events where significant quantities of forced air enter the PHTS following postulated HRR tube break events are also excluded from the design basis. Chapter 5 discusses the design features of the HRR that limits the quantities of air ingress during salt spill transients.

The effects of non-forced air ingress on reactor vessel and vessel internal components will remain bounded by the materials qualification testing programs for at least seven days during air ingress events as described by Section 4.3. Beyond seven days, defense in depth strategies include: implementing repairs on damaged SSCs, replenishing the argon supply, and removal of fuel from the vessel (full core offload capability discussed in Section 9.3).

13.1.10.6 DHRS Reactor Cavity Flooding

The DHRS is a water-based system that removes heat from the reactor vessel shell. Events where the water from the DHRS leaks into the reactor cavity in quantities significant enough to wet the reactor vessel are excluded from the design basis. Leak prevention, including double walled components and leak detection, for the DHRS is described in Section 6.3.

13.1.10.7 Insertion of Excess Reactivity Beyond Rate Assumed in Postulated Events

The insertion of excess reactivity postulated event category includes a limiting reactivity insertion rate based on the maximum control element drive withdrawal rate. Multiple control elements moving simultaneously is excluded from the design basis. Control element movement is limited to one element at a time, as described in Section 7.2. A control element withdrawing faster than the limit is excluded from the design basis. The maximum drive withdrawal speed is limited by the drive hardware, as described in Section 4.2.2. A rapid control element ejection is excluded from the design basis because the reactor operates at low pressures.

The insertion of reactivity due to an overcooling event is also bounded by the limiting reactivity insertion rate. Pump overspeed from the PSP, ISP, or heat rejection blower (which would increase core cooling) is limited by the programmed normal operating range discussed in Section 7.2.

13.1.10.8 Criticality Occurrence External to Reactor Core

Pebbles outside of the reactor core are contained in the PHSS. The PHSS includes pebbles in transit during handling, in storage, and in a transport configuration. The PHSS is designed to preclude criticality assuming postulated event conditions using design features that maintain a non-critical geometry of pebbles in each of these areas. The design features of PHSS preventing criticality are described in Section 9.3.

13.1.10.9 Excessive Radionuclide Release from Flibe

The postulated events assume a release of radionuclides from the free surfaces of Flibe. The assumed release of radionuclides from Flibe could be affected by the characteristics of the cover gas such as a higher pressure affecting the cover gas flow or the purity of the cover gas affecting the radionuclides available for release. The cover gas is maintained by the inert gas system, described in Section 9.1.2.

13.1.10.10 Internal or External Events Interfering with SSCs

SSCs that perform safety functions are located in a portion of the reactor building designed to preclude damage from both internal and external hazards that could interfere with those functions. Additionally, SSCs containing Flibe are protected from internal floods to preclude the potential for Flibe-water interactions. Components containing Flibe will be located in areas that have design features such as steel liners to prevent Flibe-concrete interaction. Components containing BeNaF that are located where BeNaF-concrete interactions could prevent safety-related SSCs from performing their function will be in locations that have design features such as steel liners to prevent BeNaF-concrete interactions. As described in Section 13.1.9, turbine blade missiles could be generated due to a postulated turbine failure. However, those missiles will not affect safety-related SSCs to the extent that they could not perform their safety function due to the favorable orientation of the turbine. Piping associated with the steam and condensate and feedwater systems is routed such that postulated failures do not adversely affect safety-related SSCs.

The failure of safety functions due to internal or external hazards is excluded from the design basis. The reactor building design features, including flood prevention, are described in Section 3.5. The fire protection system is described in Section 9.4. The power generation systems are described in Section 9.9.

13.1.10.11 IHX Failure Due to Superheater Tube Rupture or Leak

A superheater tube rupture could lead to over-pressurization of the intermediate heat transport system. The safety-related pressure relief feature on the IHTS precludes failure of the IHX due to steam over-pressurization. The pressure relief feature is located outside of the safety-related portion of the reactor building to prevent adverse effects on safety-related SSCs. The design of the IHTS is described in Section 5.2.

Operational controls are expected to ensure postulated leaks in the superheater do not lead to corrosion of the IHX beyond an allowable amount. Superheater leaks are expected to be limited by technical specification, as discussed in Chapter 14.

13.2 ACCIDENT ANALYSIS AND DETERMINATION OF CONSEQUENCES

13.2.1 Maximum Hypothetical Accident

13.2.1.1 Methodology and Inputs

The calculation of the dose consequences of the MHA uses the source term methods for design basis accidents presented in Reference 1. Section 13.1.1 provides the MHA narrative and assumptions. This section provides a high level summary of the methods used and the inputs to the calculation.

The evaluation of the MHA dose consequences first identifies and accounts for the sources of MAR and the barriers to release. Each barrier is then evaluated for a release fraction to provide dose consequences at the exclusion area and low population zone boundaries.

The four sources of MAR and the associated barriers to release in the MHA:

- TRISO fuel in the reactor core
    - Barriers: TRISO layers, Flibe, and gas space
- Circulating activity
    - Barriers: Flibe and gas space
- Structural MAR
    - Tritium retained by graphite and in Flibe
        - Barriers: Graphite grains (for non-Flibe tritium) and gas space
    - Argon-41 retained in closed graphite pores
        - Barriers: Graphite pores and Gas space

Section 13.1.1 describes several non-physical conditions that are hypothesized to ensure a bounding MHA:

- Pre-transient diffusion of radionuclides from the fuel in the reactor core is neglected: This conservatism is achieved in the evaluation by assuming that the full radionuclide inventory of the fuel is available for release at the initiation of the MHA. The circulating activity is still assumed to be at an upper bound level. Therefore, any MAR originating in the fuel that contributes to the circulating activity is effectively double counted.
- Hypothetical temperature histories are applied to the transient: the hypothetical temperature histories applied to the MHA is provided in Figure 13.2-1. These temperatures set an upper limit for the figure of merit temperatures in the postulated events.
- The gas space is not credited for confinement of the radionuclides that release from the Flibe free surface: radionuclide transport in the gas space barrier is modeled using the conservative building transport and off-site dispersion methods described in Reference 1.
- Conservative, unfiltered, ground level releases: the gas space transport evaluation assumes a conservative leakage rate for the reactor building that releases the entire volume within a 2 hour window to avoid crediting the building as a confinement structure. The dispersion evaluation assumes no radionuclides are filtered after the building transport is evaluated to avoid taking credit for any radionuclide filtering that could occur in the HVAC system.
- Initial tritium inventories are calculated for an assumed 50MWth core that operates with an assumed 100% capacity factor over ten years. Lower operating powers result in a lower tritium production rate and lower capacity factors allow for the graphite grains to experience time periods of tritium desorption instead of sorption.
- A bounding vessel void fraction of 0.1 is assumed to facilitate the release of low volatility species in the vessel via bubble burst.

Quantification of MAR Sources

The fuel MAR consists of radionuclides produced by normal operation. A Serpent2 evaluation provides the fuel inventory. The fuel MAR is assumed to transport in the radionuclide groups described in Reference 1.

A bounding value of circulating activity is assumed for Flibe MAR in the analysis. The Flibe MAR is assumed to transport in the groups described in Reference 1.

The quantity of retained tritium is conservatively bound within graphite and structures over 10 years of operation. The tritium speciation is simplified to fully tritium fluoride for an oxidizing salt. A fully molecular tritium case for a reducing salt is calculated, but the fully tritium fluoride case is used because it leads to a higher graphite inventory and higher total release of tritium. The tritium fluoride is assumed to be retained by the graphite, but does not permeate, and its evolution to off-gas is neglected. The steady-state tritium fluoride distribution is determined by mass transfer in Flibe, where the graphite is treated as a perfect absorber. Distribution fractions to each region are calculated by mass transfer coefficient multiplied by the surface area. The mass transfer coefficients are calculated by the correlations in Reference 1. Once the transient begins, the concentration of tritium in the Flibe is reduced to zero and thus the concentration gradient reverses, moving tritium out of grains and back into the Flibe. Tritium release fractions are calculated using a numerical solution to diffusion equations. The tritium transport through graphite pores is assumed to be instantaneous, and all graphite grains are exposed to the same tritium uptake conditions. Strong tritium trapping sites are neglected to bound release fractions.

The Ar-41 buildup and release models predict the diffusion of argon cover gas into graphite closed pores which are then activated in the core and reflector regions. Graphite used for the reflector as well as carbon matrix used for fuel and moderator pebbles are porous materials. Small entrance pore sizes of the graphite prevent salt intrusion into the bulk material, and the volume of pores is available for occupancy by the cover gas. The closed porous volume of graphite and carbon matrix is occupied by the cover gas for the reactor. Cover gas also diffuses through the Flibe and enters graphite closed pores during reactor operations since the argon cover gas has small, but non-zero solubility in Flibe. The inventory of Ar-41 is “puff” released directly into the gas space.

Radionuclide Transport in Fuel

The grouped fuel MAR diffuses through the TRISO layers, driven by the hypothetical temperature history in Figure 13.2-1. As discussed in Reference 1, the transport of mobile fission products through the TRISO fuel particle is modeled by Fick’s laws of diffusion.

No further generation of radionuclides occurs after the reactor trip. Additionally, no radioactive decay is modeled in the mass diffusion equations. The short-time approximation of the Booth solution is used to determine the fractional release of fission product from the kernel for conditions where $\tau \leq 0.155$ (no power, no further generation of nuclides) (“MELCOR Fission Product Release Model for HTGRs”, Reference 3):

$$RF(t = T) = 6\sqrt{\frac{\tau}{\pi}} - 3\tau$$

where:

$RF(T)$ = release fraction of fission product up to time *t=T*

τ = reduced diffusion coefficient = $\frac{DT}{a^2}$ (unit-less) *a* = radius of equivalent sphere (m)

D = diffusivity coefficient of the representative radionuclide (m^2/s) (consistent with the values from Reference 1)

t = time (s)

For conditions where $\tau \geq 0.155$ and radioactive decay are ignored, the long time approximation for release fraction for the kernel is modeled as ("Modelling of Short-Lived Fission Product Release Behavior During Annealing Conditions," Reference 4):

$$RF(t = T) = 1 - \frac{6}{\pi^2} e^{-\pi^2 \tau}$$

The short time approximation for fractional release of a coating layer is ("High Temperature Gas Cooled Reactor Fuels and Materials," Reference 5):

$$RF(T) \approx \frac{24\gamma(1+\gamma)}{\sqrt{\pi}} e^{-\frac{1}{4\tau}} \tau^{1.5} [1 - 6\tau(1+\gamma) + 12\tau^2(5 + 6\gamma + 6\gamma^2)]$$

where:

$RF(T)$ = release fraction of fission product up to time $t = T$

γ = ratio of layer thickness d to the inner radius r_i of the layer = $\frac{d}{r_i}$ (unit-less)

τ = reduced diffusion coefficient = $\frac{DT}{d^2}$ (unit-less)

D = diffusivity of the diffusing species in the diffusing medium (m^2/s) (consistent with the values from Reference 1)

t = time (s)

d = thickness of the coating layer (m)

This short time approximation is applied to conditions where $\tau \leq 0.2$. When $\tau > 0.2$, the following long time approximation equation is used to calculate the fractional release for a coating layer (Reference 5):

$$RF(t) \cong 1 - (1 + \frac{\gamma}{2}) e^{-3\gamma\tau}$$

Radionuclide diffusion through TRISO layers is employed for fuel release assuming no depletion of the radionuclide inventory due to operation time. In this bounding model, radionuclides are assumed to continuously challenge each barrier independent of quantity of radionuclides actually challenging a barrier at any given time. For example, radionuclides that reach the outer pyrolytic carbon (OPyC) layer at day five of the simulation would instantly be released from the OPyC layer with a fraction equivalent to radionuclides that have been diffusing through the barrier since the initiation of the transient. The release fraction (RF) of compromised layers is conservatively set to 1.0.

$$RF_{fuel}^{i}(t = T) = \prod_{j=kernel}^{OPyC} RF_j^i(t = T)$$

<u>Structural MAR Transport from Structural Materials</u>

The tritium is released from the system in the following batches which roughly corresponds to the X/Q dispersion bins:

1) "Puff" release of all tritium in both the Flibe and pebble carbon matrix, due to the high diffusivities at the prescribed pebble carbon matrix temperatures, at the beginning of the transient
2) A bounding diffusion model estimates the fraction of tritium that transports out of reflector graphite grains from
   i. 0 to 10 min
   ii. 10 min to 2 hours
   iii. 2 hours to 8 hours
   iv. 8 hours to 14 hours
   v. 14 hours to 24 hours
3) Remaining tritium in the system transports out of the system by an assumed puff release 24 hours into the transient

All Ar-41 predicted to be contained within graphite structures at the initiation of the transient is puff-released into the gas space.

<u>Transport of MAR from Flibe to the Gas Space</u>

The two release mechanisms for MAR in the circulating Flibe are bubble burst from entrained cover gas in the vessel coolant and evaporation driven by the MHA temperature curve. Bubble burst occurs before transient diffusion can occur from the fuel into the Flibe but evaporation mobilizes both circulating activity and MAR that has diffused from the fuel into the Flibe.

For a two-phase flow, the void fraction of the flow is designated by α. The volumetric flow rate of gas $Q_{g,2p}$ is related to the two-phase mass flow rate of Flibe $W_{f,2p}$ by the following expression:

$$Q_{g,2p} = \frac{W_{f,2p}}{\rho_f}\frac{\alpha}{1-\alpha}$$

The aerosol generation rate $W_{a,2p}$ is obtained through the volumetric ratio E_a(the ratio of the volume of particles generated by a single bubble bursting to the volume of the bubble) as:

$$W_{a,2p} = \rho_f E_a Q_{g,2p} = W_{f,2p} E_a \frac{\alpha}{1-\alpha}$$

The bounding value of E_a = 2.1 x 10^{-6} is chosen for the Flibe-argon system, consistent with Reference 1.

For conservatism, no deposition is assumed during the aerosol generation process. The total mass of aerosol is given by:

$$m_{a,2p} = \int_0^t W_{a,2p} dt = m_{f,2p} E_a \frac{\alpha}{1-\alpha}$$

where: $m_{f,2p}$ is the mass of two-phase Flibe. Thus, the aerosol release fraction from bubble burst is calculated using:

$$ARF_{2p} = E_a \frac{\alpha}{1-\alpha}$$

The release rates for gases and high volatility noble metals in the circulating activity are conservatively bounded by instantaneous (or "puff") releases at the beginning of the transient. Other radionuclides are assumed to be released from the Flibe at a rate determined by the general evaporation law, as described in Reference 1. Conservative mass transfer coefficients that neglect liquid side mass transfer resistance are used.

The radionuclides evaporated from the Flibe free surface are separated into the following release inventories:

1) "Puff" release of dissolved noble gases and bubble burst Flibe aerosols at the beginning of the transient;
2) One linear release for evaporation of radionuclides over the first 10 min temperature interval corresponding to pre-scrammed fuel temperature
3) One linear release for evaporation of radionuclides over the next 110 min temperature interval;
4) One linear release for evaporation of radionuclides over the next 70 hour release interval;
5) One linear release per day for the next seven days for the reactor cool down period; and
6) One final linear release over the remaining 20 days.

Gas Space

The gas space transport evaluation is divided into two models: building transport and atmospheric dispersion. The methodology for Design Basis Accidents in Reference 1 is used to evaluate the gas space transport. The $\frac{\chi}{Q}$ values are calculated as described in Section 2.3.4 for an assumed exclusion area boundary (EAB) distance of 250 m and a low population zone (LPZ) distance of 800 m as provided in Section 2.1.1.2. The X/Q values are provided in Table 13.2-1.

13.2.1.2 Results

The dose consequences of the MHA are provided in Table 13.2-2. The dose consequence results meet the site dose limits in 10 CFR 100.11(a)(1-2) at the EAB and LPZ with significant margin.

13.2.2 Postulated Event Methodology and Sample Results

The evaluation models and methodologies used to analyze the postulated events described in Section 13.1 are detailed in the Postulated Event Methodology Technical Report (Reference 2). The methodologies include the rationale for the figures or merit and the associated acceptance criteria (provided in Table 13.1-1) for each postulated event category. The figures of merit are figures of merit that when analyzed against the acceptance criteria ensure that the postulated events result in doses bounded by the MHA.

13.3 REFERENCES

1. Kairos Power LLC, "KP-FHR Mechanistic Source Term Methodology Topical Report," KP-TR-012-P-A. May 2022.
2. Kairos Power LLC, "Postulated Event Methodology Technical Report," KP-TR-022-P, Revision 0. June 2023.
3. "MELCOR Fission Product Release Model for HTGRs," Sandia National Laboratories, 2010.
4. B. J. Lewis, D. Evens, F. C. Iglesias, and Y. Liu, "Modelling of Short-Lived Fission Product Release Behavior During Annealing Conditions," Journal of Nuclear Materials, vol. 238, pp. 183–188, 1996.
5. "High Temperature Gas Cooled Reactor Fuels and Materials," INTERNATIONAL ATOMIC ENERGY AGENCY, Vienna, IAEA-TECDOC-1645, 2010.

Table 13.1-1: Acceptance Criteria for Figures of Merit

Figure of Merit	Acceptance Criterion	Applicable Events
Peak TRISO temperature-time	Generally bounded by temperature-time curves derived from the assumed MHA fuel temperature-time curve	Salt Spill, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break, Seismic
TRISO failure probability	Negligible TRISO fuel failure probability	Salt Spill, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break
Peak Flibe-cover gas interfacial temperature	Generally bounded by temperature-time curves derived from the assumed MHA Flibe-cover gas interfacial temperature-time curve	Salt Spill, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break
Peak vessel and core barrel temperatures	Bounded by both the maximum allowable temperature derived to limit excessive creep deformation and damage accumulation and by 750°C (highest vessel temperature covered by qualification description in Section 4.3.3)	Salt Spill, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break
Minimum reactor vessel inner surface temperature	Above Flibe melting temperature	Loss of Forced Circulation (overcooling)
Airborne release fraction of spilled/splashed Flibe	Below airborne release fraction limit derived to bound total releases of the postulated event to less than the MHA	Salt Spill, Seismic
Volatile product formation from Flibe-air reaction	Negligible amount of additional volatile products formed	Salt Spill, PHSS break
Volatile product formation from Flibe chemical reaction with water, concrete, and/or construction materials (e.g., insulation, steel)	Negligible amount of additional volatile products formed	Salt Spill
Mass loss of pebble carbon matrix due to oxidation	Mass loss does not extend into the fueled zone	Salt Spill, PHSS break

Figure of Merit	Acceptance Criterion	Applicable Events
Mass loss of structural graphite due to oxidation	Bounded by the MHA release	Salt Spill, PHSS break
Peak structural graphite temperature-time	Generally bounded by temperature-time curves derived from the assumed MHA structural graphite temperature-time curve	Salt Spill, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break
Peak pebble carbon matrix temperature-time	Generally bounded by temperature-time curves derived from the assumed MHA pebble carbon matrix temperature-time curve	Salt Spill, Reactivity Insertion, Increase in Heat Removal, Loss of Forced Circulation, PHSS break
Peak TRISO temperature-time ex-vessel	Generally bounded by temperature-time curves derived from the assumed MHA fuel temperature-time curve	PHSS break
Amount of materials at risk released	Less than limit derived to bound total releases of the postulated event to less than the MHA	PHSS break

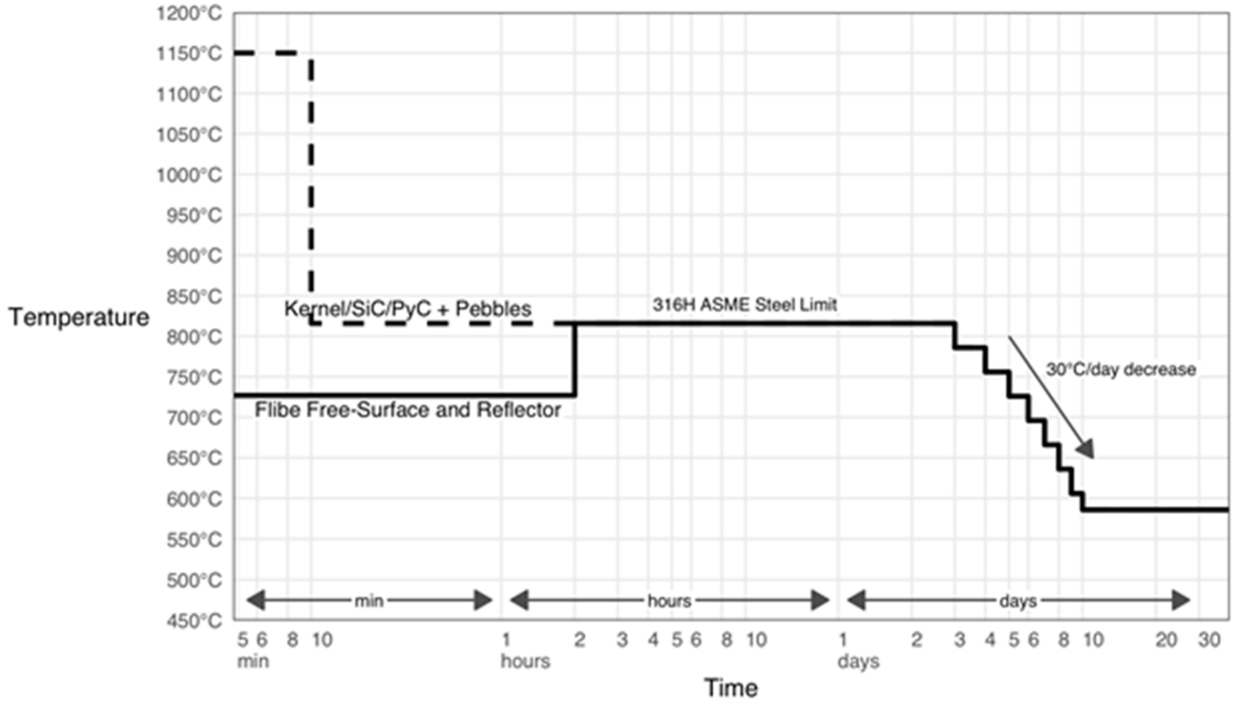
Table 13.2-1: Site-Specific χ/Q Values

Distance (m)	$\frac{\chi}{Q}$ (s/m³)				
	0-2 hrs	**2-8 hrs**	**8 hrs – 1 day**	**1 – 4 days**	**4 – 30 days**
250	1.51×10^{-4}	N/A	N/A	N/A	N/A
800	3.61×10^{-5}	3.51×10^{-5}	1.45×10^{-5}	1.54×10^{-5}	1.49×10^{-5}

Table 13.2-2: Maximum Hypothetical Accident Dose Consequences

Location and Duration	Whole Body Dose (rem)		Thyroid Dose (rem)	
	10 CFR 100 Limit	MHA Result	10 CFR 100 Limit	MHA Result
Exclusion Area Boundary (First 2 hrs at 250m)	25	0.227	300	0.235
Low Population Zone (30 days at 800m)	25	0.059	300	0.081

Figure 13.2-1 Hypothetical Temperatures Prescribed for MHA



Chapter 14

Technical Specifications

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

None

CHAPTER 14 TECHNICAL SPECIFICATIONS

14.1 INTRODUCTION

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 test reactor facility are provided in Table 14.1-1. These variables and conditions are the result of the preliminary safety analyses described elsewhere in this report. The variables and conditions apply to both Unit 1 and Unit 2.

The technical specifications and parameter limits will be submitted with the application for an Operating License, consistent with 10 CFR 50.34(b)(6)(vi) and address the requirements in 10 CFR 50.36. Note that in a Kairos Power Fluoride Salt-Cooled High Temperature Reactor (KP-FHR), the reactor coolant boundary does not serve a fission product barrier function. 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.”

The format and content of the technical specifications are consistent with the guidance provided in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 15.1, “The Development of Technical Specifications for Research Reactors” (ANSI/ANS, 2007) and include:

- Safety Limits and Limiting Safety System Settings
- Limiting Conditions for Operation
- Surveillance Requirements
- Design Features
- Administrative Controls

14.2 OPERATING MODES

The operational modes for the reactor are summarized in Table 14.2-1. These operational modes apply to both Unit 1 and Unit 2. Each operational mode is defined in terms of combinations of core reactivity, reactor power, and nominal outlet reactor coolant temperature. These modes are described individually in the following subsections.

14.2.1 MODE 1: Full Power

In this mode, the reactor is critical and the thermal output ranges between 20% and 100% of rated power. In this mode reactor temperature (outlet) is between 550°C - 650°C. This power level may be desired for testing purposes such as irradiation data collection, transient maneuvers, and other system testing at elevated power levels. Higher powers may be desired for testing purposes such as the collection of irradiation data and system performance evaluations. This mode is achieved during a controlled power ascension from MODE 2 and is declared when the reactor power reaches 20% or higher. MODE 1 may be exited as part of a controlled power reduction to MODE 2 or automatically as a result of a reactor trip to MODE 3.

14.2.2 MODE 2: Low Power

In this mode, the reactor ranges from the state point of hot zero power (also known as “zero power critical”) up to less than 20% of rated power. This is the point where the reactor is critical but with very low neutron flux and very low nuclear heat generation (maintained at temperature with the reactor auxiliary heating system (see Section 9.1.5.1) as needed). In this mode the reactor temperature (outlet) is between 550°C - 650°C. This mode serves as a transition from shutdown conditions to power operation and vice versa. This mode is typically used to perform nuclear and non-nuclear testing and calibration activities at low power, as well as some maintenance activities that may be performed online but at reduced power levels. Core physics testing and control rod calibrations are also performed at hot zero power, after an approach to criticality process. MODE 2 may be exited as part of a normal controlled power ascension to MODE 1, a controlled reactivity shutdown to MODE 3, or automatically as a result of a reactor trip to MODE 3.

14.2.3 MODE 3: Hot Shutdown

In this mode, the reactor is maintained subcritical. Fuel is in the core and molten reactor coolant is present in the reactor vessel. In this mode reactor temperature (outlet) is between 550°C - 600°C. This mode may be entered as part of a controlled reactivity shutdown from MODE 2, as a result of a reactor trip from any of the three power operating modes, or during a plant startup from the hot defueled MODE 4.

14.2.4 MODE 4: Defueled

In this mode, the reactor vessel is fully defueled with the fuel secured in storage. The vessel contains molten reactor coolant and is maintained at a hot temperature by the primary heaters. Operation in this mode is expected to be infrequent. This mode is a transition mode during an initial plant startup or shutdown to hot drained or cold conditions. Operation in this mode could also be used for maintenance or repair activities requiring defueling.

14.2.5 MODE 5: Drained

In this mode, the reactor vessel is fully defueled and drained with the fuel secured in storage. The vessel is drained of the molten reactor coolant. This mode is primarily for hot drained or cold conditions. Operation in this mode could also be used for maintenance or repair activities requiring draining of the system.

14.3 REFERENCES

1. Kairos Power LLC Topical Report, "Regulatory Analysis for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor," KP-TR-004-NP-A. June 2022.

Table 14.1-1: Proposed Variables and Conditions for Technical Specifications

Section	Section Name	LCO or Condition	Basis
2.0	Safety Limits (SL) and Limiting Safety System Settings (LSSS) Safety Limits 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. Limiting Safety System Settings 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 Safety Limit is exceeded. This Table consists of the proposed subjects of Safety Limits and Limiting Safety System Settings. These are provided below.		
2.1	SL	The fuel temperature shall not exceed an upper bound operating range under any operating conditions.	The maximum fuel temperature Safety Limit is established to ensure fuel integrity based on temperatures assumed in the safety analysis.
2.1	SL	The reactor vessel surface temperature shall not exceed an upper bound temperature under any condition of operation.	The maximum reactor vessel surface temperature Safety Limit is the maximum temperature that can be permitted with confidence that vessel integrity will be maintained.
2.2	LSSS	The core exit reactor coolant temperature(s) shall not exceed an upper bound temperature under any condition of operation.	Limiting the maximum core exit coolant temperature will ensure that the Safety Limits are not exceeded and that the reactor will trip prior to reaching a Safety Limit.
2.2	LSSS	The coolant level shall not fall below a lower bound limit under any condition of operation.	Limiting the coolant low level will ensure that adequate core cooling is available so that the Safety Limits are not exceeded.
2.2	LSSS	The rate of flux trip function shall not exceed an upper bound limit as specified in the safety analysis.	Limiting the rate of power/flux increase will ensure that the reactor will trip prior to challenging the integrity of fuel (or a limitation set in fuel performance methodology).

Section	Section Name	LCO or Condition	Basis
2.2	LSSS	The high-power flux 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 safety limit assumed in the safety analysis.
3.0	Limiting Conditions for Operation (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	Pebble wear is within acceptable limits to support pebble reinsertion.	The objective is to ensure that pebble wear is controlled within limits assumed by or associated with safety analyses, to prevent reinsertion if wear exceeds those limits.
		Reactor power shall not exceed the licensed reactor power level.	The objective is to limit the maximum operating power to ensure that the safety limits will not be exceeded.
3.2	Reactor Control and Safety Systems	Reactivity coefficients are within limits over the allowable range of operation.	The objective is to infer or calculate reactivity coefficients during normal plant operation to limit the severity of a reactivity transient.
		Reactor protection system operability	The objective is to specify the requirement to have an operable reactor protection system to ensure that the safety limits will not be exceeded.
3.3	Coolant Systems	Reactor coolant chemical composition is maintained within allowable limits.	The objective is to ensure that the thermophysical properties and chemical composition of the reactor coolant are maintained within limits assumed by or associated with safety analyses.

Section	Section Name	LCO or Condition	Basis
		The radionuclide inventory of the reactor coolant in steady state (e.g., from transmutation of actinides) is maintained within an upper bound limit.	The objective is to limit key radionuclide inventories in the reactor coolant during steady state to ensure that any postulated event does not exceed limits.
		Primary heat transport system pressure and flow rate are maintained within an upper bound limit.	The objective is to limit the quantity and pressure of spilled Flibe to ensure a postulated event does not exceed limits.
		Inert gas system pressure is maintained within an upper bound limit.	The objective is to limit the quantity and pressure of spilled Flibe or cover gas to ensure a postulated event does not exceed limits.
		Argon purity in the cover gas is maintained within an upper bound limit.	The objective is to limit radionuclides in the Flibe below solubility limits where solute-solute interactions can be neglected.
		The quantities of materials at risk in the gas space of the primary heat transport system, the intermediate heat transport system, and the power generation system are maintained within upper bound limits.	The objective is to limit the quantities of materials at risk in the primary heat transport system cover gas, the intermediate heat transport system, and the power generation system to ensure a postulated event does not exceed limits.
		The quantity of air in the reactor system during steady state is maintained within an upper bound limit.	The objective is to limit the air ingress to the reactor system to prevent void accumulation and corrosion.
		Intermediate heat transport system pressure relief device operability	The objective is to specify the requirements to have operable intermediate heat transport system pressure relief devices to prevent the intermediate heat transport system pressure from exceeding the intermediate heat exchanger design limits during postulated events.

Section	Section Name	LCO or Condition	Basis
		The quantity of Flibe in the intermediate coolant shall be maintained below an upper bound limit.	The objective is to limit Flibe ingress into the intermediate heat transport system to ensure a postulated event does not exceed limits.
		The quantity of water in the intermediate coolant shall be maintained below an upper bound limit.	The objective is to limit water ingress into the intermediate heat transport system to limit corrosion and to ensure a postulated event does not exceed limits.
3.4	Engineered Safety Features	Decay heat removal system operability	The objective is to specify the requirement to have an operable decay heat removal system to ensure that the safety limits will not be exceeded.
3.5	Ventilation Systems	N/A	N/A
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.	Radiation in plant effluents is measured against applicable limits.
3.8	Experiments	N/A	N/A
3.9	Facility Specific LCOs	N/A	N/A
4.0	Surveillance Requirements (SRs) Surveillance requirements relating to test, calibration, or inspection, to assure the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that limiting conditions for operation will be met, will be provided in the technical specifications.		
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 application for an Operating License.		

Section	**Section Name**	**LCO or Condition**	**Basis**
6.0	Administrative Controls Administrative controls are the programmatic provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting, necessary to assure operation of the facility in a safe manner. Administrative controls will be provided in the application for an Operating License.		

Table 14.2-1: Operating MODES for Technical Specifications

Operating Modes		Criticality (Keff)	Reactor Power[1]	Nominal Outlet Reactor Coolant Temp[2] (°C)
MODE 1	NORMAL POWER	≥0.99	20% - 100% (7-35 MWth)	550 - 650
MODE 2	LOW POWER/STARTUP	≥0.99	<20% (<7 MWth)	550 - 650
MODE 3	HOT SHUTDOWN (FUELED)	<0.99	0	550 - 600
MODE 4	DEFUELED	N/A	0	Molten Flibe
MODE 5	DRAINED (No Flibe in the System)	N/A	0	N/A

Notes:

1. Value reported for reactor power does not include power from decay heat for MODE 3.
2. The nominal reactor coolant outlet temperature has a control band of +/- 7°C around the reported values.

Kairos Power

Chapter 15

Financial Qualifications

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 15 FINANCIAL QUALIFICATIONS

This chapter provides the financial information which establishes that Kairos Power is financially qualified to own, construct, operate, and decommission the Kairos Power test reactor facility. The Kairos Power 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 in NUREG-1537, Part 1 and the Final Interim Staff Guidance Augmenting NUREG-1537, Part 1.

15.1 FINANCIAL ABILITY TO CONSTRUCT THE KAIROS POWER FACILITY

The Nuclear Regulatory Commission (NRC) has set forth requirements for applicants for a Construction Permit pursuant to 10 CFR 50.33(f) to submit sufficient information to demonstrate that the applicant possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs, including the source(s) of funds to cover these costs.

Appendix C to 10 CFR 50 provides financial guidelines and distinguishes between applicants that are established organizations and those that are newly-formed entities organized primarily for the purpose of engaging in the activity for which the permit is sought. Appendix C provides a guide for the financial data and related information required to establish financial qualifications for Construction Permits. Kairos Power is considered a newly-formed entity. As stated in Appendix C, the information required by the NRC that will normally be required of applicants which are newly-formed entities, and specific to construction cost estimates, will not differ in scope from that required of established organizations.

Pursuant to 10 CFR 50.33(f)(1), and in accordance with Appendix C guidelines of this regulation, Kairos Power has provided estimates associated with the total construction of the facility and related fuel costs, as well as funding sources, in an enclosure to the letter submitting the Construction Permit application.

15.2 FINANCIAL ABILITY TO OPERATE THE KAIROS POWER FACILITY

Kairos Power expects to apply for a Class 104 license per 10 CFR 50.21(c) (for testing, research, and development activities), and receipt, possession and use of source material under 10 CFR 40, byproduct material under 10 CFR 30, and special nuclear material under 10 CFR 70. Kairos Power financial projections assume an 411-year operating period for the non-power reactor facility.

Kairos Power has reasonable assurance of obtaining the necessary funds to cover estimated facility operation costs for the period of the license. Operating costs for the facility will be covered by sustained private investment from Kairos Power investors, with potential supplements from other funding sources. Estimates of the total annual operating costs for each of the first five years of operation of the facility will be provided with the application for an Operating License consistent with 10 CFR 50.33(f)(2).

15.3 FINANCIAL ABILITY TO DECOMMISSION THE KAIROS POWER FACILITY

Kairos Power has reasonable assurance that funds will be available to decommission the facility in accordance with 10 CFR 50.33(k). This information is to be submitted to the NRC for decommissioning in accordance with 10 CFR 50.75(d)(1) as part of the application for an Operating License.

Kairos Power will provide a site-specific decommissioning plan with estimated costs and financial assurances to support those costs in the application for an Operating License.

15.4 FOREIGN OWNERSHIP, CONTROL, OR DOMINATION

Kairos Power LLC is the applicant for the construction permit and subsequent operating license for the test reactor. Kairos Power LLC is a limited liability company formed in the State of Delaware with a principal place of business in Alameda, California and is not acting as an agent or representative of another person in filing the application. Kairos Power is a privately held company with a limited number of investors that solely own the company and its assets. In addition, current employees of Kairos Power hold options to purchase shares in the future, but at the time of this application, such shares have not been established. Current investors are United States citizens or entities owned or controlled by United States citizens. Employees with the options to hold future shares totaling one percent or more of Kairos Power's stock or options are United States citizens or entities owned or controlled by United States citizens. Therefore, ownership and control are not dominated by foreign entities or individuals.

15.5 NUCLEAR INSURANCE AND INDEMNITY

Kairos Power intends to obtain insurance and financial protection consistent with the requirements of the Price-Anderson Act, pursuant to Section 170 of The Atomic Energy Act of 1954, as amended and the requirements in 10 CFR 140.

After receipt of the construction permit and 10 CFR 70 license to possess fuel, Kairos Power will obtain financial protection of $1 million in insurance consistent with 10 CFR 140.13. Prior to operation, Kairos Power will obtain the full financial protection required by 10 CFR 140 using the formula provided in 10 CFR 140.12(b). The amounts of financial insurance required by 10 CFR 140.12(b) and documentation required by 10 CFR 140.15 will be provided with the application for an Operating License.

Chapter 16

Other License Considerations

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

List of Figures

CHAPTER 16 OTHER LICENSE CONSIDERATIONS

16.1 PRIOR USE OF FACILITY COMPONENTS

The facility is constructed of new and appropriately qualified structures, systems, and components to conduct operations. Discussions regarding used systems and components are not applicable to the facility.

16.2 MEDICAL USE OF NON-POWER REACTORS

The facility does not contain equipment or facilities associated with direct medical administration of radioisotopes or other radiation-based therapies and has no plans at this time to support medical uses. Therefore, discussions involving medical use of the facility are not applicable.

Chapter 17

Decommissioning and Possession-Only License Amendments

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

None

List of Figures

None

CHAPTER 17 DECOMMISSIONING AND POSSESSION ONLY LICENSE AMENDMENTS

17.1 DECOMMISSIONING

A decommissioning report for the facility will be provided with the application for the Operating License consistent with 10 CFR 50.33(k) and 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.

17.2 POSSESSION-ONLY LICENSE AMENDMENTS

This section relates to a possession-only license and is not applicable to the construction and operation phases of the facility.

Kairos Power

Chapter 18

Highly Enriched To Low Enriched Uranium Conversion

Hermes 2 Non-Power Reactor

Preliminary Safety Analysis Report

Revision 0

July 2023

TABLE OF CONTENTS

List of Tables

None

List of Figures

None

CHAPTER 18 HIGHLY ENRICHED TO LOW ENRICHED URANIUM CONVERSION

18.1 HIGHLY ENRICHED TO LOW ENRICHED URANIUM CONVERSION

The reactor fuel is a high-temperature graphite-matrix coated TRISO particle using high assay, low enriched uranium. The reactor facility does not perform conversion activities nor does it utilize highly enriched uranium that is enriched to 20% or more in 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.