Enclosure 1
Presentation Slides for the March 23, 2023
ACRS Kairos Power Subcommittee Meeting
(Non-Proprietary)

Introduction and Hermes PSAR Chapter 1

DREW PEEBLES – SENIOR LICENSING MANAGER

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 23, 2023

Kairos Power's mission is to enable the world's transition to clean energy, with the ultimate goal of dramatically improving people's quality of life while protecting the environment.

Overview of Kairos Power

- Nuclear energy engineering, design and manufacturing company ***singularly focused*** on the commercialization of the fluoride salt-cooled high-temperature reactor (FHR)
  - Founded in 2016
  - Current Staffing:
    - Over 300 Employees (***and growing***)
    - ~90% Engineering Staff
- Private funding commitment to engineering design and licensing program and physical demonstration through nuclear and non-nuclear technology development program
- Schedule driven by the goal for U.S. commercial demonstration by 2031 (or earlier) to enable rapid deployment in 2030s
- Cost targets set to be competitive with natural gas in the U.S. electricity market

Kairos Power Headquarters

Kairos Power Team

Hermes PSAR Overview

- 10 CFR 50 Licensing Pathway
  - Construction Permit Application – Submitted Fall 2021
    - Environmental Report
    - Preliminary Safety Analysis Report (PSAR)
  - Next Licensing Step: Operating License Application
    - Final Safety Analysis Report (FSAR)
- Hermes PSAR Application Format and Content
  - Developed using guidance in NUREG 1537
  - Presents preliminary design and preliminary safety analysis consistent with 10 CFR 50.34(a)
  - PSAR does not request commission approval of the safety of any design feature or specification
    - *10 CFR 50.35(b) A construction permit will constitute an authorization to the applicant to proceed with construction but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests such approval and such approval is incorporated in the permit.*

* Not Applicable to Hermes – Chapter has no content

** Minimal Content at PSAR

Hermes PSAR Format

- Chapter 1 – The Facility
- Chapter 2 – Site Characteristics
- 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*

Kairos Power Reports Referenced in PSAR

- Topical Reports
  - KP-TR-003 – Principal Design Criteria
  - KP-TR-004 – Regulatory Analysis
  - KP-TR-005 – Reactor Coolant
  - KP-TR-007 – Quality Assurance Plan
  - KP-TR-010 – Fuel Performance Methodology
  - KP-TR-011 – Fuel Qualification Methodology
  - KP-TR-012 – Mechanistic Source Term Methodology
  - KP-TR-013 – Metallics Qualification Methodology
  - KP-TR-014 – Graphite Qualification Methodology
- Technical Reports
  - KP-TR-017 – Core Design Methodology
  - KP-TR-018 – Postulated Event Methodology

Hermes PSAR Chapter 1 – The Facility

- The purpose of Hermes is to test and demonstrate the key technologies, design features, and safety functions for KP-FHR technology
  - 35 MWth non-power reactor facility, 4 year licensed lifetime
  - Located in Oak Ridge, Tennessee at the East Tennessee Technology Park (Former site of Oak Ridge Gaseous Diffusion Plant)
- Principal Design Criteria based on NRC-approved topical report, KP-TR-003-NP-A "Principal Design Criteria"
- Low consequences due to inherent safety features
  - TRISO fuel
  - Flibe coolant
- Engineered safety features are provided to contain fission products and passively remove decay heat
- Instrumentation and control system provides monitors and controls plant operations. Electrical System provides the normal and backup power to the facility
- Auxiliary systems include a chemistry control system, inert gas system, tritium management system, fire protection system, heating and cooling systems, etc.

Hermes PSAR Chapter 1 – The Facility

- Nuclear Safety Classifications: Safety-Related or Non-Safety Related
- Potential events are evaluated using a deterministic safety analysis with a Maximum Hypothetical Accident
- Radioactive waste management controls wastes produced by plant operations and radiation protection program protects health and safety of workers
- Experimental capabilities include testing of fuel irradiation, materials corrosion and irradiation, and transient and power maneuvering
    - Capability to perform these activities is included in normal system design described in PSAR
    - No additional facilities or capabilities required
- Research and development programs to resolve safety questions will be resolved before the completion of construction
- Hermes is a single unit reactor that does not share any systems or equipment to perform safety functions

Hermes PSAR 3.1 Introduction and 3.6 Systems and Components

DREW PEEBLES – SENIOR LICENSING MANAGER

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3.1 Applicable Regulations and Guidance

- Kairos Power is pursuing a construction permit for the Hermes reactor under 10 CFR 50
- The NRC regulations in Title 10 to the CFR were evaluated for applicability and documented in "Regulatory Analysis for the Kairos Power Fluoride Salt-Cooled, High Temperature Reactor" topical report (KP-TR-004-NP-A)
- PSAR Table 3.1-1 identifies the design-related regulations that are applicable to the Hermes Test Reactor
  - No specific exemptions from regulations were identified
  - Regulations related to combustible gas control were concluded to be not technically relevant
- Kairos Power evaluated NRC regulatory guides for applicability to the Hermes Test Reactor
  - NRC Division 1 regulatory guides are not applicable to research and test reactors
  - Divisions 2, 4, and 8 apply and were considered for the Hermes Test Reactor, as shown in Section 3.1

3.1 Principal Design Criteria

- Kairos Power has developed a set of Principal Design Criteria (PDC) for KP-FHR technology
- The design criteria were approved in a Topical Report titled "Principal Design Criteria for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor" (TR-003-NP-A)
- These PDCs have been applied to the design of the Hermes Test Reactor, with the following exceptions:
    - PDC 5, Sharing of structures, systems, and components (SSCs) - Satisfied because there is only one reactor and no SSCs are shared with another reactor
    - PDC 73, Reactor coolant system interfaces - Not Applicable to the Hermes Test Reactor because there is no secondary coolant fluid
- The terms “safety-significant,” “anticipated operational occurrences,” and “accidents” used in the PDCs are not applicable to the Hermes reactor and are not used in the PSAR
    - These terms are relevant to power reactors which use frequency to bin postulated events
    - The Hermes safety analysis utilizes a deterministic Maximum Hypothetical Accident (MHA)

3.6 Fundamental Safety Functions

- Prevent uncontrolled release of radionuclides
  - Functional containment (TRISO fuel and Flibe coolant) retains fission products and limits release during normal and postulated events
  - Safety-related fluid systems that may contain circulating radiological activity are designed to ASME Section III
  - Non-safety-related fluid systems that may contain circulating radiological activity are designed to ASME Section VIII, B31.1/B31.3, or applicable API standards
- Remove decay heat in the event of a postulated event
  - Natural circulation and the passive decay heat removal system reject residual heat from the reactor core to the atmosphere
- Control reactivity in the reactor core
  - Reactivity control and shutdown system provides reactivity control during normal and postulated events

3.6 SSC Safety Classification

- SSCs are classified as safety-related or non-safety related
- The 10 CFR 50.2 definition of safety-related for light water reactors is modified for the Hermes Test Reactor as follows:
  - 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 ensure:
    - The integrity of the portions of the reactor coolant ~~pressure~~ 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 that 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
- This departure from 10 CFR 50.2 is necessary because the 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
- The classification of SSCs is shown in PSAR Table 3.6-1

3.6 Seismic and Quality Classifications

- Seismic Classification
  - Safety-related SSCs are classified as SDC-3 in accordance with ASCE 43-19
  - Safety-related SSCs are located in the safety-related portion of the reactor building
  - Non-safety-related SSCs are designed to local building codes (ASCE/SEI 7-10)
- Quality Classification
  - Safety-related SSCs are classified as quality-related
  - Non-safety-related SSCs are classified as not quality-related
  - Quality-related SSCs conform to the requirements of the quality assurance program for the Hermes Test Reactor, which is based on ANSI/ANS 15.8
- The seismic and quality classification of SSCs is shown in PSAR Table 3.6-1

Hermes PSAR Chapters 2.1-2.4, 3.2, and 3.3

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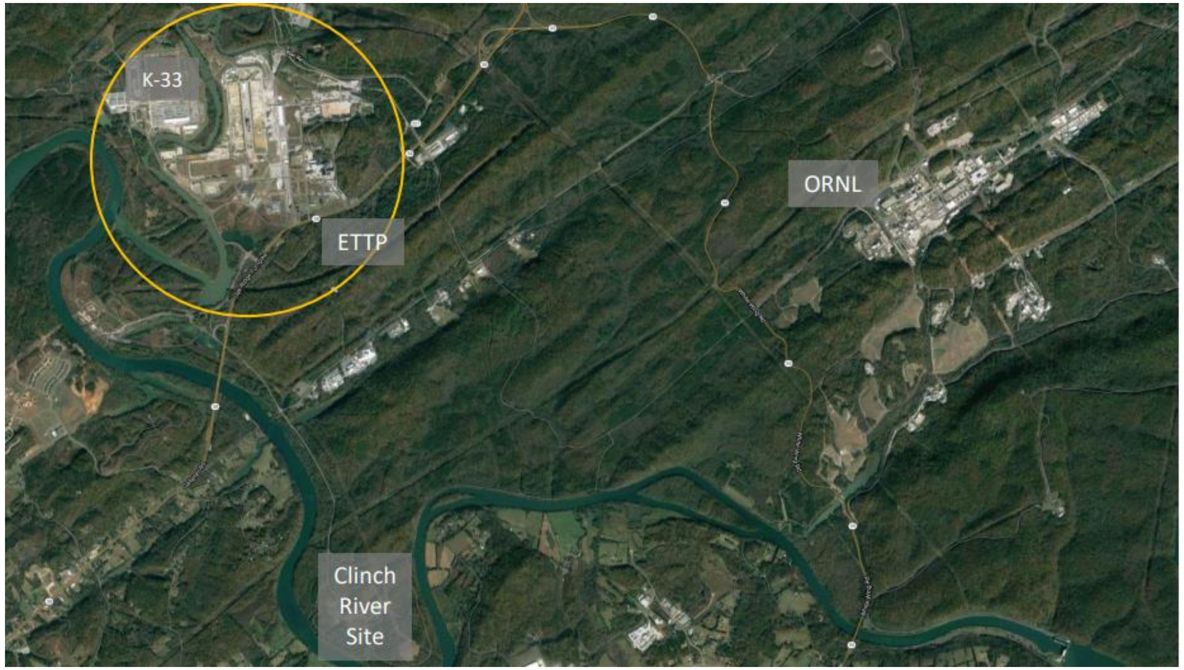
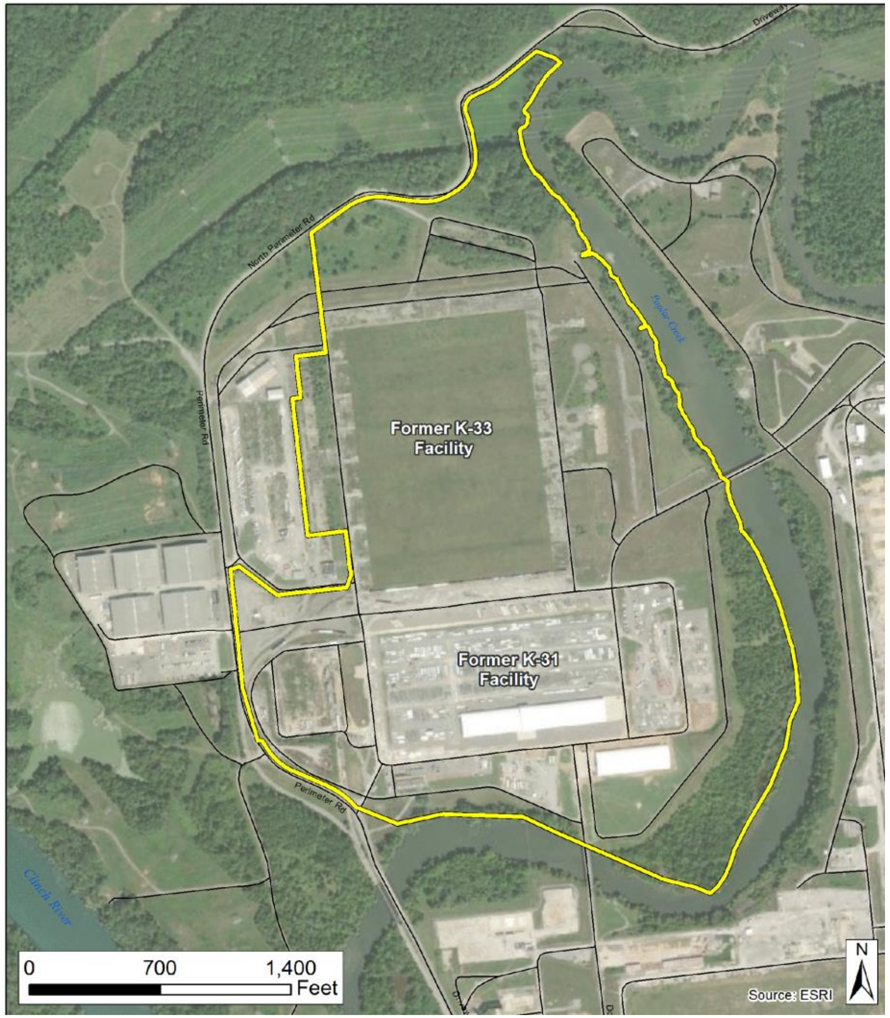
2.1 Geography and Demography: Hermes Site Location

- The site is located in Oak Ridge, Tennessee in Roane County within the East Tennessee Technology Park (ETTP)
- The Hermes test reactor will be located on former Department of Energy gaseous diffusion plant (K-33) building site
- The site boundary encompasses approximately 185 acres
  - About 30 acres would be permanently disturbed for operations of the facility

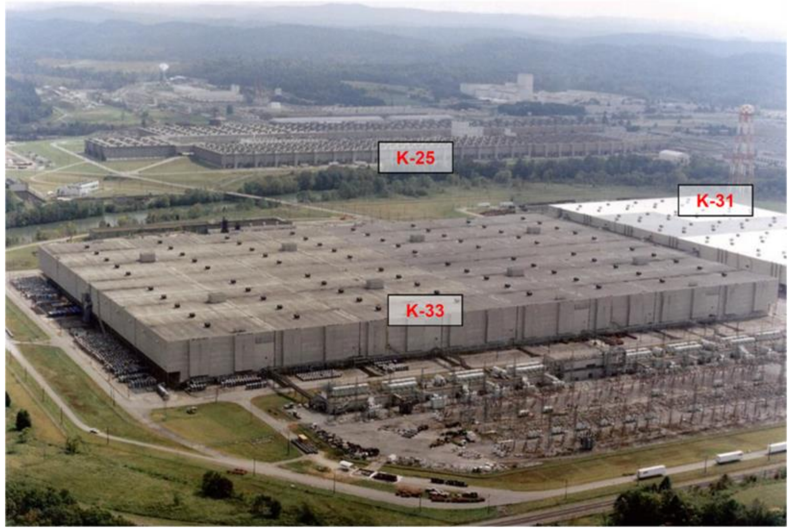




2.1 Geography and Demography: Hermes Site Location

- The original K-33 Building was constructed in 1954
- The uranium enrichment facility ceased operations in 1985
- DOE began reindustrialization of the ETTP in 1996
- The site was released for industrial use in 2011

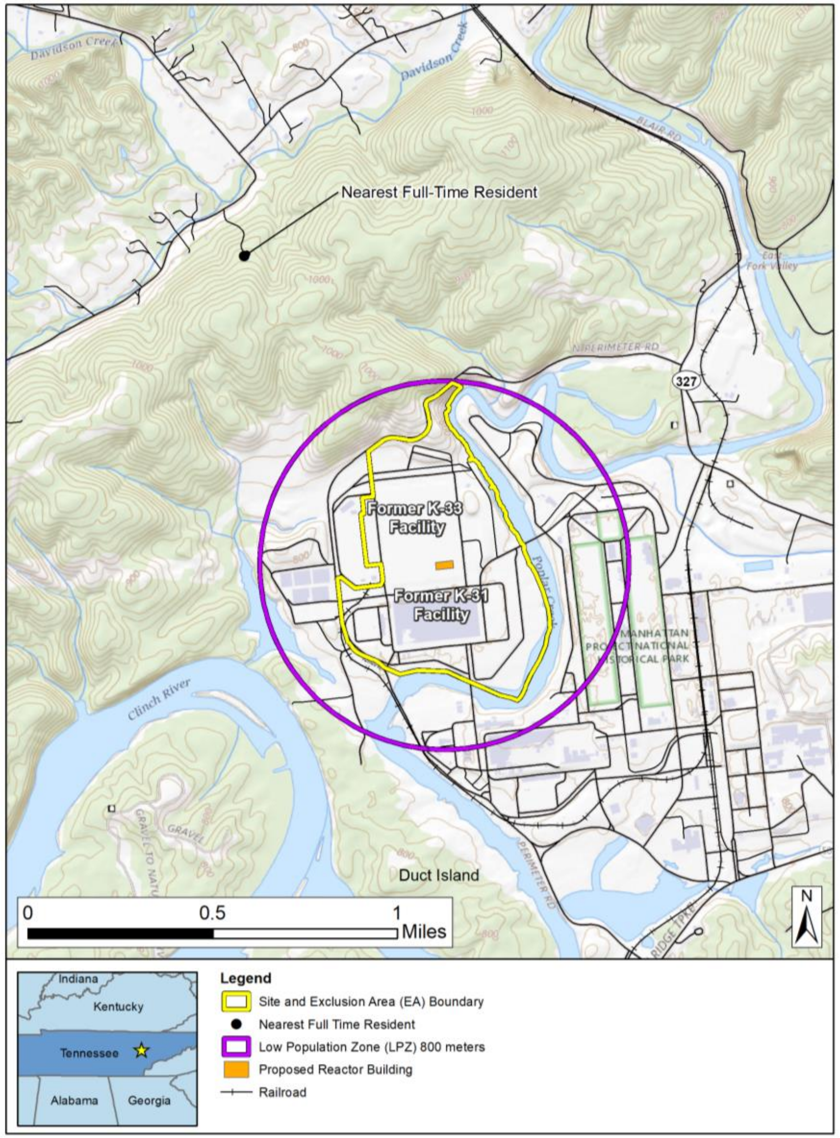
View from West to East



View from North to South

2.1 Geography and Demography: Boundary and Zone Area Maps

- The site boundary is defined by the area owned, leased, or controlled (10 CFR 20.1003)
- The exclusion area boundary is defined as the area within the site boundary where the reactor site management has direct authority over all activities (10 CFR 100.3 and ANSI/ANS-15.16-2015)
- The low population zone is conservatively set at 800 meters from the reactor
  - The nearest resident is 0.7 mi NW from the site boundary
  - The PSAR includes population data 5 miles from the reactor
- The emergency planning zone is coincident to the site boundary (10 CFR 50, Appendix E.I.3)



2.2 Nearby Industrial, Transportation, and Military Installations

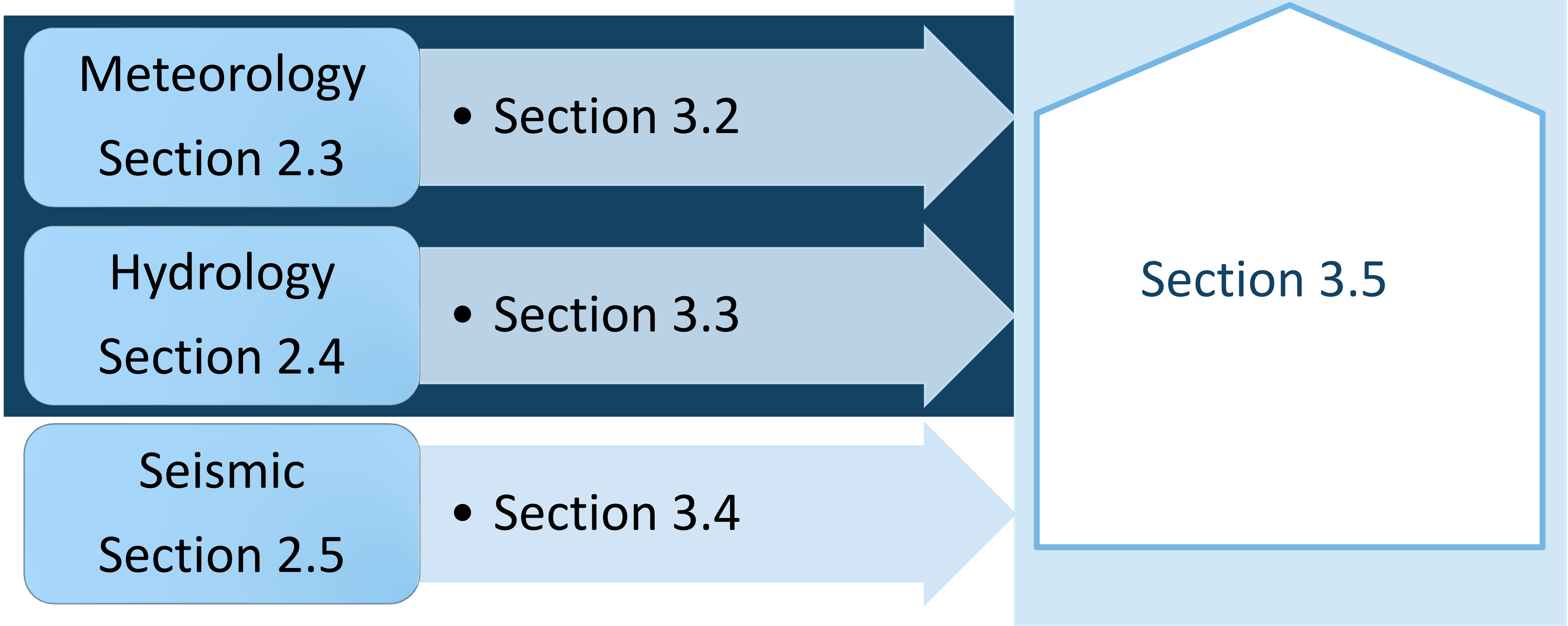
- An investigation of industrial, transportation, and military facilities within 5 miles (8 km) of the site was performed to identify potential external hazards (explosions, flammable vapor clouds [delayed ignition], toxic chemicals, and fires)
- The effects from potential external hazards within 5 miles of the site were determined to not warrant further analysis with the exception of:
  - The distance from the Hermes site to TN-58 was less than the safe distance calculated for shipments of chlorine or anhydrous ammonia. Therefore, the main control room will be designed with detectors for these chemicals.
- There are no existing commercial airports located within 10 miles of the site, however a general aviation airport is proposed to be located less than 1 mile SE of the site
  - The annual probability of an aircraft crashing into the facility was evaluated using the methodology outlined in DOE Standard DOE-STD-3014-2006
    - The total crash frequency for small, non-military aircraft from general aviation or helicopter operations is above the screening acceptance frequency threshold
  - The safety-related portion of the Reactor Building structure will be designed to withstand the impact of a small non-military general aviation aircraft

Chapter 2 and 3 Relationships

Step 1: Define design basis parameter input envelope

Step 2: Define methods to translate inputs into design loads

Step 3: Define protections for safety-related SSCs using design loads



2.3 Meteorology

- The Hermes site is located on a prior U.S. Department of Energy (DOE) nuclear facility site within the DOE-managed Oak Ridge Reservation (ORR)
  - The ORR includes an extensive network of meteorological towers
  - Historical meteorological studies from 1953 and 2011 indicate that basic flow patterns have been in place during the recorded weather history of the ORR area
- Topography influences the weather and climate of the region around the site due to its location between the Cumberland Mountains to the northwest and the Great Smoky Mountains to the southeast.
- Prevailing winds in the region reflect the channeling of airflow from southwest to northeast caused by the orientation of the valleys and ridges

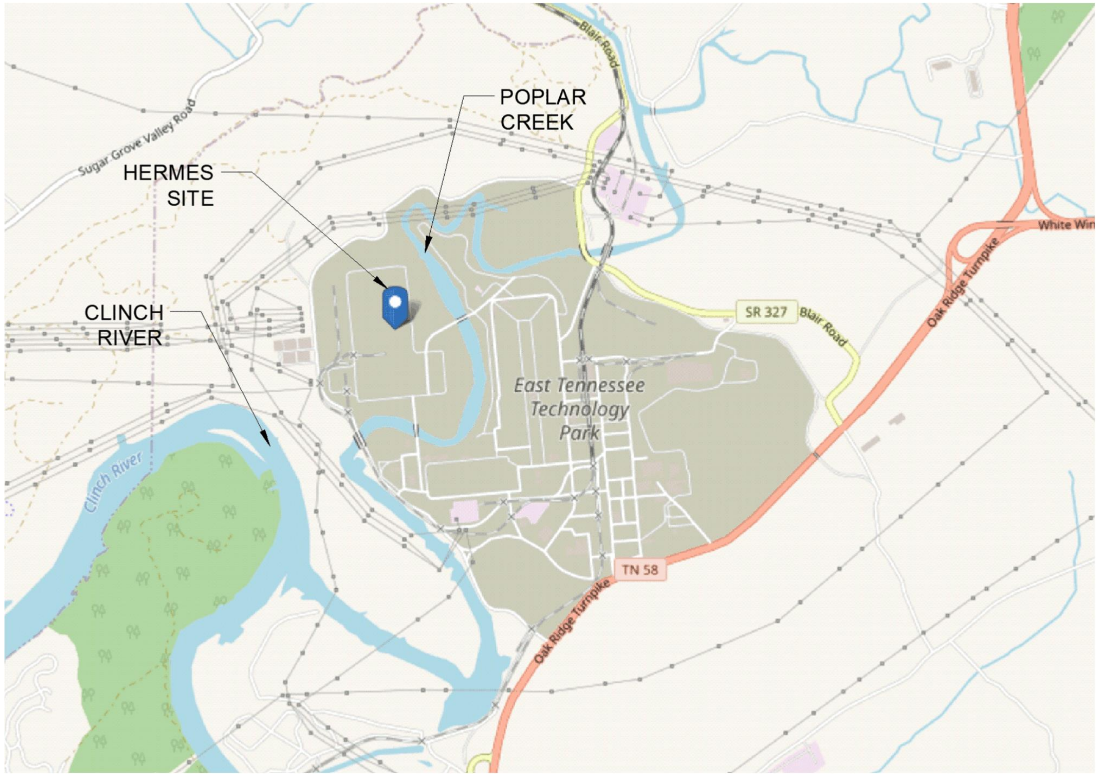
2.3 Meteorology (continued)

- Extreme Winds
  - Estimated extreme winds are based on climatological data from Oak Ridge and Knoxville, Tennessee, and hourly observations from meteorological Tower J (1.1 km southeast of the site) and Tower L (1.6 km southeast of the site)
    - For a 100-year return period, the maximum wind speed is 90 mph
  - Hurricane winds are mainly a concern for coastal locations as shown by the wind speed contours presented in Regulatory Guide 1.221
  - The probability of a tornado occurring at the site is low based on records from the NWS Morristown Tornado Database

- Extreme Precipitation
  - Historical precipitation data for the site were obtained from several surrounding National Weather Service (NWS) and Tennessee Valley Authority (TVA) sites
  - 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
  - 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

2.4 Hydrology: Description

- The site is located near the confluence between Clinch River and Poplar Creek
  - TVA manages water levels year-round for dam safety and flood control
  - Both Clinch River and Poplar Creek are considered as potential flooding sources
- The grade level for the site is 765 feet above mean sea level (feet msl)
  - The normal water surface elevation for Poplar Creek near the site is 744 feet msl (21 feet lower than site grade)



2.4 Hydrology Characterization: Previous Flood Studies

- There are two previous flood studies with estimated flooding elevations in the vicinity of the ETTP Hermes site:
  - FEMA Flood Insurance Study for Roane County, TN
    - Includes 10-, 50-, 100-, and 500-year return periods
    - All flood elevations from this study are below the Hermes site grade of 765 feet above mean sea level (feet msl)
  - Flood Hazard Evaluation for UCOR dated April 2015
    - A large range of return period floods (25 year to 100,000 year) were modeled and estimated
    - Results were assessed and used to identify a preliminary design-basis flood

2.4 Hydrology: Credible Hydrology Events and Design Basis

- The credible hydrological event for the Hermes site is selected as a 25,000-year return period (exceedance probability of 4E-5), consistent with Flooding Design Category 4 (FDC-4)
  - This results in a design basis flooding level for the site at 759.9 feet msl, based on previous studies
    - 5.1 feet below plant grade of 765.0 feet msl
- The Hermes site layout and grading plan takes advantage of the existing site topography so that storm water runoff naturally drains to the east, south, and west with flow directed to Poplar Creek

3.2 Meteorological Damage

- The design of SSCs considers the potential for meteorological damage, including rain, snow, wind, tornado, and tornado and wind-borne missiles for the site
- The safety-related portion of the reactor building structure provides protection to safety-related systems and components from meteorological damage
  - No credit is taken for the non-safety-related portions of the reactor building (exterior shell)
- Design basis meteorological parameters applicable to the design of the safety-related portion of the reactor building structure are established for: normal wind loads, high wind loads (tornados and hurricanes), and precipitation loads
- Normal wind load design basis:
  - Local building codes cite ASCE/SEI 7-10, “Minimum Design Loads for Buildings and Other Structures”. This standard defines risk categories for structures and includes design basis normal wind velocities for each risk category.
  - Risk Category IV (for hazardous substances) is the most stringent and selected as the design basis for the safety-related portions of the Reactor Building
  - Risk Category IV results in a design basis wind velocity of 120 miles per hour (mph)
    - This wind velocity bounds the site characterization meteorological data
    - This is based on a 1700-year mean recurrence interval, which is more conservative than the 100-year return period
  - The applied normal wind loads are determined using ASCE/SEI 7-10 Risk Category IV and exposure category C

3.2 Meteorological Damage (continued)

- High wind load design basis:
  - 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
    - The applied tornado wind loads are determined using the methods in ASCE/SEI 7-10 and the wind speeds from RG 1.76 for Region I
    - The loads from tornado-generated missile impacts are transformed into an effective or equivalent static load consistent with NUREG-0800, Section 3.5.3, Subsection II using the missile spectrum and maximum horizontal speeds provided in Table 2 of RG 1.76 for Region I
  - Guidance from RG 1.221, Revision 0, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” was used to determine applicable design parameters for hurricane loads
    - The applied hurricane wind loads are determined using the methods in ASCE/SEI 7-10 with a maximum wind speed of 130 mph and velocity pressure based on the guidance in RG 1.221 for the site location
    - The loads from hurricane-generated missile impacts are transformed into an effective or equivalent static load consistent with NUREG-0800, Section 3.5.3, Subsection II using the missile spectrum from RG 1.221

3.2 Meteorological Damage (continued)

- Precipitation load design basis:
  - The grading and drainage design for the site will preclude loads from precipitation accumulation on the ground affecting the safety-related portion of the Reactor Building
  - 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 snow load design parameters are based on Chapters 1 and 7 of ASCE/SEI 7-10 for Risk Category IV structures and site location
  - The applied structural snow loads are determined based on the ground snow load of 21.9 psf and using the methods in ASCE/SEI 7-10 for Risk Category IV structures
    - Load considerations include balanced snow loads, unbalanced snow loads, snow drift loads, and rain on snow surcharge loads

3.3 Water Damage

- The design of the safety-related portions of the reactor building considers the loads from both external and internal flooding events
- External flooding postulated events do not pose a hydrologic load because the grade elevation is above the design basis flood elevation determined in PSAR Section 2.4
- Internal flooding postulated events consider the water sources within the safety-related portions of the reactor building
  - As discussed in Section 3.5, safety-related SSCs are protected from internal flooding:
    - Safety-related SSCs vulnerable to flooding are elevated, shielded or otherwise protected from spray. This includes Flibe-bearing components.
    - Design features direct water flow and prevent it from entering enclosures containing safety-related SSCs.
    - The volume of water in the safety-related portions of the reactor building is limited by design. For water systems that cross the base isolation moat, automatic or manual termination of flow will be specified in the operating license application.

Hermes PSAR 2.5, 3.4, and 3.5
Geology, Seismic Design, and Reactor Building Structures

BRIAN SONG – SENIOR MANAGER, CIVIL STRUCTURES

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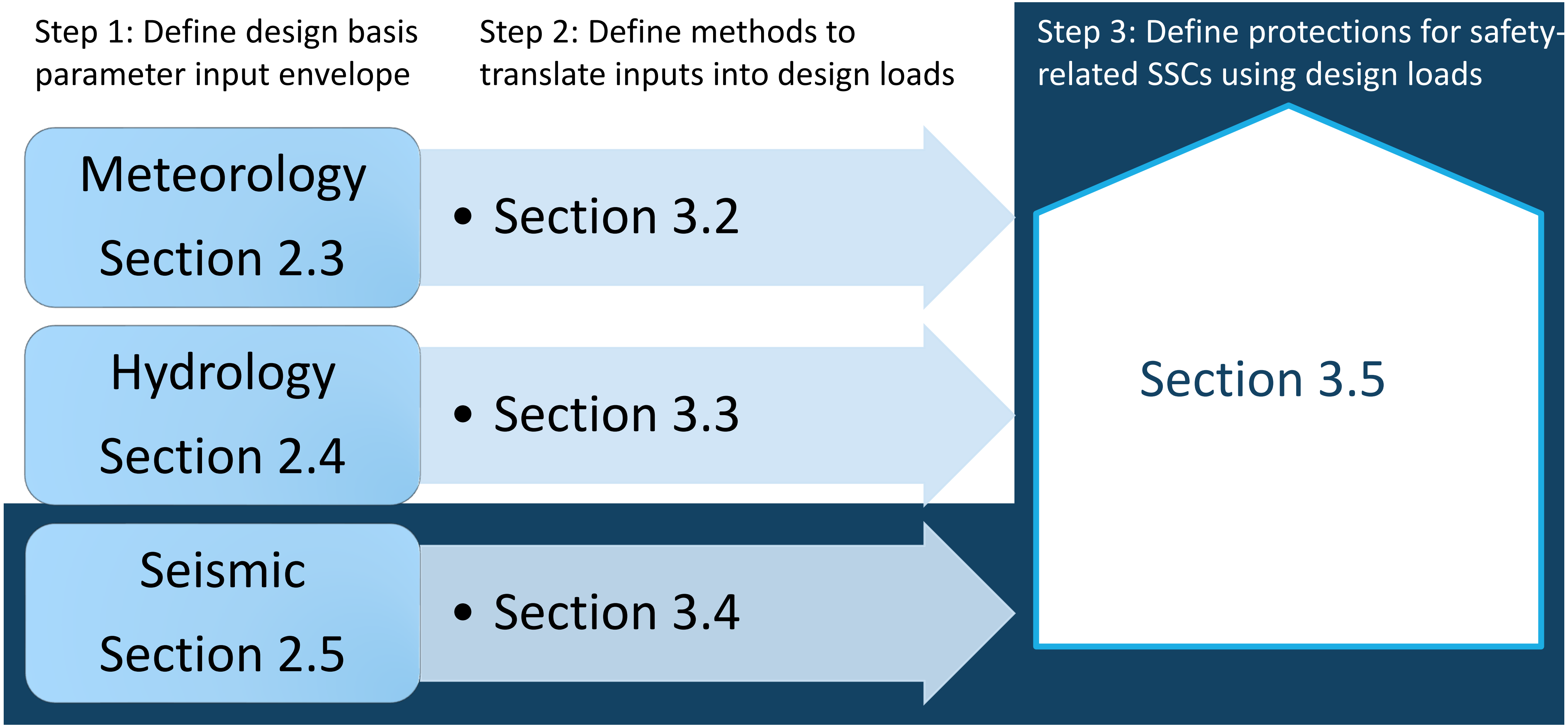
Chapter 2 and 3 Relationships



2.5 Geology, Seismology, and Geotechnical Engineering

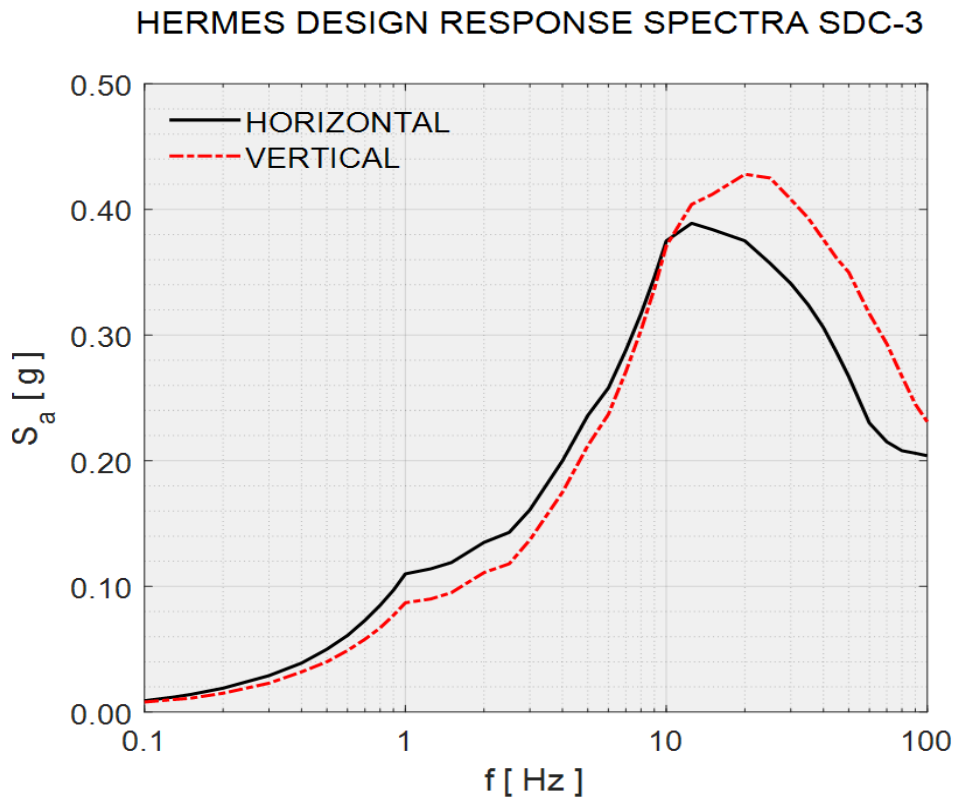
- Section 2.5 characterizes the geologic, geophysical, seismic and geotechnical aspects of the region and site to develop a seismic design basis for the facility
- The Hermes PSAR relies on existing information from the Clinch River Early Site Permit Application (CR-ESPA) for the regional and local geologic description, with supplemental information as needed
  - Covers 200 miles around the site
  - The CRNS site is close (3.5 miles) to the Hermes site and shares the same regional geology
- The Hermes Probabilistic Seismic Hazard Analysis (PSHA) is adapted from the CR-ESPA PSHA supplemented with consideration of current seismic hazard publications for the site and regional area
  - The PSHA methodology is an enhancement over the guidance in NUREG 1537
  - The CRNS PSHA meets ANSI/ANS 2.29 "Probabilistic Seismic Hazard Analysis”

2.5 Geology, Seismology, and Geotechnical Engineering: Site Geology

- CRN site geology information is directly applicable to the Hermes site
- A subsurface stratigraphy was developed for the Hermes site from a geotechnical boring program
- The placement of the facility on the site was informed by the geotechnical information

2.5 Geology, Seismology, and Geotechnical Engineering: Vibratory Ground Motion Analysis

- Uses CRN PSHA to develop the Seismic Design Response Spectra (DRS)
- Analysis relies on information from the CR-ESPA, with supplements
  - Use of the CR-ESPA, PSHA is both appropriate and reasonable given the proximity between both sites
- The Seismic Source Characterization is based on the CEUS (Central and Eastern United States) Seismic Source Characterization report
- The DRS meets ASCE 43-19 and uses Seismic Design Category 3 for safety-related SSCs which is appropriate for a non-power reactor application



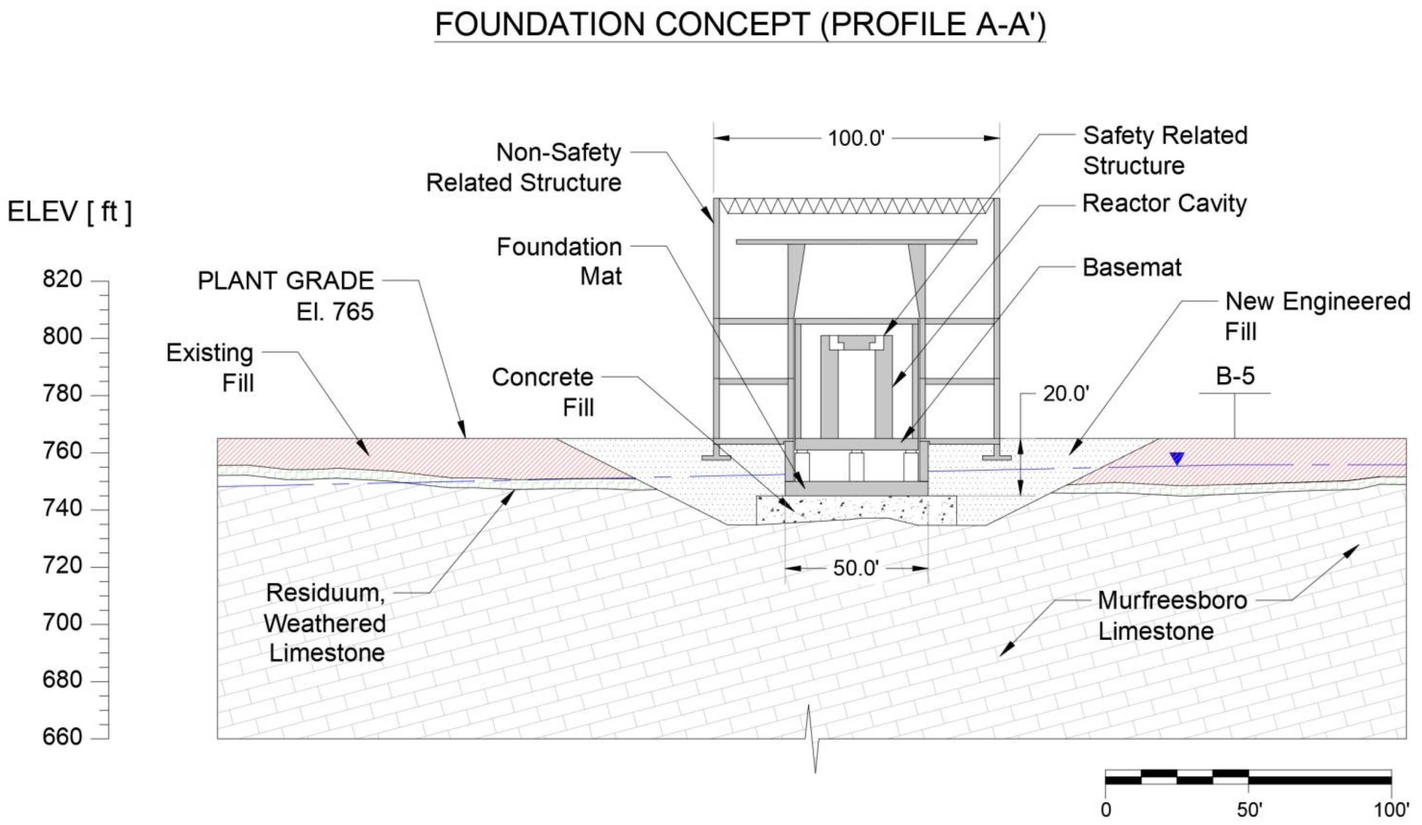
SDC-3 Performance Goal: 1E-4

2.5 Geology, Seismology, and Geotechnical Engineering: Subsurface Deformation

- Relies on information from the CR-ESPA, supplemented by site-specific assessments to assess the potential for 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
  - Only inactive surface faults have been documented within the site area
  - The foundation rock for the Hermes reactor is at depths at which no evidence of karstic dissolution is encountered

2.5 Geology, Seismology, and Geotechnical Engineering: Foundation Interface

- The foundation layout has been established based on knowledge of the site subsurface conditions gathered from both historical documentation, including the CR-ESPA, and the subsurface boring exploration campaign
- The bearing system for the safety-related structure is a foundation mat resting on concrete fill over the Murfreesboro rock



3.4 Seismic Damage

- The graded performance-based approach from ASCE 43-19 is used to design the protections for safety-related SSCs from design basis earthquakes
  - Safety-related SSCs are designed to SDC-3, non-safety related SSCs are designed to local building code, which is consistent with NUREG-1537, IAEA-TECHDOC-403, and IAEA-TECHDOC-1347
  - The return period associated with the design basis ground motion corresponding to SDC-3 is similar to the maximum earthquake specified in building codes with a 2% probability of exceedance in 50 years
    - Consistent with NRC approvals for other non-power reactors
    - Additional margin exists due to the short operating lifetime of Hermes
- Seismic performance criteria are consistent with ANSI/ANS 15.7, Research Reactor Site Evaluation
- The 5% damped horizontal and vertical design response spectra are developed consistent with ANS 2.29, using the DRS defined in Section 2.5
- Structural design of non-safety related SSCs is performed in accordance with the 2012 International Building Code and the Tennessee Building Code

3.4 Seismic Damage: Analysis Models

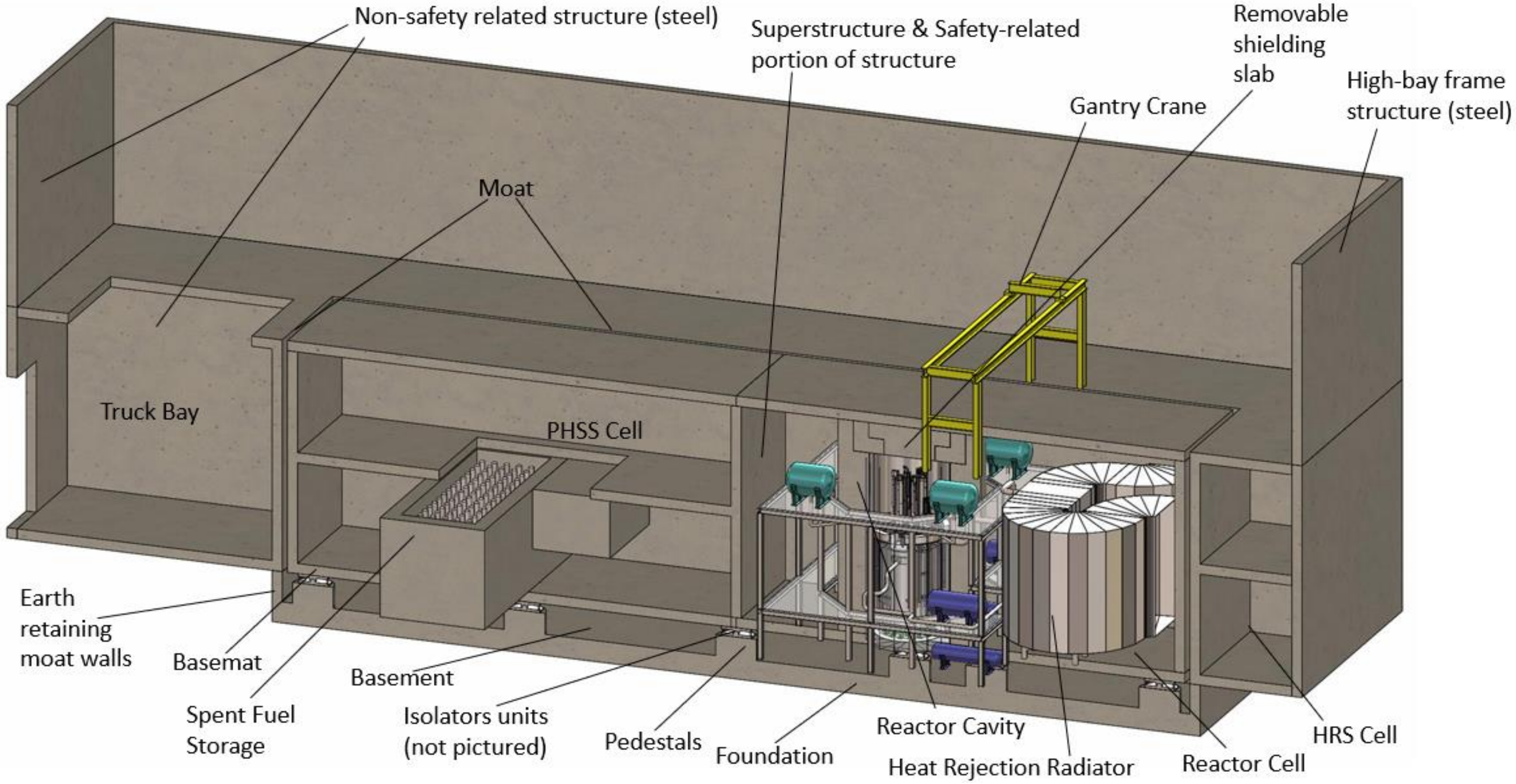
- A 3-D finite element model of safety-related structures will be used for seismic analysis consistent with ASCE 4-16
  - Cracking analysis applies ASCE 4-16 Table 3-2
  - Structural damping applies ASCE 4-16 Table 3-1
  - Structural mass captures self-weight of structural elements as well as portions of design live loads and design uniform snow load
- Models use 3-component seismic input to develop structural forces and in-structure response spectra. Used for SDC-3 structural and equipment qualification.
- Seismic response analysis meets ASCE 4-16, Chapter 4, using deterministic, linear analysis
- Soil-structure analysis will be consistent with ASCE 4-16, Chapter 5

3.4 Seismic Damage: Seismic Instrumentation

- Seismic instrumentation will be installed for monitoring seismic events
- Tri-axial time-history accelerometers will be located in the free field and in the safety-related portion of the reactor building

3.5 Plant Structures: Reactor Building

- ~200’ long, 100’ wide
- Sloped roof
- The safety-related portion of the building uses base isolation using spring/dashpot elements
  - Reactor Cell: vessel, Flibe inventory, and HRR
  - Fuel Cell: PHSS, spent fuel storage
- No other building on the site performs a safety function, including the building that houses the main control room



3.5 Plant Structures: Reactor Building (continued)

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

Note: No part of the reactor building is credited to meet the functional containment safety function

3.5 Plant Structures: Reactor Building Design Criteria

PDC	Description
1	Designed using consensus standards and in accordance with the applicable quality assurance program (ASCE/SEI 7-10).
2	Protects safety-related SSCs from the effects of design basis meteorological, flooding, and seismic events (see Slide 14 for seismic events).
3	Design minimizes the probability and the effect of fires and explosions. (Use of low-combustible materials, separation, fire protection program.)
75	Design protects the geometry of the decay heat removal system from postulated natural phenomena events. (DHRS is located in the safety-related portion of the Reactor Building.)
76	Design permits periodic inspection and surveillance of safety-related structural areas (to be demonstrated in the final safety analysis report).

PDC 2: Seismic Events

- The safety-related portion of the reactor building is a reinforced concrete structure designed to meet ACI 349-2013, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary." Internal steel structures are designed to meet AISC N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities."
- By meeting ASCE 43-19, the safety-related portion of the building provides protection to safety-related SSCs from design basis earthquakes
  - Seismic acceptance is checked for both strength- and displacement-based criteria
  - Limit states are set based on the target performance goals
- Safety-related portion of the Reactor Building uses a spring/dashpot seismic isolation system, which lowers seismic demands on safety-related reactor building and safety-related SSCs in both horizontal and vertical directions
  - The moat is sized to accommodate a displacement consistent with the isolation system meeting the performance goal of 1E-4 per year
  - Design features accommodate potential differential displacements for SSCs that cross the moat

Hermes PSAR 4.2 Reactor Core

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ODED DORON – SENIOR DIRECTOR, REACTOR SYSTEMS DESIGN

ACRS KAIROS POWER SUBCOMMITTEE MEETING

MARCH 23, 2023

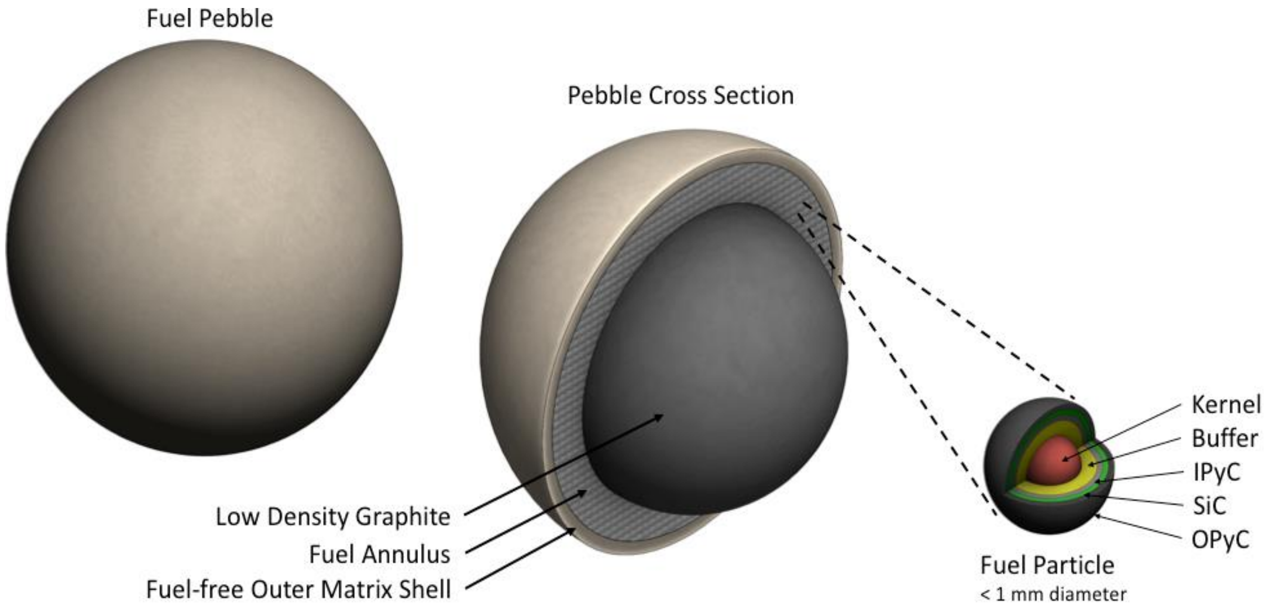
4.2.1 Reactor Fuel

RYAN LATTA – PRINCIPAL ENGINEER, FUELS AND MATERIAL

4.2.1 Reactor Fuel: Fuel Description

- Hermes Test Reactor uses tri-structural isotropic (TRISO) fuel particles in a pebble-based fuel form
- TRISO particle fuel specification is equivalent to the DOE Advanced Gas Reactor (AGR) program
- The kernel and multiple layers of the TRISO fuel particle constitute a primary portion of the functional containment
- Hermes fuel pebble design consists of three regions:
  - Low-density carbon matrix inner core
  - Fuel annulus with TRISO-coated fuel particles embedded in a carbon matrix
  - Fuel-free carbon matrix outer shell
- Moderator pebbles are homogeneous carbon matrix pebbles that do not contain fuel
  - The mixture (ratio) of fuel and moderator pebbles is designed for optimal moderation in Hermes

Annular Fuel Pebble



4.0-cm diameter, annular fuel pebble is about the same size as a golf ball

4.2.1 Reactor Fuel: Fuel Description

Fuel Particle Description

Property	Nominal Value
Kernel diameter (μm)	425
Buffer thickness (μm)	100
PyC thickness (μm)	40
SiC thickness (μm)	35
Kernel density (g/cm^3)	$\geq$ 10.4
Buffer density (g/cm^3)	1.05
PyC density (g/cm^3)	1.90
SiC density (g/cm^3)	$\geq$ 3.19

Fuel Pebble Description

Property	Nominal Value
Pebble radius (cm)	2.0
Overall density (g/cm^3)	1.74
TRISO particles packing fraction	~37%
Pebble uranium loading (g)	6.0
Number of particles per pebble	~16,000

4.2.1 Reactor Fuel: Fuel Qualification

- The Hermes fuel qualification approach is described in topical report KP-TR-011-P "Fuel Qualification Methodology for the Kairos Power Fluoride Salt-Cooled High Temperature Reactor"
- The Hermes TRISO particle fuel specification is equivalent to the DOE AGR fuel specification
  - The EPRI TRISO topical report (EPRI-AR1(NP)-A) demonstrated that the AGR-2 irradiation test resulted in low failure fractions in particles manufactured and inspected to meet the fuel specification
- A PIRT was conducted to evaluate fuel particle and pebble phenomena against a figure of merit
  - The results of the PIRT informs the fuel qualification program
- Pebble laboratory testing in the fuel qualification program demonstrates reasonable assurance the annular pebble will meet functional requirements
  - Mechanical tests – structural integrity
  - Tribology in molten salt and inert gas environments – wear
  - Molten salt infiltration tests – buoyancy
  - Material compatibility tests in salt and air environments – material interaction

4.2.1 Reactor Fuel: Fuel Qualification Envelope

- The Hermes fuel operating envelope is bounded by the fuel qualification envelope established in the fuel qualification methodology topical report
  - The fuel qualification envelope is based on the DOE AGR-2 irradiation and safety tests

Parameter	TRISO Particle Qualification Envelope
Peak SiC Layer Temperature – Normal Operation (°C)	1360
Peak SiC Layer 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

4.2.1 Reactor Fuel: Fuel Surveillance

- The inert cover gas and Flibe coolant activity levels are monitored to detect an increase in fuel particle failure
- Fuel pebbles are examined in the pebble handling and storage system (PHSS) after exiting the core
  - Pebbles are examined for gross damage – wear, cracking, missing surfaces
  - Burnup is measured to confirm it is less than the qualification envelope, allowing pebble recirculation
- Pebbles near the design burnup limit and those exhibiting indications of damage are removed from service and placed in storage

4.2.1 Reactor Fuel: Fuel Design Bases

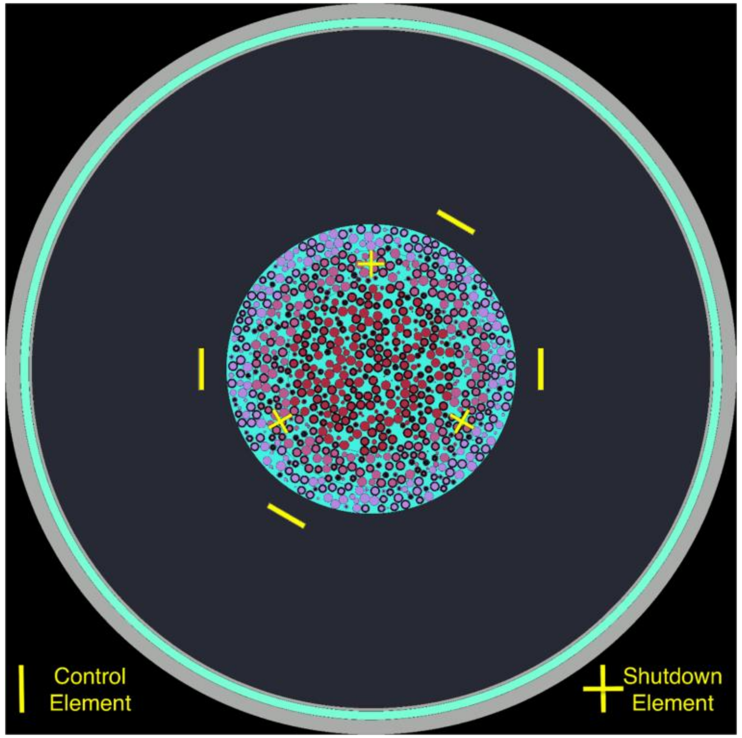
- The fuel is designed with appropriate margin to ensure that specified acceptable system radionuclide release design limits (SARRDLs) are not exceeded (PDC 10)
  - The annular fuel pebble design improves heat transfer by locating TRISO particles near the coolant allowing high operating powers while remaining within temperature limits
  - The TRISO fuel particle design has an equivalent fuel manufacturing specification as the AGR program
  - Fuel particles operate within the qualification envelope that is based on the AGR-2 irradiation and safety tests
- The fuel particle is designed with multiple barriers to constitute the primary portion of the functional containment which controls the release of radioactivity to the environment (PDC 16)
  - The TRISO fuel particle contains four barriers to the release of radionuclides
  - Pebble inspection in the PHSS ensures pebbles operate within the qualification envelope and are not damaged
  - Pebble laboratory testing confirmations that pebbles meet functional requirements, protecting the TRISO particles from damage

4.2.2 Reactivity Control and Shutdown System

ODED DORON – SENIOR DIRECTOR, REACTOR SYSTEMS DESIGN

4.2.2 Reactivity Control and Shutdown System

- Reactivity Shutdown System (RSS)
  - Credited for reactor trip and shutdown
  - 3 safety-related shutdown elements that insert directly into pebble bed
- Reactivity Control System (RCS)
  - Inserted on reactor trip, but not credited
  - 4 non-safety-related control elements that insert into reflector
- Release Mechanism
  - Safety-related electromagnetic clutch
- Drive Mechanism
  - Non-safety-related motor-driven sheave to position element
  - Provides for position indication
- Testing and Inspection
  - RCSS periodically inspected for wear
  - Reactor coolant periodically sampled for an increase in boron concentration that could indicate shutdown element cladding failure
  - RCSS elements can be replaced if necessary



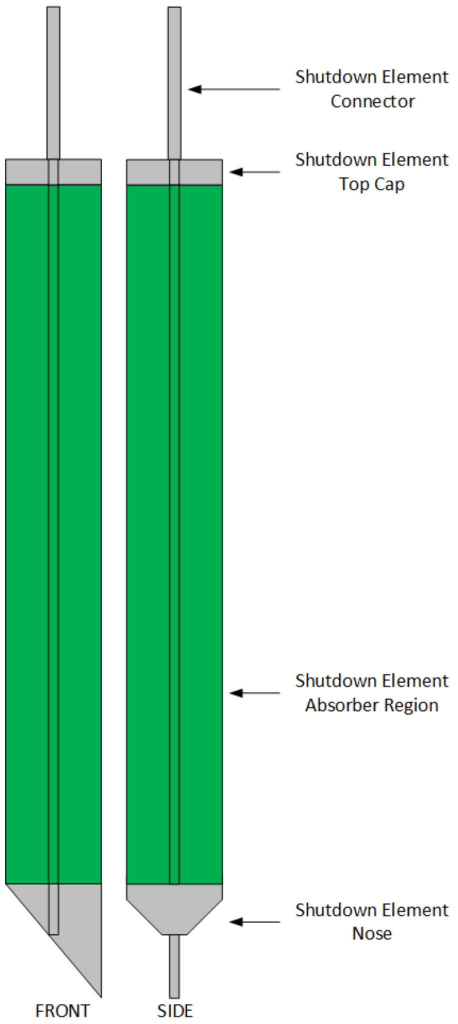
Hermes Core Layout

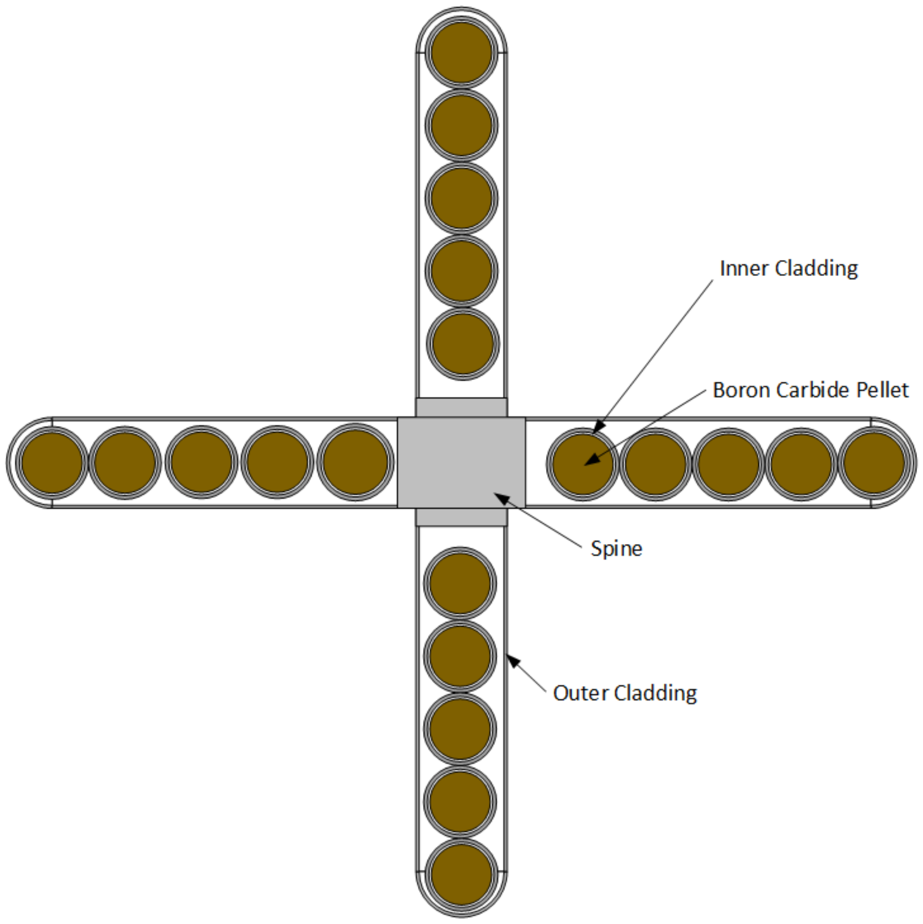
3 in-bed shutdown elements

4 ex-core control elements

4.2.2 Reactivity Control and Shutdown System: Shutdown Elements

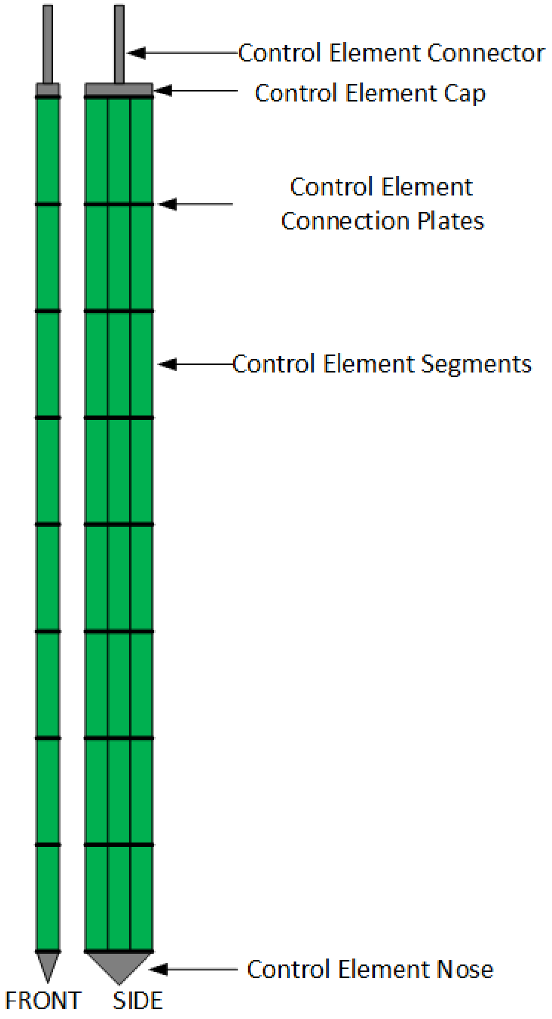
- Shutdown Element
  - Cruciform Design
  - Inner Cladding contains absorber
  - Argon fill
  - Absorber: B_4C
  - Cladding: SS-316H

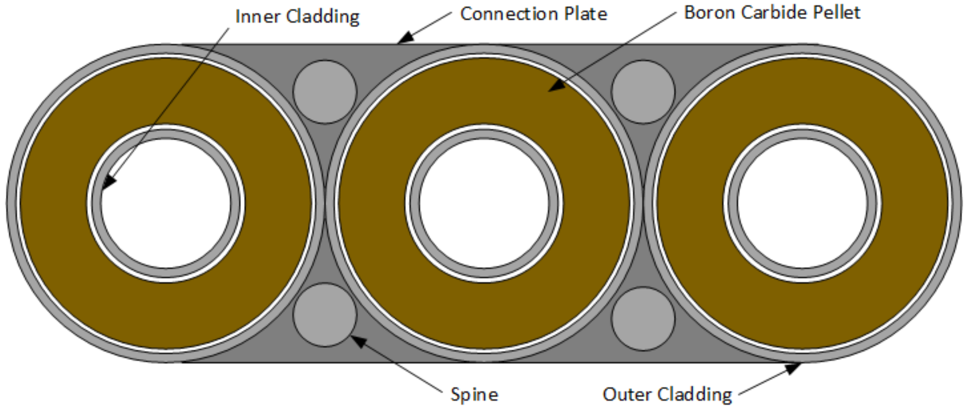




4.2.2 Reactivity Control and Shutdown System: Control Elements

- Control Element
  - Segmented Annular Design
  - Individual Capsules
  - Argon fill
  - Absorber: B_4C
  - Cladding: SS-316H





4.2.2 Reactivity Control and Shutdown System: Design Bases

- Safety-related RSS is capable of operating during an earthquake. Insertion capability confirmed via testing with maximum deflection of insertion path due to an earthquake. (PDC 2)
- RSS is compatible with environmental conditions and confirmed by qualification testing. Analysis demonstrates internal gas pressure due to irradiation does not exceed safety-related RSS element clad stress limits. (PDC 4)
- RSS is designed to fail in a safe state when the plant trips or upon loss of normal power. The energy holding relays close to remove power supply holding shutdown elements in place and a loss of power allows shutdown elements to drop via gravity. (PDC 23)
- The RCSS (RCS and RSS) meets PDC 26 (discussed in Section 4.5, Nuclear Design)
- RCSS (RCS and RSS) is designed to limit the amount and rate of reactivity insertion by controlling the maximum withdrawal speed of control and shutdown elements (PDC 28)
- The design of the RSS trip function in conjunction with the reactor protection system assures an extremely high probability of accomplishing its safety-related function. Both the RSS and the RCS provide significant negative reactivity insertion into the core via gravity and motor driven means upon a reactor trip. (PDC 29)

Hermes PSAR 4.5 Nuclear Design

NADER SATVAT – SENIOR MANAGER, REACTOR CORE DESIGN

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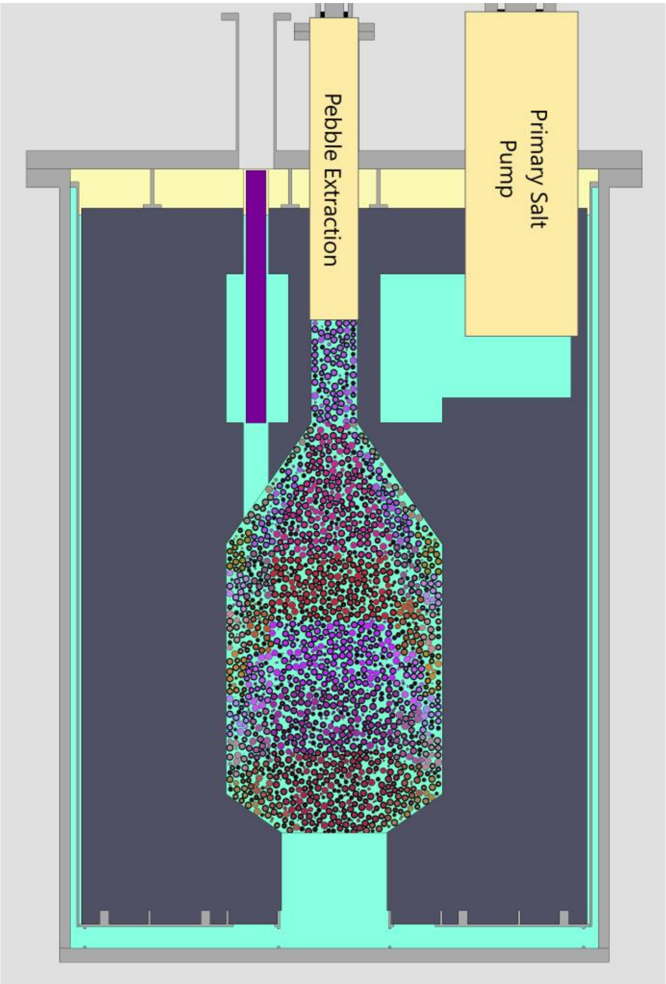
4.5 Nuclear Design

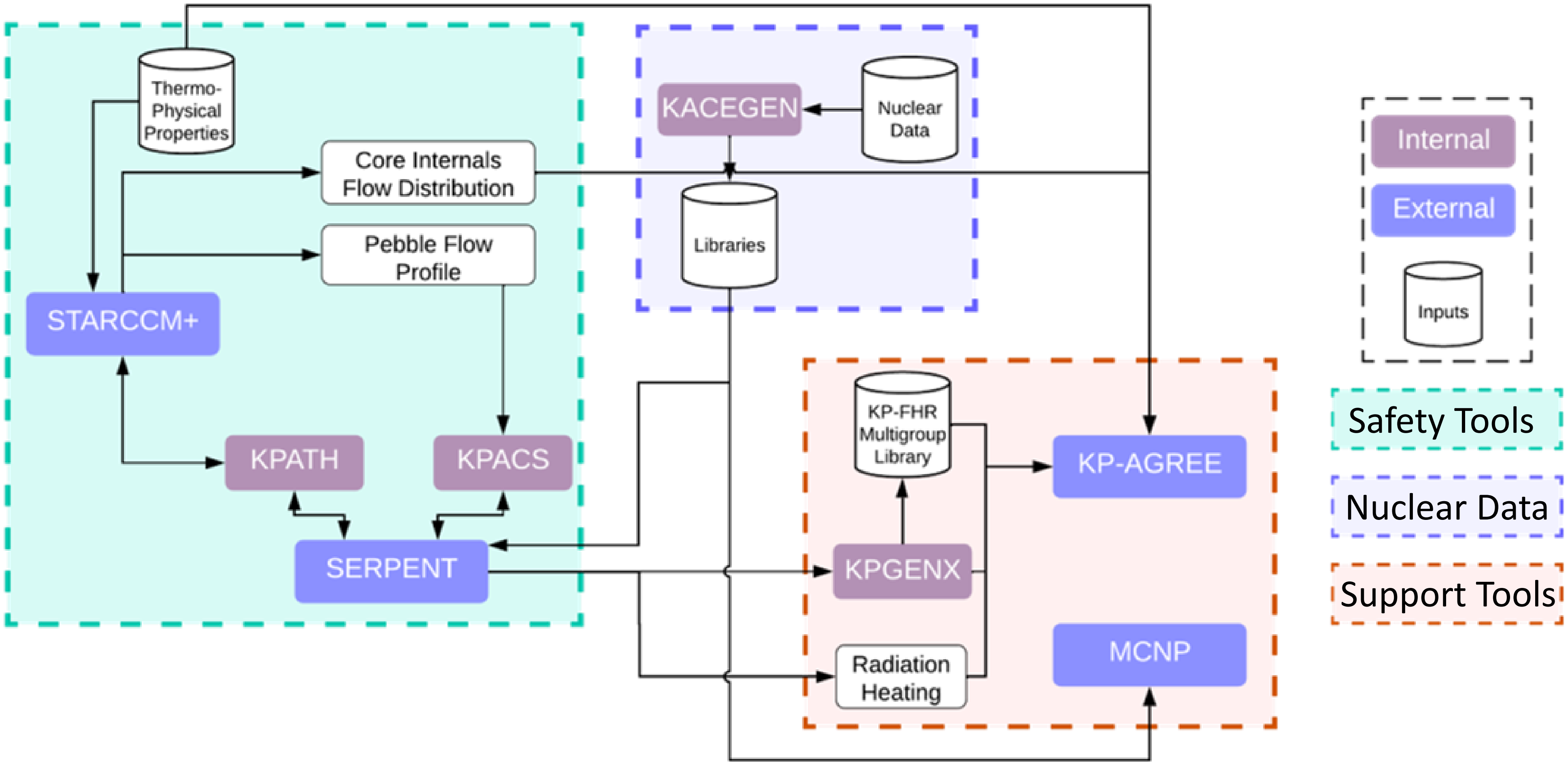
- Reactor core is a packed bed with spherical pebbles
  - Fuel pebbles contain ~6 grams of uranium
  - Fuel pebbles have enrichment up to 20 wt% U-235
  - Moderator pebbles used to improve neutron moderation
  - Core contains approximately 60% pebbles (fuel and moderator) and 40% reactor coolant by volume
  - Core is under-moderated (negative temperature and void feedback)
- Reactor core is continuously refueled
  - Both fuel and moderator pebbles are introduced into the core from the bottom by the pebble handling and storage system (PHSS) and slowly move to the top in ~30-50 days and removed from the core by the PHSS
  - Pebbles inspected for physical damage and burnup
  - Pebbles discharged as they approach their design burnup
- Reactor core is surrounded by a graphite reflector
  - Increases neutron economy, provides moderation/reflection, shields the reactor structures, and maintains the core geometry
- Core design methodology described in "KP-FHR Core Design and Analysis Methodology" (KP-TR-017)



4.5 Nuclear Design

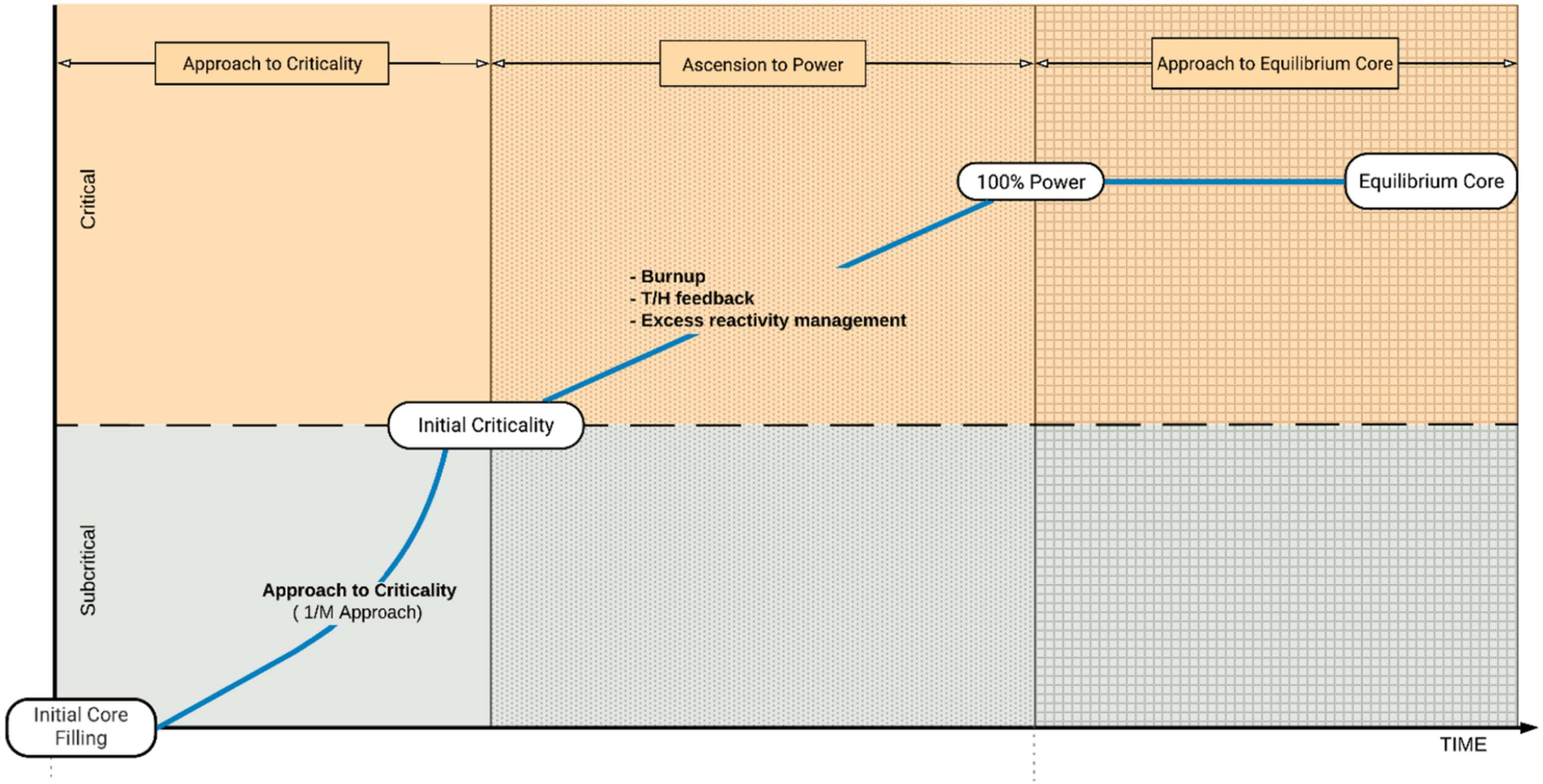
Power	35 MW_{th}
Method for Calculation	Serpent 2 (neutronics); STAR-CCM+ (DEM and T/H)
Coolant	Flibe
Shutdown margin	$k_{eff} < 0.99$
Reactivity Control Elements	7 total; 3 shutdown elements, 4 control elements
Vessel Irradiation	< 0.1 dpa
Reactor Inlet Temperature	550°C
Max Core Outlet Temperature	650°C
Core Volume	2.0 m^3
Enrichment	< 20 wt% U-235
Reactivity Coefficients	Net negative reactivity coefficient; under-moderated

4.5 Nuclear Design: Analytical Methods



4.5 Nuclear Design: Core Life Cycle

- Four cycles of life of the core:
  - Startup and approach to criticality
  - Power ascension
  - Transition to equilibrium (initial power plateau)
  - Equilibrium



4.5 Nuclear Design: Core Operational Regimes

- Approach to criticality
  - A combination of fresh fuel, natural uranium, and moderator pebbles are added into the core using 1/M approach
- Low power through ascension to power
  - Primary salt pump follows the power. Power defect, xenon, and burnup is compensated by control rods and fresh fuel addition
- Approach to equilibrium core
  - During the transition to full power, core composition will evolve: fresh fuel pebbles are added, and depleted pebbles are removed via the pebble handling and storage system (PHSS)
- All core states will operation within coolant reactivity coefficients, power per particle limits, and excess reactivity constrains

4.5 Nuclear Design: Design Basis

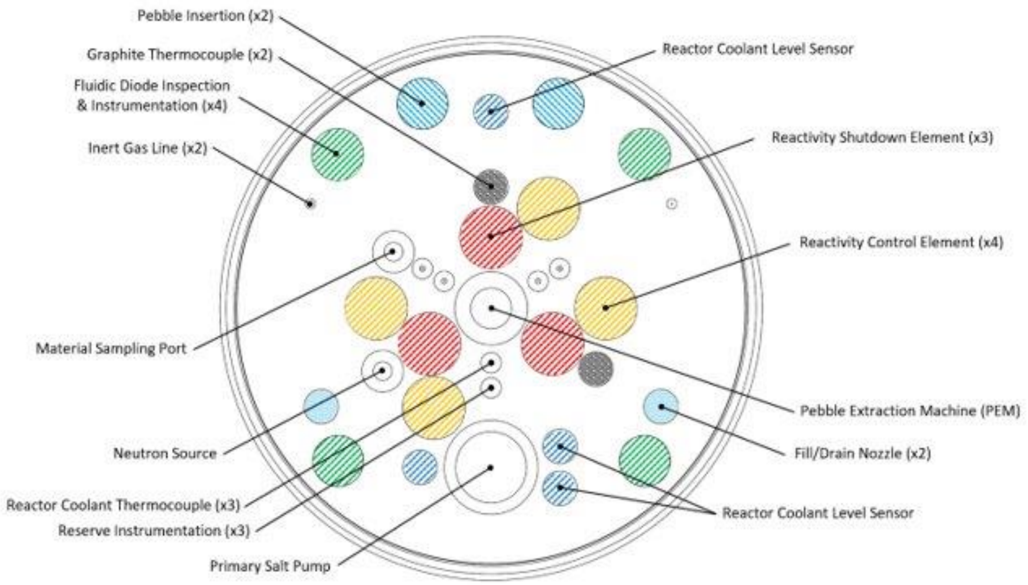
- The reactor core is designed so that the power oscillations that could result in conditions exceeding SARRDLs are not possible (PDC 12)
  - Due to the small core and the long neutron diffusion length (neutronically connected)
- The reactor core is designed so that the net effect of prompt inherent nuclear feedback tends to compensates for rapid increases in reactivity. The overall reactivity coefficient is negative. (PDC 11)
  - Large negative fuel doppler feedback
  - Positive reflector temperature coefficient due to spectrum hardening shifts flux toward core (reduces leakage) plus locally over-moderated conditions
  - Methodology used does not assume any thermal expansion of reflector (could counter-act positive feedback effect)
  - Reactivity impact due to the reflector temperature is delayed compared to fuel and coolant temperature feedback

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

4.5 Nuclear Design: Design Basis (cont.)

- A limiting power distribution for the core design is used to ensure that the reactor core has appropriate margin to SARRDLs (PDC 10)
  - Serpent 2 used to calculate power distribution using methodology described in "KP-FHR Core Design and Analysis Methodology" (KP-TR-017-P)
  - Flux distributions are verified during startup using ex-core detectors. Flux measurements compared to predicted calculations to ensure core is operating as designed.
  - There are no consequence from control and shutdown elements not being quarter core symmetric due to the small core size and long neutron diffusion length

Power Distribution	Equilibrium
Axial Peak (F_Z)	1.2
Radial Peak (F_R)	1.2
Total Pebble Peaking (F_Q)	1.8



4.5 Nuclear Design: Design Basis (cont.)

- Shutdown elements credited to provide means to ensure SARRDLs are not exceeded, and safe shutdown is achieved; met assuming highest worth shutdown element fully withdrawn. Shutdown elements insert reactivity at a sufficient rate and amount to ensure the capability to cool the core is maintained, the reactor is shut down and can be maintained in a shutdown condition; met assuming highest worth shutdown element fully withdrawn (PDC 26, Condition 1)
- Control elements provide the capability to control reactivity changes during normal power changes, ensure SARRDLs are not exceeded and provide an independent and separate means of reactivity control from RSS. Control elements are diverse from shutdown elements (different geometry, different locations, different insertion mechanisms) (PDC 26, Condition 2)
- Shutdown elements insert reactivity at a sufficient rate and amount to ensure the capability to cool the core is maintained, the reactor is shut down and can be maintained in a shutdown condition; met assuming highest worth shutdown element fully withdrawn (PDC 26, Condition 3)
- Shutdown elements provide a means of maintaining the reactor in a shutdown state to allow for fuel loading, inspection, and repair. (PDC 26, Condition 4)

4.5 Nuclear Design: Design Basis (cont.)

- The shutdown margin calculation accounts for:
  - Power defect
  - Xenon decay
  - Operational excess reactivity
  - Margin for uncertainties

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

1. Required worth considers highest worth shutdown element withdrawn (which is 6,266 pcm)

4.5 Nuclear Design: Interfaces

- The output from nuclear design is used in interfaces with other calculations
- Vessel Fluence – Supports reactor vessel design
  - Fluence on vessel accounts for core, pebble insertion and extraction lines. Fluence is attenuated by the core barrel, reflector and coolant
  - Preliminary best estimate dpa + uncertainty is within 30% of the low-level irradiation value provided in "Metallic Materials Qualification for the Kairos Power Fluoride Salt-Cooled High-Temperature Reactor" (KP-TR-013-P)
- Nuclear Transient Analyses – Supports safety analysis
  - Conservative values used for power distribution, reactivity coefficients and shutdown margin provided as initial conditions for postulated reactivity transient events
- Core Design Limits – Supports technical specifications
  - Core design parameters during normal operation are within the fuel qualification envelope for peak fluence, peak particle power, burnup and peak fuel temperature
  - Shutdown margin
  - Coolant outlet temperature
  - Moderator pebble to fuel pebble ratio