INL/RPT-25-84855

SPARC - Plans for a New Critical Experiment Facility with a Horizontal Split Table

DOE/NRC Criticality Safety for Commercial-Scale HALEU Fuel Cycle and Transportation (DNCSH)

JULY 2025

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EXECUTIVE SUMMARY

Several critical experiment facilities, sometimes referred to as zero power reactor facilities, have provided crucial data to aid understanding and validate nuclear-physics models since the beginning of nuclear technology. Indeed, the first man-made reactor, Chicago Pile-1, was essentially this type of reactor. However, there was a downturn in nuclear technology development toward the turn of the millennium, and the need for these specialized research facilities waned. Now there are few of these experimental facilities operational in the world and those that remain have relatively small critical assembly machines. The need for criticality safety benchmark experiments at intermediate neutron energy levels and the modern resurgence of interest in advanced reactors designs, many of which do not have historical precedents in terms of nuclear fuel composition, moderator, and coolant combinations, all combine to create a substantial need for a critical experiment facility with a large horizontal split-table (HST) machine. A HST machine is used to arrange two separate and subcritical parts of a core assembly, bring them together in a precise manner to achieve criticality using remote controls, and separate them to achieve a subcritical configuration again.

A new effort was recently performed to develop user needs for a HST, assess candidate locations at the Idaho National Laboratory (INL), and develop a plan for deployment. This project is referred to as the System Physics Advanced Reactor Critical facility (SPARC). A few months after this assessment began, and shortly after as a viable pathway was emerging, a series of important presidential executive orders were issued to revitalize nuclear energy in the United States (U.S.). The relevance of SPARC to these executive orders was immediately apparent. The far-reaching potential of SPARC to these executive orders will reside in its ability to produce data which facilitates licensing of advanced nuclear reactor designs while reducing uncertainties to help increase energy production alongside new criticality safety data to enable more efficient nuclear fuel manufacture, transport, and storage.

Previous U.S. Department of Energy (DOE)-funded efforts had already established the need for a HST capability and investigated experimental uncertainties that could arise from the machine itself. The present effort was built upon this work and established design requirements corresponding to opportunities for larger core volumes available in candidate INL facilities. Neutronic calculations were performed to ensure that the machine can be adequately specified for the mass and volume of core materials needed. It was found that a HST able to support a cube-shaped core at least 24,000 kg and 2.0 m in each dimension (when assembled) would be suitable to address crucial testing needs spanning the range of foreseeable experiments. These calculations and constraints formed key design inputs to equipment and facility design requirements.

A “requirements first” approach was taken to provide rigor and aid in impartiality in the process of considering facility options. Numerous resources were used to identify approximately 20 existing candidate facilities at INL. Options with untenable mission conflicts or unworkable building layouts were screened out, leaving six facilities for more detailed consideration. Consultation with facility experts, walkdowns of structures, and research of archival documents were used to assess these options. Materials and Fuels Complex (MFC)-776 received special consideration because it was the only option with the infrastructure for transuranic experiments, it once housed a HST, and ongoing facility modifications will greatly facilitate installation of a new HST. This option, however, was viewed as a longer-term “Phase II” option as it will not become available until well into the 2030s due to other priority work. Thus, assessments proceeded to identify a more near-term option.

MFC-793C and MFC-723 were both deemed feasible, but only after sizable modifications which would still pose innate characteristics that limit capability and efficiency. Two slightly better options, Test Reactor Area (TRA)-641 and Power Burst Facility (PBF)-612, had the advantage of housing significant historic nuclear missions which manifested in more advantageous building layouts. However, both posed liabilities in dealing with potential contamination during modifications and would offer an unexceptional capability package when considered broadly. Finally, PBF-613 emerged as the uncontested winner. This former research reactor building's robust floor, large basement, radiological cleanliness, ample working room, and a host of other features satisfied, and in some cases far surpassed, every attribute that had been put forth. While this facility was determined to be unexpectedly ideal, it was noted that some refurbishments and modifications would be needed. More detailed information was gathered to outline project scope and estimate costs and schedules. Like all projects, some risks and opportunities were identified as sources of uncertainty in these estimates. Still, these efforts showed that PBF-613 is not only the most technically capable option, but also the most cost-effective and expeditious option.

Based on preliminary analysis and a conceptual layout for the facility, the main project scope included installation of the reactor system, placement of a new modular building nearby to house the reactor control console, and creation of safety basis documentation to establish the facility as a DOE hazard category 2 nuclear facility. The scope for establishing the reactor system included design and manufacturing of the HST machine, procuring neutron detection equipment, and integration with reactor trip and control systems. Calculations were performed and it was determined that the most straightforward way to deal with radiation during critical operations would be to evacuate personnel to the new control room building. Similarly, calculations were performed which showed that building does not require significant structural modifications. Cost and schedule estimates were produced for this scope based on a minimum viable product approach for deploying a suitable facility. Other capability enhancements were identified, which would increase the impact, versatility, and efficiency of the project, but these upgrades were not included in the base cost estimate.

Experienced personnel produced this class 5 cost estimate including peer review and cross-checking across different organizations. While some project risks and uncertainties cannot be dismissed at this formative stage of the project, the relative simplicity of a HST system combined with the known quantities of retrofitting an existing building helped to solidify many key assumptions. The point estimate to execute the project was estimated to be $34M with slightly more than three years to achieve first criticality. Opportunities, uncertainties, and risks were also considered to derive a planning range from $30M - $45M and 3 - 5 years. Naturally, these estimates will be refined as further design work is accomplished.

Critical experiments require relevant nuclear fuels, moderators, and reflector materials in sufficient quantity (on the order of hundreds of kilograms). A few hundred fuel rods (~5% enriched UO_2 in ~1m long stainless-steel cladding) from the Walthousen critical facility at Rensselaer Polytechnic Institute are planned to be returned to INL as part of its ongoing decommissioning project. These rods are easily contact handleable and have a rich legacy in the criticality benchmarking community. The opportunity was identified to reconstitute a critical experiment configuration using these rods. This simple experiment design will have great value in supporting the inaugural SPARC experiments, supporting criticality safety training, and in conducting thermal spectrum experiments. SPARC must also be outfitted with a library of other key fuel and moderator systems to conduct experiments which address crucial data gaps. Currently nuclear fuels are not readily available in any form except Light Water Reactor (LWR) fuel with ~5% enriched UO_2 pellets in ~3.5 m long zirconium alloy tubes. Experiments focused on intermediate neutron energy spectra or on non-LWR physics will require fuels with High-Assay Low Enriched Uranium (HALEU, typically 19.75% enriched). Commercially available HALEU fuels are a nascent development but are expected to become more available as the project progresses. It will be important for the project to stay in sync with the advanced nuclear fuels community to develop plans to populate its library with HALEU options for metallic fuel, coated particle fuel, and high-density ceramic fuels.

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ACRONYMS

AI	Artificial Intelligence
ANL	Argonne National Laboratory
ANS	American Nuclear Society
BEA	Battelle Energy Alliance
CITRC	Critical Infrastructure Test Range Complex
DNCSH	DOE/NRC Criticality Safety for Commercial-Scale HALEU Fuel Cycle and Transportation
DOE	United States Department of Energy
DOME	Demonstration of Microreactors Experiments
EBR-II	Experimental Breeder Reactor–II
F&OR	Functional and Operational Requirements
FASB	Fuels and Applied Science Building
FCF	Fuel Conditioning Facility
HALEU	High Assay Low Enriched Uranium
HFEF	Hot Fuel Examination Facility
HMI	Human-Machine Interface
HP-MR	Heat-Pipe Microreactor
HST	Horizontal Split Table
HTGR	High-Temperature Gas-cooled, graphite moderated Reactor
HVAC	Heating, Ventilating, and Air Conditioning
I&C	Instrumentation and Control
IBC	International Building Code
ICSBEP	International Criticality Safety Benchmark Evaluation Project
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology Engineering Center
IRPhE	International Reactor Physics Experiment Evaluation Project
KRUSTY	Kilowatt Reactor Using Stirling Technology
LANL	Los Alamos National Laboratory
LLNL	Lawrence Livermore National Labatory
LOTUS	Laboratory for Operation and Testing in the United States
LWR	Light Water Reactor
MCRE	Molten Chloride Reactor Experiment
MFC	Materials and Fuels Complex

NCERC	National Criticality Experiments Research Center
NCS	Nuclear Criticality Safety
NCSP	Nuclear Criticality Safety Program
NDC	Natural Phenomena Design Category
NE	DOE Office of Nuclear Energy
NEA	Nuclear Energy Agency
NHS	National and Homeland Security
NMIS	Nuclear Material Inspection and Storage
NNSA	National Nuclear Security Administration
NPH	Natural Phenomena Hazards
NRC	Nuclear Regulatory Commission
NRIC	Nuclear Reactor Innovation Center
OECD	Organization for Economic Cooperation and Development
ORNL	Oak Ridge National Laboratory
ORR	Operational Readiness Review
PBF	Power Burst Facility
RIA	Reactivity Initiated Accident
SCMS	Sodium Component Maintenance Shop
SFR	Sodium-cooled Fast Reactor
SNL	Sandia National Laboratories
SPARC	System Physics Advanced Reactor Critical Facility
SPERT	Special Power Excursion Reactor Test
SPRF/CX	Sandia Pulse Reactor Facility - Critical Experiments
STAR	Safety Tritium and Applied Research Facility
TESB	TREAT Experiment Support Building
TRA	Test Reactor Area
TREAT	Transient Reactor Test Facility
TRISO	Tri-Structural Isotropic
TRU	Transuranic
U.S.	United States
USL	Upper Subcritical Limit
UZrH	Uranium Zirconium Hydride
ZPPR	Zero-Power Physics Reactor
ZPR	Zero-Power Reactor

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SPARC - Plans for a New Criticality Experiment Facility with a Horizontal Split Table

1. INTRODUCTION

This report outlines the needs and deployment plans for a critical experiment facility to be established at the Idaho National Laboratory (INL) to house a new Horizontal Split Table (HST). This project, referred to as the System Physics Advanced Reactor Critical facility (SPARC) aims to enhance national criticality experiment infrastructure, support the United States (U.S.) Department of Energy (DOE) stakeholders, and advance nuclear technology through reliable and versatile experimental facilities. The key mission is to recover capabilities which once existed in the U.S. for performing geometrically large critical experiments. SPARC will be a cornerstone facility that will continuously support cutting-edge research, adapt to evolving nuclear technologies, and provide sustained value to the nuclear community for decades to come.

Figure 1 below shows facility locations and illustrates the vision where SPARC becomes the third major criticality benchmarking site in the U.S. The existing Sandia Pulse Reactor Facility – Critical Experiments (SPRF/CX) at Sandia National Laboratories (SNL) contains a small pool type system and excels at producing water-moderated, UO_2-fueled critical experiments. For example, a series of critical experiments at SPRF/CX using 7% enriched fuel rods was central for the rapid licensing approval of recent U.S. Light Water Reactor (LWR) industry applications to transport fuel with 5 to 8 wt.% enrichment. The existing National Criticality Experiments Research Centers (NCERC) operated by Los Alamos National Laboratory (LANL) has a few critical assembly machines and a deep library of fuel materials. Both SPRF/CX and NCERC are primarily funded by the Nuclear Criticality Safety Program (NCSP) of the National Nuclear Security Administration (NNSA). Both facilities are used extensively in criticality safety workforce training and often fill every available opening in training courses offered. SPARC will offer a complimentary capability to expand upon existing capabilities at NCERC and SPRF/CX while increasing the bandwidth in the U.S. for this type of important and broadly relevant, low-uncertainty nuclear experiment, with a focus on unique large scale experiment with DOE Nuclear Energy (NE) applications. In addition to its unique large-scale criticality test capabilities, SPARC will provide additional capacity for criticality safety training to the nuclear energy workforce.

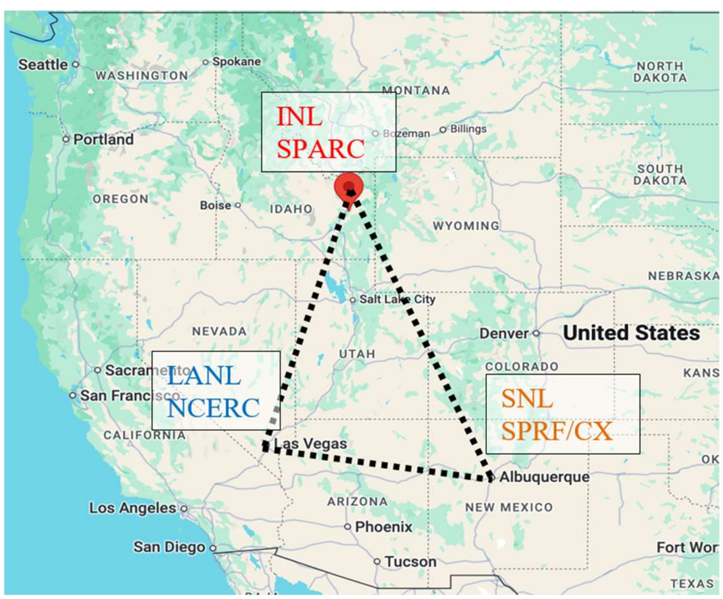


Figure 1. Vision for U.S. critical benchmark facilities with NCERC and SPRF/CX focusing on NNSA needs and SPARC on DOE-NE needs

1.1. The Need for Criticality Safety Experiments

When working with significant quantities of nuclear material, such as during reactor fuel fabrication, transportation, and storage, the potential exists for inadvertent assembly of a critical configuration. As the consequences of a criticality accident can be lethal to nearby personnel due to the large release of radiation accompanying such an event [1], Nuclear Criticality Safety (NCS) as a discipline seeks to mitigate these consequences by prevention of such an accident. Before operations begin, NCS organizations are charged with evaluating the expected and potential abnormal conditions of operations and establishing limits and controls to prevent inadvertent criticality. Historically, this evaluation was completed largely based on conducting directly relevant experiments in a dedicated critical mass laboratory [2][3] that accompanied most nuclear facilities. Large or heavy experiments, such as those that were meant to approximate storage or transportation arrays, were often assembled on HSTs. As computational power increased, radiation transport codes began to be used to analyze operations to ensure subcriticality of operations. By the mid 1990's, most operations were being evaluated for NCS with the help of computer codes instead of being based on directly applicable experimental data, and the majority of the critical mass laboratories were closed.

With the transition to relying on calculational methods for NCS came the need to validate those methods with real experimental data to build trust in the model predictions. The International Criticality Safety Benchmark Evaluation Project (ICSBEP) was established in 1995 as an activity under the auspices of the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Working Party for Nuclear Criticality Safety. The goal of ICSBEP is to provide a collection of high-quality experimental benchmarks relevant to the nuclear criticality safety community for computational method validation. The ICSBEP is a collection of evaluated historical experiments conducted at critical mass laboratories since the 1950's and also includes modern experiments that have been conducted and evaluated contemporaneously. The ICSBEP provides a standardized format and content requirements for benchmark evaluations and requires a rigorous review process. The ICSBEP Technical Review Group meets in person each year to review and add new benchmarks, with a new handbook released annually [4].

The ICSBEP is the most highly trusted source for criticality safety benchmarks. These computational benchmark models, which represent real geometric configurations of fissionable and other materials in a critical configuration, are used to test how well a radiation transport code can predict the effective multiplication factor (k_{eff}) of a system. An exactly critical system would have a k_{eff} of 1.0. There can be a number of reasons why a modeled critical system might over or underpredict k_{eff}, including errors in the code or data processing, nuclear data deficiencies, or inaccuracies in the benchmark model. The goal of an NCS evaluation, then, is to set a defensible Upper Subcritical Limit (USL) that is reduced from k_{eff} of 1.0 to account for any of these problems that might be present in the calculations. Depending on the operation and how much appropriate validation data exits, the USL for NCS evaluations is generally between 0.9 and 0.98.

Despite the ICSBEP containing over 5,000 critical benchmark cases, NCS still has many needs for additional critical experiments. There are known gaps in the existing benchmarks for intermediate energy systems for fissile nuclides and many important non-fissile structural materials (such as Fe, Ni, Mn, Cr, etc.). If new fuel forms are developed for advanced reactors, including higher enrichments up to 20% (High Assay Low Enriched Uranium, or HALEU), new alloying materials, new fuel forms, and advanced, integrated moderators, the fuel will need to be fabricated, stored, and transported while ensuring subcriticality. Many of these new materials and enrichments lack adequate experimental data to validate radiation transport codes. SPARC will help fill these validation gaps.

For technical reviews performed by the U.S. Nuclear Regulatory Commission (NRC), staff evaluate licensee and applicant criticality safety evaluations for fuel cycle facility licensing and fissile material transportation package certification. Applicants rely on applicable benchmark experiments to validate criticality models of fuel facility processes and transportation package designs under normal conditions of transport and hypothetical accident conditions. To the extent that applicant and licensee systems and processes are well validated, calculated k_{eff} values may be as high as the calculated USL which are typically determined using well-established methods reflected in the American Nuclear Society (ANS) criticality safety standards and in NRC guidance documents (e.g., NUREG/CR-7311 [5]). For systems and processes where applicable critical benchmark data is limited, applicants and licensees will typically develop conservative USLs based on non-parametric validation methods and maintain significant margin to offset the lack of relevant data. This approach reduces the throughput of fissile material processes, the capacity of fissile material packages, and the accumulation of fissile material packages on transportation conveyances. Having access to more applicable critical experiments for validating criticality safety models of fuel cycle processes and fissile material transportation designs will allow applicants and licensees to use more precise validation approaches and to maintain lower margins to their calculated USLs, especially for Non-LWR and HALEU systems. Thus, new experiments will allow for greater optimization of fuel cycle processes and transportation package designs for these materials.

One specific gap not currently filled by existing critical assemblies are large leakage dominated systems. These systems are fairly decoupled neutronically. Specific examples include storage arrays and shipping configurations. The array size of the assembly and area of validation are currently constrained by the footprint of existing critical assemblies at NCERC and SPRF/CX. Historically, large leakage dominated systems were measured on large HST machines. When these HST machines went away, so did the ability to fully validate these types of systems.

The need for additional experiments of this type was realized in the 1990s when a focused effort was undertaken on intermediate energy systems. These systems need moderate and low density configurations for neutrons to scatter without immediately being captured. Each subunit within the array has low mass volume but when many are brought together, the systems are critical. They have high neutron leakage rates, which drives them towards large masses (despite low mass per subunit). These systems are currently approximated by using moderate Z-reflectors such as copper. While this approach addresses the intermediate energy nuclear data need, it does not meet all needs for spectral relevance for storage and array applications. The HST will allow for a much larger mass, thus allowing for a wider range of interstitial materials and setup.

1.2. The Need for Reactor Physics Benchmarking

The validation of computational methods is crucial to ensure the accuracy and reliability of reactor physics simulations, which are fundamental for designing safe and efficient nuclear reactors while meeting regulatory requirements. Benchmark data, derived from well-documented and peer-reviewed experiments, provide the necessary reference points against which computational models can be tested and validated. The precision and quality of these benchmarks are paramount in reducing uncertainties in reactor physics calculations and in demonstrating compliance with safety standards. The development and use of benchmark data are vital for meeting regulatory and licensing requirements in the nuclear industry. Regulatory bodies require evidence that the computational methods used in reactor design and safety analyses are accurate and reliable. Like the ICSBEP project described above, the International Reactor Physics Experiment Evaluation (IRPhE) Project, also established under the OECD NEA, focuses on compiling and evaluating a comprehensive set of reactor physics experiments. Benchmark data from both ICSBEP and IRPhE are used to serve as reference points against which the performance of computational models can be assessed. By validating these models against well-documented and qualified benchmarks, reactor designers can demonstrate compliance with safety standards and reduce uncertainties.

The primary objective of IRPhE is to preserve and disseminate reactor physics experimental data that are crucial for validating computational methods, system response behavior, and nuclear data used in reactor design and safety analyses. The experiments cover a wide range of reactor types and configurations including thermal and fast reactors. By providing well-reviewed and high quality benchmark specifications, IRPhE ensures that historical data are retained, and that current and future measurements are accurately archived. The evaluated datasets are published in the IRPhE Handbook which serves as a valuable resource for researchers and engineers in the nuclear field.

New reactor concepts in the U.S. often rely on novel nuclear fuel, moderator, and coolant types that significantly depart from the existing database of criticality safety and reactor physics experiments and benchmarks [6]. Historically, integral experiments in Zero-Power Reactor (ZPR) facilities were used to qualify neutronics design codes and inform full-scale reactor design, providing essential data to assess potential biases in computational models [7]. Zero-power facilities, like those at Argonne National Laboratory (ANL) and INL [8]; the Eole, Minerve, and Masurca reactors in France; and the early reactors in the United Kingdom [9], played critical roles in validating and improving reactor designs. The world's first ZPR, Chicago Pile-1, was instrumental in developing power-generating reactors [10]. However, these facilities have all been decommissioned.

Few test beds exist today. The Planet and Comet facilities at NCERC are valuable general purpose test assembly, but limited in their ability to support large and heavy experiments due to their vertical lift configurations [11]. Godiva and Flattop, also operated at NCERC, are fixed small core systems and are not representative of HALEU-fueled reactors. The Fast Critical Assembly in Japan was a large HST akin to ZPPR [12], but this facility is no longer operational. Historical data from these facilities have shaped today's reactor technologies by serving as benchmarks for validating modern nuclear data and simulation codes [13]. To address the lack of data for advanced reactor design, the OECD NEA has created an international task force to review the needs for new reactor physics validation data and recommend actions [14].

Some vendors are developing low-power reactor prototypes which will include physics testing as a key mission, such as the Hermes reactor demonstration from Kairos, or the Molten Chloride Reactor Experiment (MCRE) from TerraPower. However, these reactors are focused on demonstrating system operation more than producing critical experiments, and they are tailored to specific designs which limits broader applicability. Other designers must rely on historical benchmarks for software validation. Benchmarking often reveals the need for more data on prototypic systems. To mitigate risks and facilitate advanced reactor design and operation, a new multipurpose critical facility is necessary. A new HST will enable large-scale and heavy critical experiments that are not possible with existing facilities to validate nuclear data, support nuclear criticality safety, and advance innovative reactor designs.

One key capability of SPARC will be conducting large-scale critical experiments able to provide valuable data on neutron fluence distributions, reactor kinetics parameters, thermal neutron scattering behavior, reactivity contributions from reflector elements, reactivity coefficients, temperature coefficients of reactivity (via electric heating), and control element or neutron poison worth. Accurate data for these parameters will be crucial for safety and performance assessments for all reactor designs. SPARC will facilitate experiments with novel fuel forms to validate nuclear data for new reactor concepts. Large-scale testing is particularly important for reactors moderated in part or whole by elements other than hydrogen (e.g., graphite, beryllium). In these configurations neutrons have longer mean free paths than in water-moderated reactors. In these systems, neutrons travel significant distances between interactions with the moderator or fuel, creating complex neutron fluence patterns and gradients. Large-scale tests are needed to accurately measure neutron distribution which are crucial for reactors using graphite moderation.

SPARC will also study high-leakage reactor designs like fast-spectrum reactors which have significant neutron escape from the core. As shown in Section 1.4, achieving a critical configuration without moderator materials in the active core - prototypical of sodium-cooled fast reactor concepts using HALEU fuel - leads to large critical core dimensions, which also increase the size of the reflector. Critical mockups of fast spectrum reactors will require a large HST capable of sustaining heavy materials. Representative-scale testing is crucial to understand and quantify neutron leakage effects on core reactivity, power distribution, and safety margins. Fast reactors require precise neutron leakage data to optimize core design and control strategies. SPARC will enable comprehensive experiments with the spatial and temporal resolution needed to study neutron leakage effects.

Many advanced reactor designs, particularly microreactors, incorporate control systems such as control drums or innovative reactivity control mechanisms relying on neutron absorber materials with potentially larger cross section uncertainties than current reactors. Representative-scale testing is vital to evaluate these external control systems in realistic configurations to ensure accurate modeling of their interactions with the reactor core. SPARC will enable large-scale experiments to assess these control systems' effectiveness and reliability to provide data for design optimization and safety analysis. In addition, microreactors often introduce non-traditional materials intended to enhance neutron moderation in the active part of the core to reduce the core dimensions. Thermal scattering cross-sections for metal hydrides are known to be a source of uncertainties in neutronics calculations for these concepts. SPARC will provide a testbed to design experiments specifically aiming at improving thermal scattering cross-sections and temperature coefficients for these materials.

1.3. Significance toward Executive Orders

A few months after the present effort began a series of important presidential executive orders were issued to revitalize nuclear energy in the U.S. The relevance of a large scale critical experiment capability to these executive orders emphasizes the crucial need for SPARC. The far-reaching significance of this experimental capability will reside in its ability to produce refined data to facilitate licensing of advanced nuclear reactor designs while reducing uncertainties to help increase energy production alongside new criticality safety data to enable more efficient nuclear fuel manufacture, transport, and storage. Nine key takeaways from the executive orders summarized by DOE [15] help to illustrate this point as summarized in Table 1.

Table 1: Summary of SPARC's Impact Relating to Recent Executive Orders

Takeaway	SPARC's Relevance
Speed up Nuclear Reactor Licensing • DOE to help reduce regulatory risks to accelerate licensing • Expedite NRC pathway for reactors tested by DOE	Large, reactor-scale, critical experiments to validate neutronics codes are essential to expedite licensing. SPARC will enable experiments which faithfully represent the physics of industrial reactor designs.
Add 300 Gigawatts of New Capacity by 2050 • Including 5 gigawatts of power uprates for current fleet	LWR industry has expressed serious interest in new critical experiments on full size fuel bundles to reduce pin power peaking uncertainties for advanced fuel bundles for power uprates.
Lay the Groundwork for Faster Reactor Testing • DOE to revise regulations and guidance for expedited review and approval of reactor projects	SPARC is developing its safety design strategy to be one of the first DOE-authorized reactors under this expedited framework.
Deploy U.S. Reactors for Artificial Intelligence (AI) and Military Bases • DOE will lay the groundwork for building and operating an advanced nuclear reactor supporting AI	Experts are already preparing research proposals to use SPARC as a platform to develop AI design methods of reactors "by AI, and for AI". Reactor developers for military bases have expressed support for using SPARC to expedite and optimize advanced designs.

Explore Fuel Recycling and Reprocessing • Efficiently transfer spent fuel from LWR's, evaluate recycling from DOE & Department of Defense reactors • Utilize surplus plutonium for advanced reactor fuels	Criticality testing is crucial in maximizing batch size for every part of the supply chain including fresh manufacturing, recycling, transport, and storage.
Amp up Domestic Nuclear Fuel Production • Maximize domestic production of nuclear fuel	
Bolster the American Nuclear Workforce • Emphasis on nuclear education programs	SPARC will expand the national capacity for hosting criticality safety classes to bolster the growing nuclear workforce.
Assess Spent Nuclear Fuel Management • Recommend policy on spent fuel management with consideration for advanced fuel cycles	Criticality testing on new fuel forms will be essential in safely managing spent nuclear fuel.
Expand U.S. Nuclear Energy Exports • New international agreements for peaceful nuclear cooperation	SPARC will expand collaboration under international criticality experiment benchmarks cooperations under OECD-NEA. In 2023 NEA hosted workshop "The Demise of Zero Power Reactors: From Concern to Action" [16], but since then there has been little "action". The U.S. will lead the world to action in this crucial arena starting with SPARC.

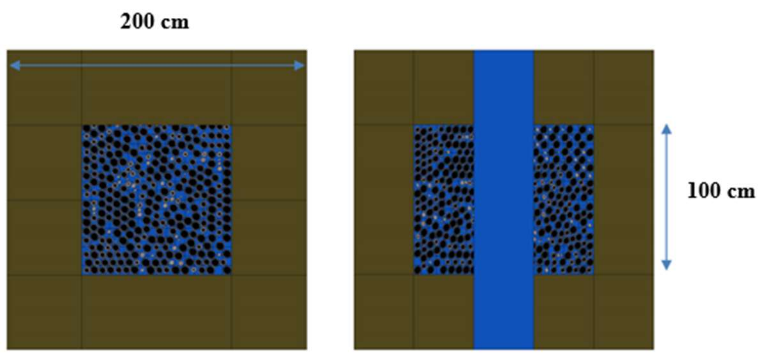
1.4. HST Requirements

Previous efforts by experts within the NCSP and DOE/NRC Criticality Safety for Commercial-Scale HALEU Fuel Cycle and Transportation (DNCSH) program created a preliminary HST design [17] and investigated the sensitivity of mechanical tolerances on the accuracy of critical experiments [18] (See Table 2 for a summary of HST specifications from this study). This previous work considered a HST constrained to fit within potential facilities with less available volume than some of the INL facilities under consideration. Thus, the opportunity and benefit of a larger HST design was investigated. A few use cases were identified to represent the largest, smallest, and heaviest use cases to develop the appropriate design input specifications.

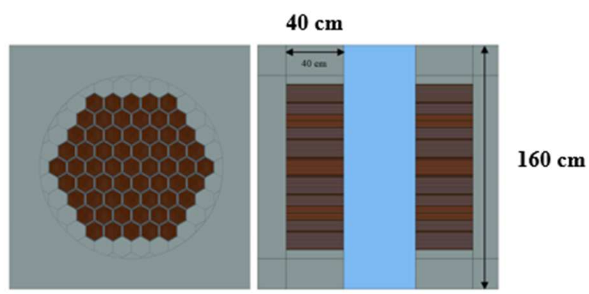
Table 2: Summary of HST Design Specification from Previous Study [17]

Parameter	Specification
k_{eff} uncertainty from the machine	5 pcm
Position Accuracy & Repeatability	0.0127 mm (gap), 0.000582 radian (angular)
Rated Load Capacity	10,900 kg
Max Experiment Height	1.52 m
Max Table Size (when closed)	1.83 m wide, 2.44 m long
Shutdown Separation Requirements	76.2 mm in 500 ms (initial) 508 mm at ≥ 0.3 m/s (full)
Design Standard	ANSI/ANS-1-2000

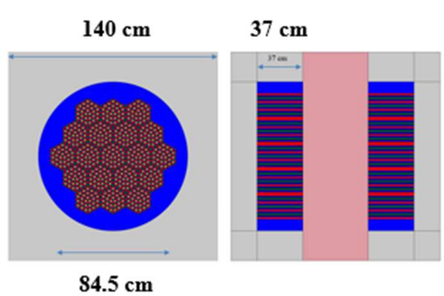
Overall dimensions and weight requirements for the HST were investigated with consideration of simplified critical experiments representative of advanced reactor concepts currently being developed by the nuclear industry, such as a pebble-bed high-temperature gas-cooled, a graphite moderated reactor (HTGR), a sodium-cooled fast reactor (SFR), a heat-pipe cooled microreactor relying on zirconium hydride (HP-MR), and a compact core relying on uranium zirconium hydride (UZrH), all fueled with HALEU. Minimum core size and material masses were determined. To provide sufficient reactivity margin to cover uncertainties due to nuclear data, impurities, mechanical tolerances, etc., the target k_{eff} value was set to 1.02. Among the different configurations studied, the largest critical core configuration was found for the HTGR case, which has a total size of around 200 cm × 200 cm × 200 cm. The SFR experiment was found to be the heaviest with a total mass of 24 metric tons due to the significant uranium mass required and large stainless-steel reflector. Experiments using hydrogenous moderators required much less uranium. For example, the UZrH case only required 24.5 kg of uranium, 49 kg for the HP-MR configuration, and 129 kg for the pebble-bed core. Images of some of the experiment concepts predicted are shown in Figure 2.



Largest Concept (Graphite Reflected TRISO Pebble Bed)



Heaviest Concept (SST Reflected U-10Zr Fast Spectrum)



Smallest Concept (BeO Reflected TRISO w/ ZrH & Heat Pipes)

Figure 2. Experiment Concepts used to Determine HST Specifications

2. FACILITY EVALUATION

2.1. Review of Other Critical Experiment Facilities

Critical experiments conducted using zero-power reactors have been crucial in understanding nuclear physics and validating predictive models since the inception of nuclear technology. HST machines have been used to support some of these experimental facilities. HSTs are used to hold two separate and subcritical sections of a core assembly and then bring them together in a controlled manner to achieve criticality. Contrasted with vertical lift machines, HSTs are needed when large and heavy core materials are of interest since their direction of motion does not oppose gravity. There have been many critical experiment facilities over the years [19], but the U.S. no longer operates any HST capabilities. Some historic and current facilities are reviewed here because they have particular relevance to a modern HST. Some of the ZPR series are reviewed because they were HST-type facilities at relevant size scales. The capabilities of SPRF/CX and NCERC are also reviewed because these facilities are operational today under modern safety bases and represent a current view on conducting criticality experiments.

2.1.1. Historic HSTs in the ZPR Series

Between 1955 and 1990, ANL played a pivotal role in the development and operation of four HST critical facilities: ZPR-3, ZPR-6, ZPR-9, and the Zero-Power Physics Reactor (ZPPR) [20]. These historic HST machines were specifically designed to analyze and validate the physics properties of fast-reactor designs. ZPR-3 was the first and smallest of the HST critical facilities at the National Reactor Testing Station, located at what is now INL. ZPR-3 enabled critical reactor experiments, particularly for the Enrico Fermi fast-reactor, providing invaluable data on fast-reactor physics, criticality, and power distributions. ZPR-6 and ZPR-9 were larger HST critical facilities which operated from the early 1960s to early 1980s at ANL-East in Illinois. These facilities conducted experiments representing various fast reactor designs to ensure the accuracy of computing methods. ZPR-6 supported nine major reactor design experiment campaigns, providing essential data for critical configurations, reaction rate distributions, control rod worths, and kinetic properties. ZPR-9 focused on other reactor types including nuclear rocket designs, gas-cooled reactor designs, and the Fast Flux Test Reactor. ZPR-9 provided detailed experimental data to understand discrepancies between measured and calculated fast-reactor physics properties and played a crucial role in criticality safety benchmarks by offering data for validating computer codes for the U.S. NRC.

ZPPR operated from 1969 to 1990 at what is now INL. It was the last and largest in the series of HST critical facilities designed by ANL. ZPPR featured a split-table with a 14 ft × 14 ft × 10 ft stainless-steel honeycomb matrix for loading reactor materials, enabling versatile configurations to simulate various fast-reactor designs. The facility's ultra-stable tables and precise measurement capabilities met stringent requirements. ZPPR supported a wide range of reactor designs. It generated high-quality experimental data to validate and refine nuclear design codes and methods, thereby advancing the development of safe and efficient nuclear reactors. A photo of ZPPR can be seen in Figure 3.

Figure 3. Photo of ZPPR

2.1.2. Currently Operating Critical Experiment Facilities

SPRF/CX is located at SNL in New Mexico [21] and is a small but well-utilized facility for criticality safety research on water-moderated configurations. Like all operating critical experiment facilities in the U.S., the current safety basis for SPRF/CX is complaint with the ANS-1 Standard [22]. Safety protocols within the facility are managed through a "two out of three" rule: only two of the following conditions can be met at any time: (1) personnel in the assembly area, (2) fuel in the core, and (3) water in the core. The reactor is housed above-grade in a water tank adjacent to and elevated above a dump tank. A reactor trip signal or loss of power will cause the tank to drain in a matter of seconds to become subcritical. SPRF/CX has a structure over the core that supports drive mechanisms and magnetic latches for B_4C-bearing control rods. The core typically operates at a flux level corresponding to around 10 W thermal energy, although the core is authorized up to 100 W. The internal structure of the core includes interchangeable aluminum spacer grids for different fuel pin pitches.

SPRF/CX has hosted several critical experiments such as the Burnup Credit Critical Experiment which measured the reactivity effects of specific fission products using zirconium clad 4% enriched UO_2 pins. The most recent series of experiments, the Seven Percent Critical Experiment series, used aluminum clad 7% enriched UO_2 pins to support the validation of reactor physics codes for commercial reactor fuel elements with increased enrichment [23] several of which have been developed into formal benchmarks for the ICSBEP Handbook [4]. SPRF/CX has made significant contributions to the criticality safety community by providing data used for validating criticality calculations. A photo of SPRF/CX can be seen in Figure 4.

Figure 4. Photo of SPRF/CX

NCERC, located within the Nevada National Security Site and operated by LANL, is dedicated to advancing nuclear research, safety, and security. NCERC is currently home to four critical assembly machines: Godiva IV, Planet, Flattop, and Comet. These machines enable the facility to conduct a wide range of criticality safety experiments [24]. NCERC is able to perform experiments on bare or encapsuled fissile material across a range of neutron spectra with different moderator and reflector types. The facility employs highly qualified personnel, who possess specialized training and expertise in nuclear engineering, criticality safety, and reactor operations, thus ensuring that all activities are conducted safely and effectively [25]. NCERC's Comet and Planet machines are vertical lift machines which have many features that are notable for present purposes. Critical experiments on these machines are also conducted in accordance with the ANS-1 Standard [22]. Neutron detectors mounted outside the core are used to measure neutron multiplication during approaches to critical. A trip signal or loss of power will cause two pressurized actuators to no longer resist gravity so that the lower half of the core reliably moves downward. Another key features to facilitate critical experiments at NCERC is onsite storage for core materials which allows efficiency in staging materials and switching between fuel and material systems.

NCERC criticality experiments include both subcritical and critical measurements. Neutron noise measurements are also used to determine kinetic parameters (e.g., delayed neutron fraction and prompt neutron generation time) using Rossi-alpha, Feynman-alpha, and power spectral density methods. NCERC maintains on site gamma spectroscopy capabilities for measurement of neutron dosimeters (e.g., flux foils/wires) irradiated in or near the critical assemblies. Recent activities at NCERC include the Deimos experiment which measured the performance of high-assay low-enriched uranium fuel fabricated in the form of tristructural isotropic (TRISO)-bearing graphite rods [26], and the Kilowatt Reactor Using Stirling Technology (KRUSTY) experiment which simulated a small heat pipe cooled reactor design for space applications [27]; both of these experiments were performed on Comet. In addition, the Planet machine was used to host the Hypatia experiment, which was employed for multiple purposes including nuclear data validation for an yttrium hydride moderator [28]. A photo of the KRUSTY experiment on the Comet machine can be seen in Figure 5.

Figure 5. Photo of KRUSTY on the Comet machine

2.2. Required and Desired Facility Attributes

The following requirements were developed based on historic and existing facilities combined with expert judgment for future needs to aid in the assessment of candidate facilities. These attributes are not viewed as the definitive engineering requirements but were put forth to give context and rigor to the process of assessing candidate facilities.

- **Required.** A facility that currently is or can be upgraded to a Hazard Category 2 Nuclear Facility to allow for critical experiments.
- **Required.** A facility with a floor-load capability where the HST can be located having at least 1,000 lbs/ft^2 to support the weight of the fully loaded table.
- **Required.** A facility with an available floor space of at least 25 ft × 30 ft in either a below-grade area (e.g., pit/basement) or an above-grade area where walls are adequately shielded. Alternatively, a non-shielded above-grade room could be workable if floor space of at least 30 × 35 ft were available to allow for supplemental shielding.
- **Required.** A facility that has an overhead crane or access for some other means (e.g., forklift, portable gantry crane) to install heavy components such as the HST and core block assemblies.
- **Required.** A facility with adequate space and services to support necessary alarms and monitoring for radiologic work (e.g., radiation detection).
- **Required.** A facility that has or can be refit with an appropriate life and fire safety system. If the fire safety system is a water sprinkler system (most likely), then a large sump or depressed area below the elevation of the HST is highly desired to mitigate the risk of accidental criticality through flooding.
- **Required.** A facility that can be secured adequately for storage of HALEU materials and neutron sources with floor space large enough to stage one full core quantity of materials.
- **Highly Desired.** A facility with adequate heating, ventilating, and air conditioning (HVAC) system and weather seals to ensure that HST operations can be performed at habitable and controlled temperatures without excessively humid, dusty, or dirty environmental conditions.

- **Highly Desired.** A facility with a high ceiling and floor space surrounding the HST with a significantly larger floor space than listed above to reduce neutron "room return" for more precise critical measurements.
- **Highly Desired.** A facility with suitable floorplan/structure or adjacent site area so that a reactor control room/building can be installed nearby without requiring significant radiation shielding for personnel.
- **Desired.** A facility that has or can be straightforwardly adapted to have negative pressure ventilation systems to pull air away from the reactor and exhaust through a filtration system, which is needed to enable experiments on bare uranium materials.
- **Desired.** A facility with extra floor space, rooms, or racks for storing more than just one core load of materials (onsite storage for a library of key fuel systems, moderators, reflectors, etc.).
- **Desired.** A facility with quality-of-life features such as a bathroom, break room, computer network, and office space.

2.3. Facility Options Screening

Several facilities at INL were identified for further consideration. The INL site has a rich history of numerous test reactors and supporting nuclear/radiologic facilities. While some facilities have been decommissioned and demolished, there are several existing facilities that were worth considering. Naturally, a new facility construction could have been envisioned to meet the attributes listed above, but the strategy taken here was to only proceed with the new construction option if all existing facility options were found to be unworkable. The three key site areas that were considered include the ATR Complex, the Materials and Fuels Complex (MFC), and the Critical Infrastructure Test Range Complex (CITRC). Some consideration was also given to a few facilities at the Idaho Nuclear Technology Engineering Center (INTEC). The location of these areas on the INL Site is shown in Figure 6.

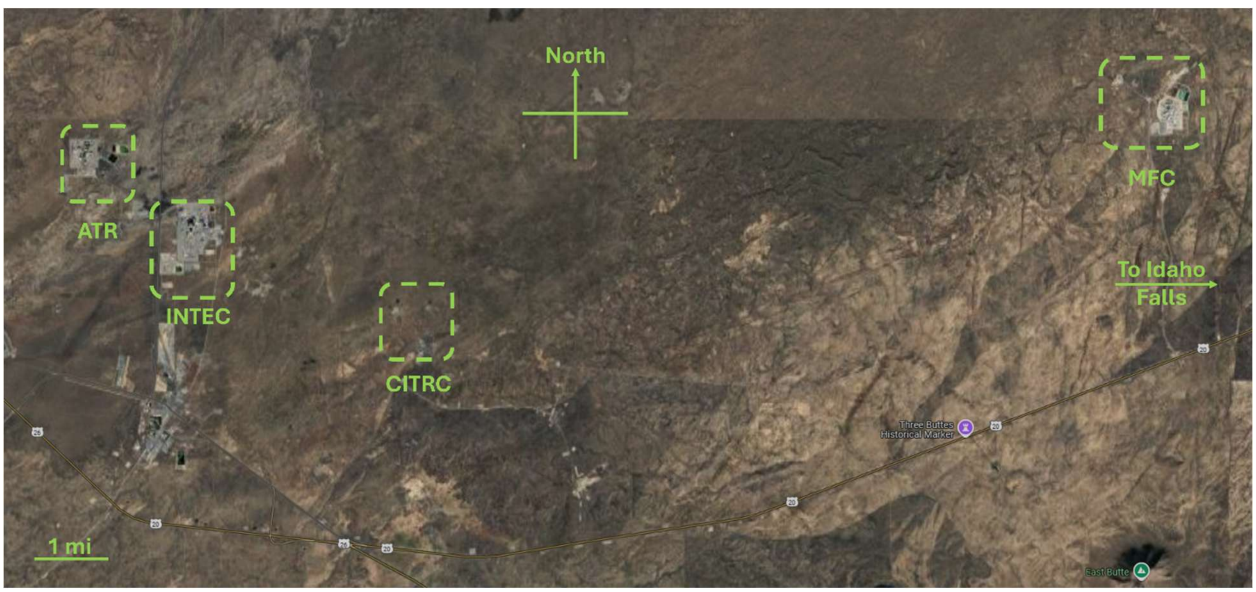


Figure 6. Locations of ATR, INTEC, CITRC, and MFC facilities(base image from www.google.com/maps).

There were several buildings to consider at each of these site areas. Site maps and facility lists were reviewed and discussions with personnel having decades of INL work experience were consulted to ensure that all credible facilities were considered. Once a list of facilities was collated, the team discussed each with the aim of determining which facilities were obviously not workable. In some cases, facility managers and documents were consulted further. In other cases, walkdowns were conducted to further understand the state of these buildings. Table 3 describes the buildings that were screened out and briefly captures the rationale for the decision.

Table 3. Facilities that were considered and screened out.

Facility	Facility Image
MFC-767, Experimental Breeder Reactor–II (EBR-II): This iconic reactor building is presently being outfitted by the Nuclear Reactor Innovation Center (NRIC) program to become a facility referred to as the Demonstration of Microreactors Experiments (DOME) testbed. Part of this process will establish DOME as a Hazard Category 2 Nuclear Facility and install many other useful features for hosting a large HST. However, there are at least a few microreactor developers already queued up to perform somewhat lengthy campaigns in DOME. These users will likely constitute a multidecadal test program, thus making DOME unavailable in the near future.	
MFC-768, EBR-II Power Plant: This large building was once used for the EBR-II power plant equipment. A sizable effort would be required to remove legacy equipment to make room and there would be inadequate floor-loading capacity to place a large HST on the second-floor area. This facility does not have other attractive features to justify taking on this burdensome task.	
MFC-720, TREAT: The Transient Reactor Test facility (TREAT) building is large and presently operated as a Hazard Category 2 Nuclear Facility. However, areas large enough for a HST are used for the storage of transient experiments in sub-grade holes or used to lay down shield blocks as part of frequent core-handling evolutions. One medium-sized storage pit in the TREAT north high-bay is to be outfitted with additional shielding and equipment to host a microreactor, but this pit is not big enough to accommodate a large HST. The TREAT building is also evacuated frequently for transient operations, which would complicate core assembly work for SPARC. TREAT is a busy facility, and the operational cadence of its existing missions is not conducive to hosting critical experiments.	

Facility	Facility Image
MFC-785, Hot Fuel Examination Facility (HFEF): The HFEF building is large and presently operated as a Hazard Category 2 Nuclear Facility. Most of the available space in HFEF is devoted to a large shielded hot cell and associated operations. HFEF has a third-floor area with a total weight rating not conducive to supporting a fully loaded large HST. HFEF is a busy facility, and the operational cadence of its existing missions are not conducive to hosting critical experiments.	
MFC-765, Fuel Conditioning Facility (FCF): The FCF building is large and presently operated as a Hazard Category 2 Nuclear Facility. Most of the available space in FCF is devoted to a large, shielded hot cell and associated operations. The only conceivable floor space adequate to host a large HST in FCF is the former remote handling mock-up area, which is presently being used for other important missions.	
MFC-787, Fuels and Applied Science Building (FASB): FASB has a vault area with some small hot cells, the need for which could be alleviated with the opening of the new Sample Preparation Laboratory and perhaps allow these cells to be removed so that a large HST could be installed. However, this room is probably not quite large enough and existing massive doors/walls combined with other gloveboxes/equipment in use at FASB make it difficult to conceive a means to install a large HST without significant building modifications.	
MFC-793, Sodium Component Maintenance Shop (SCMS): SCMS is an attractive facility for hosting a large HST in some regards. Its below-grade pit simplifies shielding approaches and side wings could be used for storage and a reactor control room. However, SCMS is presently under serious consideration to host a different reactor project, perhaps followed by others, and attempts to ascertain the availability of this building would likely result in delays as mission priorities are sorted out.	
INTEC: CPP-691, Fuel Reprocessing Restoration Facility: This large facility was constructed but never fully commissioned for fuel processing missions. Presently it represents a large and well-shielded structure that could be useful for hosting a large HST, but the facility is in limbo regarding its future use. This facility is presently managed by the Idaho Environmental Coalition as part of the "cleanup project" rather than INL's contractor, Battelle Energy Alliance, LLC (BEA). Transferring building ownership would be complicated and time-consuming as it would probably require renegotiation of DOE contracts.	

Facility	Facility Image
INTEC: CPP-651, Material Security and Consolidation Facility: Unlike CPP-691, this INTEC facility is managed by BEA and in active use for nuclear material storage. It is a well-shielded Hazard Category II facility, but the interior is divided into smaller cubicles with massive concrete walls and inadequate floor space for a large HST. It is noted, however, that this facility can be used to store fuel materials that could be used at SPARC.	
ATR Complex: Test Reactor Area (TRA)-670: This large building is currently operated as a Hazard Category I nuclear facility and not only houses the ATR itself, but also the ATR Critical (ATR-C) facility, which is presently INL's only low-power reactor used for physics testing. Unfortunately, there is no floor space in this building that could conceivably be used to support a large HST as nearly everything in ATR is devoted to the utilization of this landmark test reactor. ATR is a busy facility, and the operational cadence of its existing missions are not conducive to hosting critical experiments.	
ATR Complex: TRA-634, ATR Storage Facility: This warehouse-like building is filled with decades worth of items and equipment. Some of which could likely be dispositioned, but much probably needs to be retained to support the operation and maintenance of ATR itself. It is unclear whether enough equipment could be removed to clear enough floor space to be adequate for a large HST and, even if it could be, the effort to determine what needs to be retained would be a long process. The building does not have other features that make it attractive enough to justify taking on this onerous task.	
ATR Complex: TRA-621, Nuclear Material Inspection and Storage (NMIS): This building is currently operated as a Hazard Category 2 Nuclear Facility. NMIS is a secure facility, which would pose an inefficiency in operating a large HST, but without the benefits that other secure facilities (e.g., ZPPR) pose in being able to accommodate transuranic (TRU) bearing critical experiments. It is also unclear whether enough floor space could be arranged to accommodate a large HST.	
ATR Complex: TRA-666, Safety Tritium and Applied Research facility (STAR): This building contains several pieces of actively used research equipment and is the centerpiece of the INL fusion research program. There is little unused space in STAR and it is difficult to conceive of an area where a large HST could be installed.	

Facility	Facility Image
ATR Complex: TRA-689, Dynamic Learning Facility: This facility is partially shielded by soil and rocks and is not overly utilized by other missions. The interior of this building, however, is divided into four large rooms, two in each wing, with a central hall. All of these rooms are divided by massive concrete walls. Each of the four rooms contains a large tank, which could probably only be taken out if part of the roof was removed. It is difficult to conceive of a straightforward modification to this building that would open up floor space for a large HST.	
ATR Complex: TRA-609, Compressor Building: This large building has two sides with a central hall. Either of the sides could conceivably house a large HST, but only after a major demolition project to remove large compressors and electrical equipment. This building would then require shielding to be installed in several areas to support the operation of a critical assembly. The building does not have other features that make it attractive enough to justify taking on this onerous task.	
ATR Complex: TRA-605, Warm Waste Treatment Facility: This large building is in active use and in good repair considering its age, but its ground level is divided into smaller areas by thick concrete walls, which does not leave adequate room for a large HST. Its upper level is more open in terms of floor space but does not have a clear way of installing a large HST or the core materials that will go into it.	

2.4. Assessment of Candidate Facilities

After eliminating the above facilities from further consideration, there were a total of six remaining facilities that appeared to be credible candidates. More detailed layouts and operational considerations were developed for each of these six facilities so that their advantages and disadvantages could be weighed in greater detail as described in the subsections that follow.

2.4.1. Candidate 1: MFC-776, Zero-Power Physics Reactor

The building that housed the historic ZPPR (see Figure 2) remains operational at INL to this day. Unfortunately, the ZPPR split table itself was destructively dismantled and much of the fuel inventory has been removed. Still, the facility is operated as a Hazard Category 2 Nuclear Facility and retains a serviceable 5-ton polar crane in a well-shieled vault. One key challenge with using ZPPR for a new HST is that there is no easy way to put a new HST machine into the facility. The NRIC program is planning to modify ZPPR to become an advanced reactor testbed similar to DOME. In the coming years, ZPPR will be transformed into the Laboratory for Operation and Testing in the U.S. (LOTUS) testbed, which will include a new tunnel and removable gantry crane to allow heavy equipment to be installed into the vault. LOTUS will also include a reactor control room refit and upgraded building heat removal system.

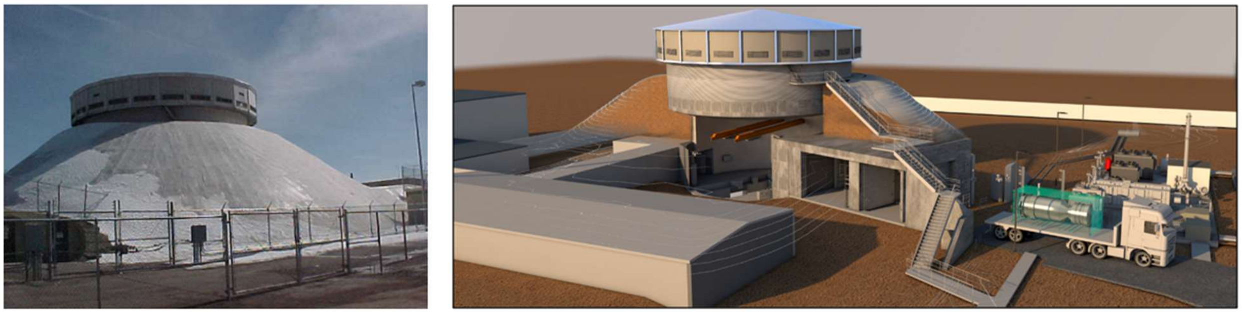


Figure 7. ZPPR building exterior photo (left) and cutaway rendering of LOTUS modifications (right).

The trouble with establishing LOTUS as the site for new critical experiments is that a well-established effort is currently underway to install and operate the MCRE in this building. MCRE is a first-ever fast spectrum molten salt reactor and will take some time to install and commission before conducting a rich test plan, and then eventually decommissioning. As with any first-of-a-kind effort, the schedule for MCRE carries large uncertainties, but it is estimated that LOTUS will not be available until well into the 2030s. Thus, the present strategy is to consider LOTUS as a potential Phase II installation with enhanced capabilities for TRU-bearing materials. As a result, other facilities need to be considered for a near-term HALEU mission.

A key requirement to enable this tactic is that large equipment, most notably the HST itself, must also be designed for installation into LOTUS. This same feature would allow for the HST to be temporarily removed from LOTUS should another reactor demonstration project be needed to make use of LOTUS further in the future. It is also worth noting that facility security requirements at LOTUS necessitate additional resources that would cause it to be an unnecessarily expensive location for conducting HALEU experiments. So, it is conceivable that the ideal future situation would have a large HST available permanently at the Phase I facility and a second HST installed in LOTUS between reactor demonstrations as needed to conduct TRU-bearing experiments.

2.4.2. Candidate 2: MFC-723, TREAT Experiment Support Building

Building MFC-723 or the TREAT Experiment Support Building (TESB) is used to store equipment and materials to support the neighboring TREAT facility. TESB has enough floor space to support an above-grade set up for the HST with supplemental shielding, but the core itself would be fairly close to the shielding which would increase neutron return. A small area somewhere in the building also would need to be dedicated to a reactor control console, or a modular building would need to be used as a control room. The building is essentially just a warehouse with two roll-up doors, but upgrades are underway to upgrade its HVAC system for improved human habitability. The building has no other notable features, but it is close to TREAT which has bathrooms. It has no crane and is probably not tall enough to permit overhead hoisting of large items on the HST. Due to the proximity to TREAT, this building must be completely evacuated for a few hours whenever TREAT is performing transient operations, which can be a daily occurrence during the normal work week. This continual interruption of work would be a source of inefficiency for assembling critical experiments. This building has never been categorized as a reactor building or considered a nuclear facility. The concrete pad in TESB is 6 in. thick, which probably cannot support the weight of a fully loaded or supplemental shielding, so some demolition of existing concrete to pour a thicker pad would be needed. This option could require significant modifications, have considerable operational inconveniences, and offers little beyond adequate floor space. Some images of this building are shown in Figure 8.

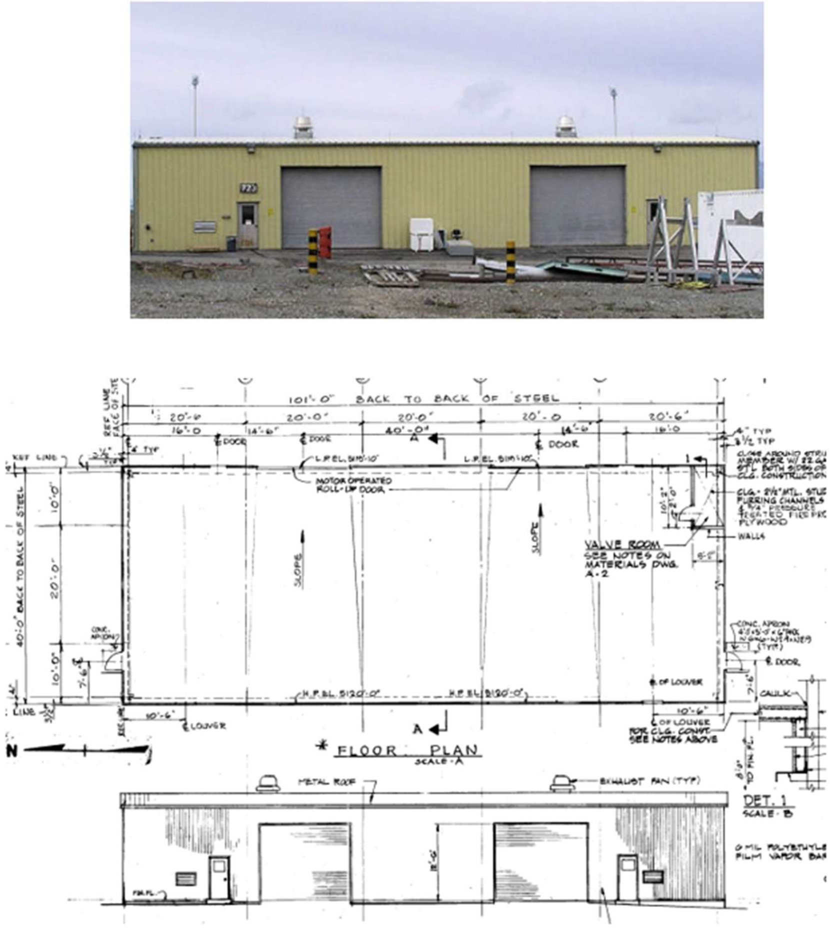


Figure 8. Photo of MFC-723 (top) and historic building drawing (bottom).

2.4.3. Candidate 3: MFC-793C, SCMS Storage Building

Building MFC-793C was constructed in 1959 as a storage building to support the previously mentioned SCMS facility. This building has a pit with is too small to house a HST below-grade. Sodium-contaminated legacy equipment would need to be removed from the pit so that it could be filled or covered to provide adequate floor space. Like TESB, this option would require supplemental shielding surrounding the HST which could block egress paths and could require special provision for fire safety. This building does not offer much extra floor space to house a control console so a modular building would be needed to establish a control room. The building itself has no bathrooms or office space, but it is within the MFC fence where all necessary amenities are within walking distance. This building has no overhead crane nor space to support a retrofit or alternate means of overhead handling. This building has never been categorized as a reactor building or considered a nuclear facility. This option could require significant modifications and barely offers enough floor space for the minimal conceivable HST layout with no extra room for other beneficial features. A key advantage of this option however, compared to MFC-723, is that it would not need to be evacuated whenever TREAT prepares for transient operations. Images of this building are shown in Figure 9.

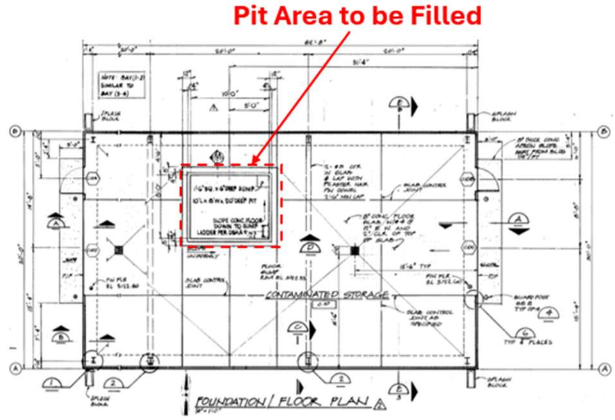


Figure 9. Photo of MFC-793C (left) and historic building drawing denoting pit location (right).

2.4.4. Candidate 4: TRA-641, Core Internal Changeout Staging Area

TRA-641 was constructed in 1955. It is presently known as the Core Internal Changeout Staging Area, although it is not presently used for anything except storage of some equipment. This building was originally constructed to perform gamma irradiation studies using irradiated nuclear fuel as a gamma source. While this mission was not technically a reactor, it housed a significant amount of radioisotope inventory, which could be helpful in streamlining environmental impact assessments. The irradiated fuel was once submerged in a canal that would now need to be filled or covered to provide enough continuous floor space for a HST. Presently, this canal is posted as a contamination area that could make it onerous to modify. The aspect ratio of the 40 ft × 60 ft building, the location of the roll-up door, and the 1,000 lbs/ft^2 floor capacity lends to a layout that could provide reasonable spacing between the HST and supplemental shielding while providing room to maneuver equipment and materials (presuming that a modular building would be used as the reactor control room). A crane partially traverses the canal area and could potentially be extended further to assist with handling materials to be placed on the table. The building has a single small bathroom that is in poor repair, but the building has adequate electrical service for supporting the HST and auxiliary equipment. The building is within the fence of the ATR Complex and within walking distance of other useful facilities and amenities. Some images of TRA-641 are shown in Figure 10.

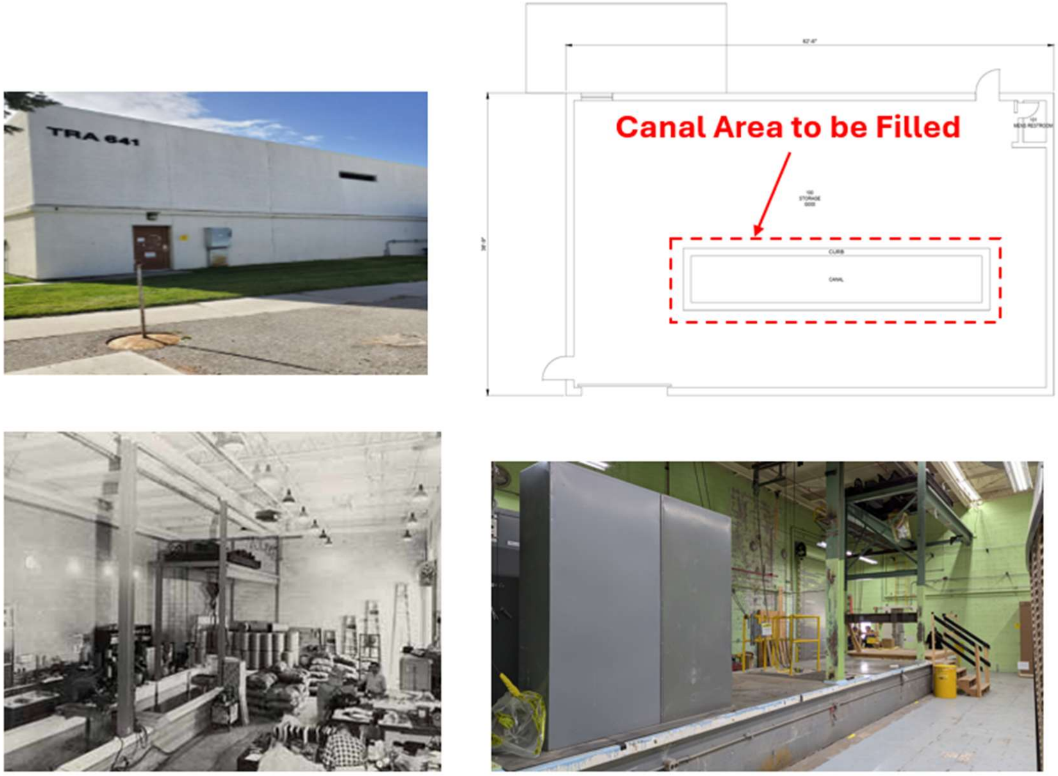


Figure 10. TRA-641 exterior photo(top left), overhead drawing denoting canal area (top right), historic photo inside (bottom left), and modern photo inside (bottom right).

2.4.5. Candidate 5: PBF-612, CITRC Control System Research Facility

Power Burst Facility (PBF)-612 was built in 1960 and is the site of the historic Special Power Excursion Reactor Test (SPERT)-II [29]. The reactor was decommissioned, and all major equipment was removed years ago, but the building remains and is actively used by INL’s National and Homeland Security (NHS) directorate for a variety of training exercises and research on electrical equipment resiliency. The building has a functioning 10-ton overhead crane, bathroom, HVAC system, and various rooms/areas that could be used for other purposes. The building’s lowest basement level is where the SPERT-II reactor vessel was once housed and is now occupied by multilevel metal decks, stairs, and other equipment that would need to be removed for access to install and operate the HST in this part of the building. The first basement “heavy water room” is probably a more attractive location due to easier overhead access and its massive 10,000 lbs/ft^2 floor capacity. All of its walls would probably require some neutron shielding to reduce room return. Still, the 27 × 25 ft room size is barely big enough and a central pillar would need to be removed to place the HST centrally in the room. The metal decking ground floor above the heavy water room would need to be modified to create an overhead access hatch large enough for the HST. The heavy water room is posted as a fixed contamination area, which could encumber any modifications that disrupt surfaces. PBF-612 is currently operated as a radiologic facility. The building once housed a reactor, and it shows in the robustness and layout of the facility, so re-establishing it as a nuclear facility should be feasible. The fact that this facility once housed a reactor also offers an advantage in streamlining environmental assessments. Some images of TRA-641 are shown in Figure 11.

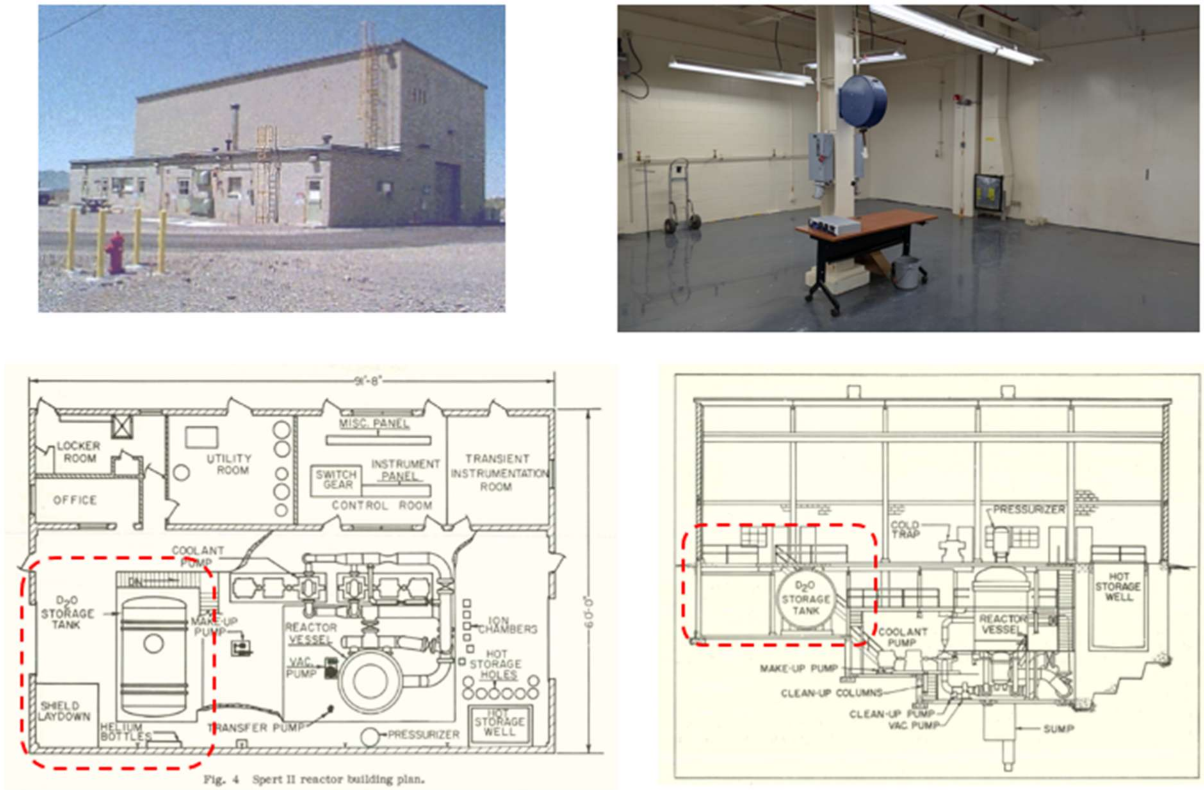


Figure 11. Exterior photo of PBF-612 (top left), modern photo of the heavy water room (top right), top view (bottom left), and side view (bottom right) of SPERT-IV highlighting the heavy water room.

2.4.6. Candidate 6: PBF-613 CITRC Communications Research Facility

PBF-613 was built in 1962 and housed another reactor facility in the SPERT series, SPERT-IV [30]. The SPERT-IV reactor pool tanks and support equipment were removed years ago, but the building remains in serviceable condition and is presently used by NHS personnel for some training exercises. The building has a functioning 12-ton overhead crane, a bathroom, ample electrical services, an HVAC system, and a fire protection system. The facility has two wings with rooms/areas that could be used for equipment, personnel desks, and material storage. Importantly, the fairly large basement measuring 48 ft × 73 ft combined with the high 22 ft ceiling would create an ideal volume for reducing room return, placing neutron sensors, staging materials, and working with bulky equipment during experiment assembly. Two roll-up doors can be used for equipment deliveries, one of which is reinforced to support 2,500 lbs/ft^2 for heavy equipment and materials. It is presumed that the HST would not be operated unless all personnel were evacuated from the basement. Still, the "open basement" architecture of PBF-613 requires some consideration to reduce radiation dose rates to workers in the building during critical experiment operation. The robust ground floor and support column locations in the basement could support a shielded basement cover, or personnel could evacuate to a control room in a modular building on the adjacent asphalt pad during critical operations.

The ~5 foot thick concrete slab in the basement can support very heavy equipment while ample vertical headroom and hoisting access can enable tall experiments such as those using full height fuel assemblies. The lower-level sump area helps to mitigate the risk of accidental criticality through basement flooding. The basement area is not posted as a contamination area, which greatly facilitates any surface-disturbing modifications such as concrete drilling to anchor the HST. Recently installed stairs facilitate entry to the basement while another set of older stairs provides a second egress path. Like PBF-612, this building was downgraded to a radiologic facility but once housed a landmark research reactor program and re-establishing it as a nuclear facility should be feasible while also streamlining environmental assessments. Building images and potential layouts of a HST in the PBF-613 basement are shown in Figure 12 and Figure 13, respectively.

Figure 12. Exterior Image of PBF-613 (left) and Interior Image into PBF-613 Basement (right).

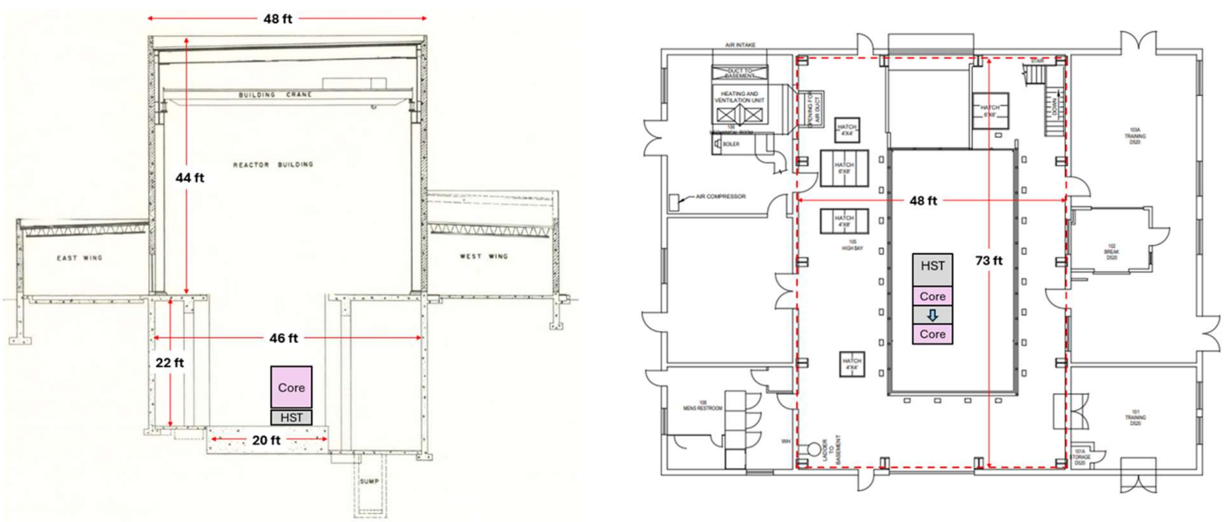


Figure 13. Potential HST placement in PBF-613.

2.5. Facility Recommendation

The evaluations described above show that MFC-723 is probably the least desirable option due to complications with facility evacuation during TREAT transients, inadequate floor capacity, and little else about the facility that makes it attractive. MFC-793C is perhaps only slightly better since it would not need to be evacuated frequently but generally has little else to offer and is rather limited in floor space. The situation improves a bit with TRA-641 where the lack of existing mission and facility layout point toward a somewhat attractive option, but only after investing in an additional control room building, supplemental shielding, and an onerous modification to cover or fill the contaminated canal area. PBF-612 is perhaps slightly better owing to its past life as a reactor facility and some of the attendant facility features, but the facility is well-utilized for other missions, and its heavy water room requires sizable modification to achieve a state barely big enough to install a large HST.

Finally, after a thorough search across INL facilities and based on the requirements described, it is clear that PBF-613 is the preferred candidate. Its sizable basement area provides space for efficient handling and assembly evolutions while being an ideal location for reducing neutron room return. Its robust concrete floor gives ample capacity for large and heavy critical experiments, and its overhead crane greatly facilitates handling tall or weighty equipment and materials. The facility layout needs much less investment than the other candidates in terms of radiation protection while providing innate mitigation for accidental criticality through flooding. The building's ground floor has suitable locations for establishing personnel desk space and storing equipment/materials. All of the other candidate facilities require more onerous retrofits and modifications while providing an inferior capability. While PBF-613 clearly emerged as the recommended option among all the available candidates, some work will be needed to establish this capability. The scope, schedule, and budget estimates for this overall effort are described below.

3. FACILITY CONCEPT IN PBF-613

3.1. Current State of PBF-613

This section provides an overview of the current state of the PBF-613 site. Figure 14 provides an overview of the existing site features. The building is color-coded green. The site boundary is approximately 266 ft wide × 380 ft long. Numerous streetlamps illuminate the site at night. The perimeter around the building is paved for parking and for the offloading of trucks. The paving also reduces the potential for dust to be blown into the building from the surrounding desert. A septic tank and drain field are provided at the southeast corner of the site for the handling of sanitary waste from the building. The site is graded to divert stormwater to an area southeast of the site, as shown in pink in Figure 14.

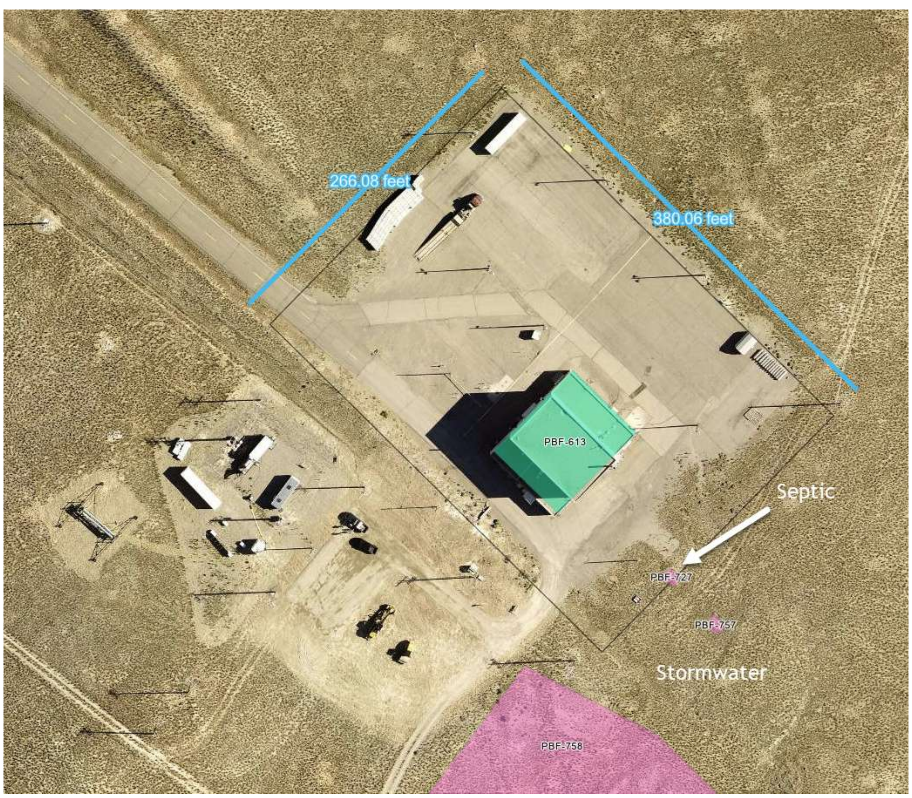


Figure 14. Site features.

The area surrounding the site shows very little sign of settling or shifting. Data from U.S. Geological Survey Wells 1086 and 1087 each located about 500 yards from the building identifies basalt bedrock 3–18 ft below the surface. Figure 15 provides a three-dimensional view of the south end of the substructures. The walled-in area on the left and the right are designated as the west and east wings, respectively. The main entry is a door in the middle of the west wing. There are roll-up doors at the north and south side of the main building. The foundation for each of the wings is basically slab on grade. The floors are a concrete finish with no apparent evidence of cracking or settling. The permissible floor-loading for each wing is 250 lbs/ft^2.

As shown in Figure 15, the main floor provides a large opening into the basement area and smaller openings in the northeast corner for a stairwell and in the southwest corner for a ladder to the basement. The main floor is supported with concrete columns rising from the basement floor. It is rated at 750 lb/ft^2, except for the south end, which is rated at 2,500 lbs/ft^2 for the loading and unloading of heavy equipment. The basement consists of concrete walls that show no obvious signs of cracking or water infiltration. The west side of the basement is a sunken area with a substantial sump on the north end. A near stair unit was recently added to the building to provide a second way to egress from the basement. These stairs are anchored into the basement floor. The central part of the basement is a slightly raised area from the east side. Figures in the SPERT-IV facility description [30] depict that part of the floor as a concrete slab that is rated at 2,500 lbs/ft^2. There are two roll up doors on either side of the reactor hall along with a few man doors and windows in each of the building's wings which will require locks and access controls adequate to ensure personnel do not accidently enter the building during critical operations. Renderings of the HST can in PBF-613 can be seen in Figure 16.

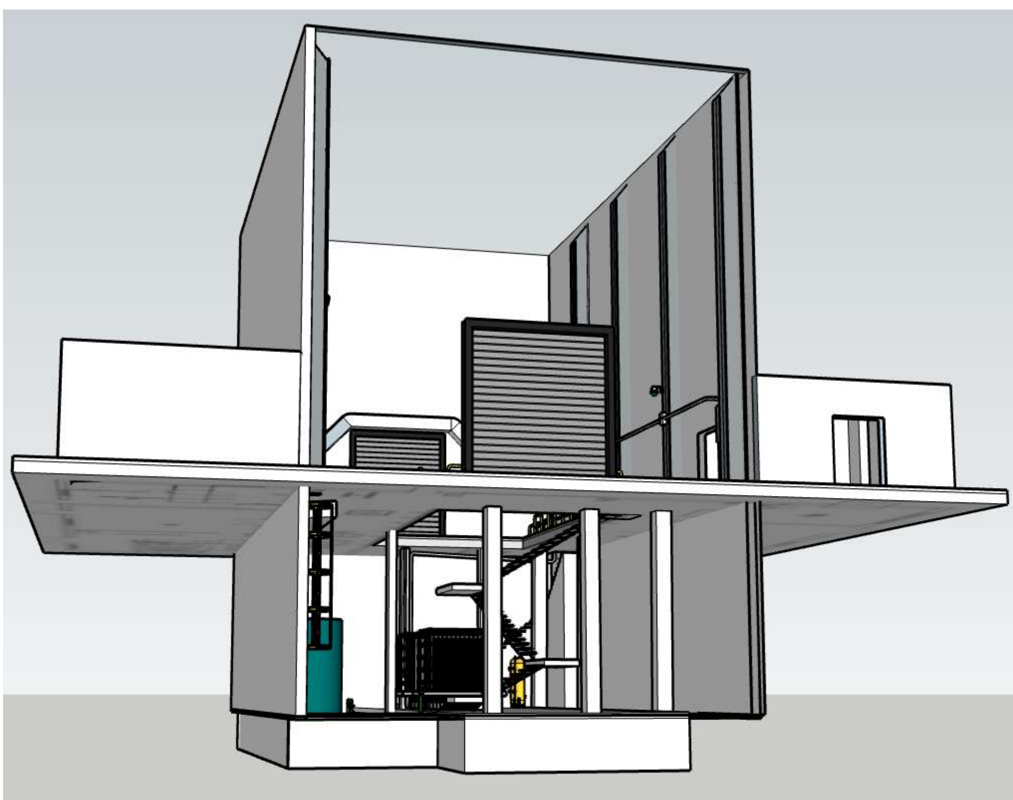


Figure 15. Building substructures.

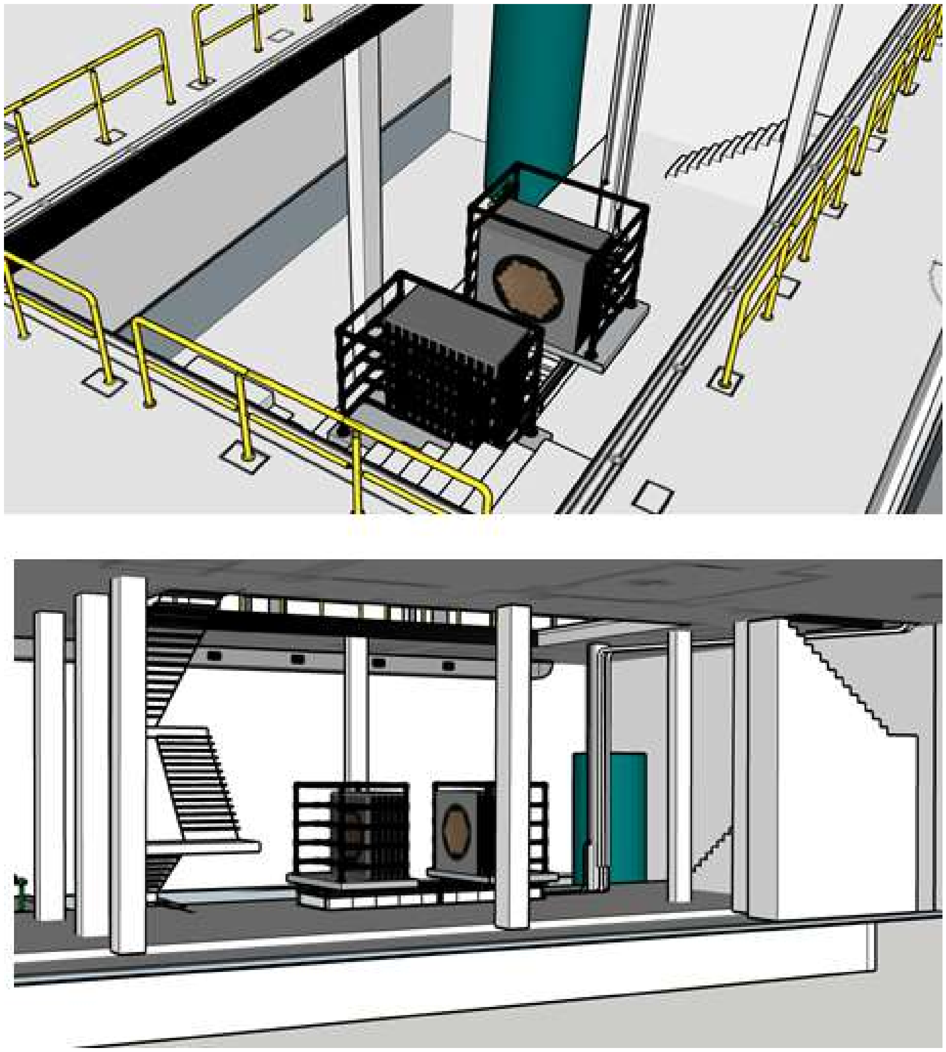


Figure 16. Renderings of the HST with Example Experiment

3.1. Shielding Calculations

The following sections describe some brief calculations that were performed to develop an understanding of radiation dose rates to inform decisions about shielding, control room location, and safety basis strategy.

3.1.1. Dose rates for nominal conditions

A simplified computational model of the PBF-613 building with the HST in the basement was conceptualized to evaluate dose rates inside and outside the building to inform decisions about potential control room locations. When modeling a large building such as PBF-613 in a neutron/photon transport code, many modeling assumptions are necessary, such as building walls and floor thicknesses and their material compositions, so only approximate estimates for the dose rates can be provided. Dimensions used were shown previously in Figure 13, and a radial cutoff of the model is depicted in Figure 17. The roof was assumed to be 0.5 inch thick steel. Walls were assumed to be 1 foot thick concrete using “Concrete [Los Alamos (MCNP) Mix]” defined in the Compendium of Material Composition Data [31]. Ambient air was assumed to be dry at 20°C. Neutron and photon dose rates were estimated using ICRP-116 fluence-to-dose conversion factors following ANSI/ANS-6.1.1-2020 standard [32].

Following the standard recommendations, the most conservative effective dose conversion coefficients corresponding to an antero-posterior exposure geometry were selected. Prompt photons emitted from fission, radiative capture, and inelastic scattering were modeled. Neutron and photon source rates from a critical core configuration were used as input. Dose rates are directly proportional to the source rates which are proportional to the power level. Dose rates reported here are for 10W core power, but it is noted that direct measurement of core power in extremely challenging in a system with low power and no heat removal system, thus significant uncertainties should be considered in predicting limiting dose rates. Photons resulting from potential activation of core and surrounding materials were neglected given the low neutron flux. Neutron and photon ambient dose rates were estimated inside the building for the basement, main floor reactor hall, and side wing location representing the west wing. Average dose rates per location are given in Table 4. For these locations inside the building, statistical uncertainties were low (<3% at one sigma) and not shown. As expected, dose rates were significant near the HST (>1 rem/hour) and preclude having personnel in this area during critical operations. Dose rates were predicted to be much lower in the side wings of the building, but still high enough that these areas would likely be considered radiation areas during critical operations. Thus, a key conclusion is that the reactor control console should be located in a modular building placed near PBF-613.

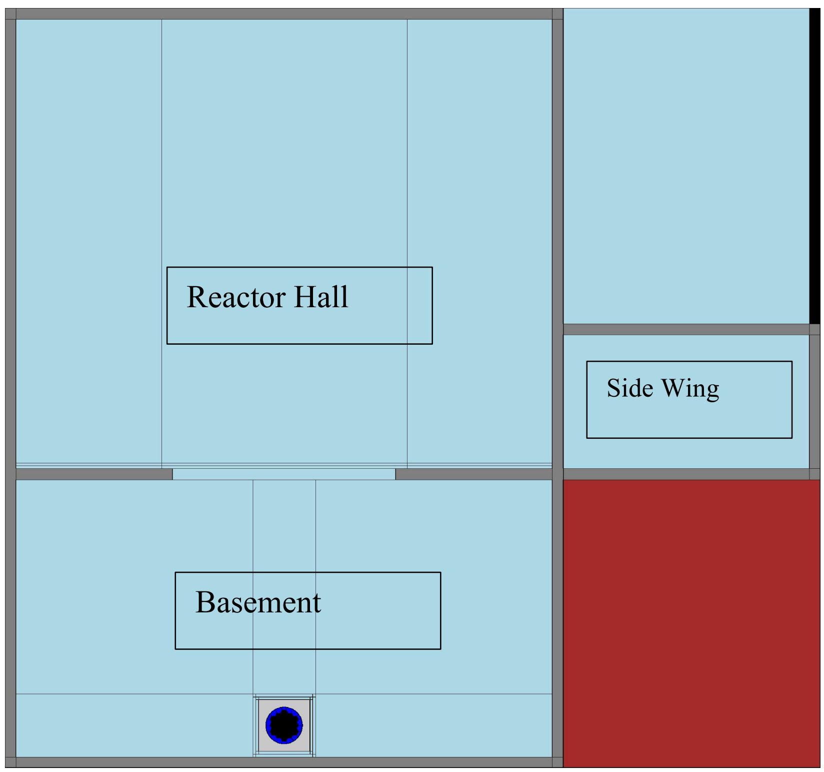


Figure 17. Simplified computational model for shielding analyses (relative scale is preserved)

Table 4. Effective dose rates in mrem/hour, 10 W critical experiment

Side wing	neutron dose	2.29E+00
	photon dose	4.52E+00
	total	**6.81E+00**
Main Floor Reactor Hall	neutron dose	7.45E+01
	photon dose	4.69E+01

	total	**1.21E+02**
Basement	neutron dose	7.47E+02
	photon dose	3.46E+02
	total	**1.09E+03**

Neutron and photon effective dose rates were also estimated for three different locations outside of the building on the paved parking lot. Distances from the core in the basement are shown in Figure 18. Given the significant distance from the core, photons that leak out the building and return through sky shine were found to be the dominant contributor. The photon fluence can be tallied using an analog Monte Carlo simulation, with a reasonable statistical uncertainty (<5% at one sigma). There is however a small neutron fluence at these locations, subject to a much larger statistical uncertainty without biasing. Hence, the Serpent built-in global variance reduction technique based on the Forward-Weighted Consistent Adjoint Driven Importance Sampling method, was employed to produce weight windows using a (Monte Carlo) adjoint calculation for each of the remote locations. These weight windows were then used in a second Monte Carlo calculation to estimate neutron fluence. No shielding was assumed for the remote-control room building. Values account for statistical uncertainties using:

$$min = \mu - 2\sigma$$
$$max = \mu + 2\sigma$$

With μ the mean value and σ the standard deviation.

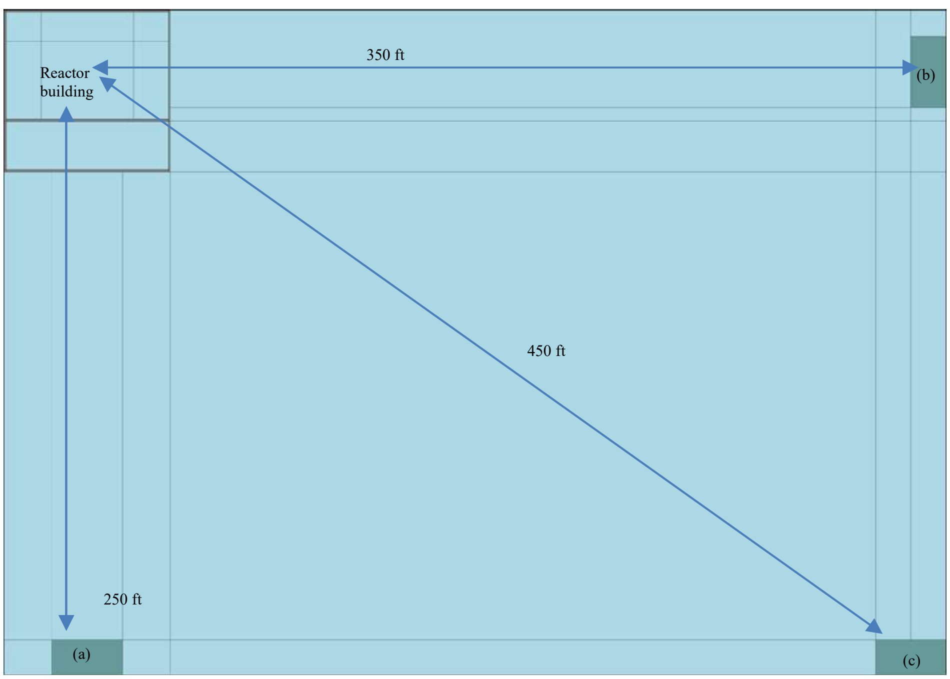


Figure 18. radial cutoff of the geometry along X-Y plane showcasing potential remote control room locations

Results shown in Table 5 show that the total dose is on the order of 0.022 mrem/hour (c location) to 0.074 mrem/hour (a location), three orders of magnitude lower than the dose rate in the building side wing (~7 mrem/hour). The duration of a critical experiment is on the order of a few minutes to a couple hours (i.e., the time required to collect measurement data) so there will be ample margin to the maximum

DOE occupational dose limits (5 rem/year) when accounting for the cumulative exposure to many experiments per year.

Table 5. Effective dose rates in mrem/hour for the three different locations, nominal operation, 10W core

	(a)		(b)		(c)	
	min	max.	min.	max.	min.	max.
neutron dose	2.4E-03	7.2E-03	1.8E-04	1.1E-03	1.3E-04	1.3E-03
photon dose	5.1E-02	6.7E-02	2.4E-02	3.2E-02	1.4E-02	2.1E-02
total	5.3E-02	**7.4E-02**	2.4E-02	**3.3E-02**	1.4E-02	**2.2E-02**

3.1.2. Dose rates for reactivity insertion accident conditions

While dose rates shown in the previous section for locations outside the PBF-613 building are very low, they correspond to nominal conditions when the experiment is critical at a very low power (~10 W). It is also necessary to investigate potential accidental conditions where a power excursion is postulated. Accident scenarios are currently unknown as a detailed safety analysis will be conducted at a later phase of the project. It is however possible to quickly estimate the energy released from power bursts that would occur from reactivity insertion accidents in a critical experiment, using an analytical solution of the point-kinetics equations, such as Fuchs-Nordheim model [33]. The analytical solution is found for reactivity insertions greater than 1$, using constant temperature coefficient of reactivity (α), and inverse heat capacity (K), defined as:

$$\rho = \rho_0 - \alpha T(t)$$
$$\frac{dT}{dt} = KP(t)$$

The Fuchs-Nordheim model predicts the maximum power during a prompt supercritical Reactivity Initiated Accident (RIA) via:

$$P_{max} = \frac{\beta^2(-1)^2}{2\alpha l K}$$

And the normalized integral of the reactor pulse as:

$$I \ = \frac{2l}{\beta(\$ - 1)}$$

With β effective delayed neutron fraction, l the prompt neutron lifetime. The total energy released is $E = P_{max} I$. The total energy released is then converted in neutron and photon dose rates using the estimates given in Section 3.1.1 which are in mrem/hour for a critical experiment operating at 10 W, so the conversion factor from Joules to rem is

$$C = 10^{-3} \frac{1}{10 * 3600} \times E$$

Since SPARC will be a multipurpose facility with different fuel and moderator materials, two distinct cases are tested, one for a thermal spectrum (HTGR experiment) and a fast spectrum configuration (SFR experiment). Kinetic parameters such as prompt neutron lifetime and fuel temperature feedback may significantly differ between critical configurations. Calculations for a reactivity insertion of 1.5$ were performed and shown in Table 6. Due to the lower prompt neutron lifetime and lower doppler coefficient, the maximum power was larger for the fast spectrum case than for the thermal spectrum. The low neutron lifetime however reduces the width of the pulse, leading to a lower integral power in the former case. Note that for a fast spectrum configuration with HALEU, β_{eff} slightly increases relative to a thermal spectrum due to contributions from ^{238}U fissions. Overall, the energy released during the transient is on the order of 10 to 100 MJ for a thermal and fast spectrum experiment, respectively. Note that this estimate is highly conservative as it does not credit the main safety mechanism (i.e., table separation) which will reduce the magnitude of the power transient. Also, the sequence of events leading to such an RIA are not identified at this moment, and if deemed possible, will be precluded by administrative controls and design features. Using the maximum energy released for the fast spectrum case, effective doses in rem are computed and listed in Table 7. While the doses in the control room inside of the building are near the regulatory limits of 25 rem, they are two orders of magnitude lower outside in the remote-control room locations, which confirm that the radiological risk is much lower for these locations in the event of a critical accident.

Table 6. Energy released during RIA events

	graphite moderated, TRISO fueled core	fast spectrum using metallic fuel core
prompt neutron lifetime	**1.00E-04**	**5.00E-07**
effective delayed neutron fraction	**7.00E-03**	**7.00E-03**
total reactivity inserted ($)	**1.50E+00**	**1.50E+00**
Inserted reactivity above 1$	**5.00E-01**	**5.00E-01**
alpha = prompt negative temperature coefficient (1/k dk/dt)	**5.00E-05**	**5.00E-06**
fuel volume (m^3)	**1.18E-01**	**8.89E-02**
fuel density (kg/m^3)	**2.25E+03**	**1.57E+04**
fuel specific heat capacity (J/kg/K)	**7.34E+02**	**1.32E+02**
C = total heat capacity of compact in core (J/K)	**1.94E+05**	**1.84E+05**
K = reciprocal of total heat capacity of fuel in core, 1/C (K/J)	**5.15E-06**	**5.43E-06**
Pmax = beta^2 ($-1)^2 / (2 alpha l K) (W)	**2.38E+08**	**4.51E+11**
Normalized integral power I = 2l/(beta*($-1))	**5.71E-02**	**2.86E-04**

integral of reactor pulse power = 2*I*Pmax (J)	**1.36E+07**	**1.29E+08**

Table 7. Effective doses in rem for accidental conditions

	control room in side wing	remote control room (a)	remote control room (b)	remote control room (c)
neutron dose	**8.19E+00**	**2.57E-02**	**3.99E-03**	**4.56E-03**
photon dose	**1.62E+01**	**2.40E-01**	**1.15E-01**	**7.37E-02**
total	**2.44E+01**	**2.66E-01**	**1.19E-01**	**7.83E-02**

3.2. Structural Investigations

A crucial consideration in defining scope of the SPARC project was determining whether PBF-613 required structural modification to resist seismic events and other loadings. The PBF-613 superstructure is constructed with steel framing with masonry infill walls. The main high-bay portion of the building consists of a steel braced frame around the perimeter with two bays of bracing in each wall in both the north-south and east-west directions. Additionally, there are three moment frames in the intermediate bays between the north and south braced frames. The diaphragm consists of metal deck over steel beams with horizontal angle bracing in all of the outer-perimeter bays of roof framing. Reinforced masonry infill walls are integrally constructed with the columns around the perimeter of the building.

The roof gravity loads (dead, live, and snow) are carried by roof framing to the steel columns which are supported by the reinforced concrete substructure at their base. The gravity loads for the masonry walls are self-supported by the walls themselves. The lateral seismic loads from the roof are transferred through the roof diaphragm to the steel braced frames and moment frames. The lateral seismic loads in the masonry walls are assumed to be resisted by the steel frames to which they are integrally attached. The wind loads are transferred through the masonry walls and are resisted by the steel frame and roof diaphragm.

The east and west wings of the building consist of reinforced masonry bearing walls with metal-deck-covered open web steel joist or steel beam framing. The metal deck roof serves as the diaphragm and the masonry walls serve as both the vertical and lateral force resisting system for the wings. Figure 19 shows a plan view of PBF-613 which designates the lateral force resisting systems for the structure.

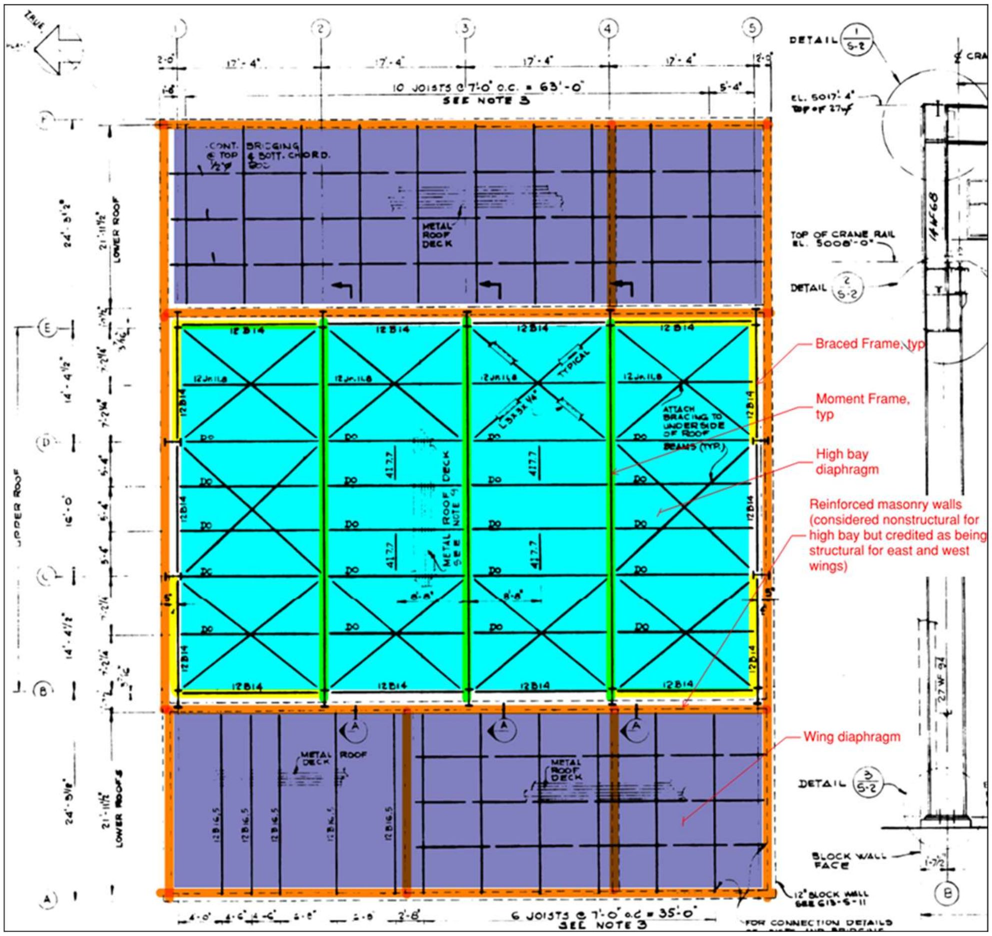


Figure 19. PBF-613 Lateral Force Resisting Systems

In order to host critical experiments, PBF-613 will need to be designated as a Hazard Category 2 nuclear facility. This requires evaluation of the existing structure for Natural Phenomena Design Category 2 Hazards (NDC-2). INL engineering standards for structural design refer to DOE-STD-1020-2016 for determination of Natural Phenomena Hazards (NPH) and Design Criteria [34]. For NDC-2 structures, DOE-STD-1020-2016, defaults to the latest version of the International Building Code (IBC) which references ASCE 7 for determination of seismic, wind, and snow loading [35]. While the specific code years referenced in DOE-STD-1020-2016 are IBC-2015 and ASCE 7-10, if a more recent version of DOE-STD-1020 were to be released it would reference IBC-2024 and ASCE 7-22.

If PBF-613 were a new structure that were to be built today it would be appropriate to evaluate it to the seismic design requirements of ASCE 7-22. The foundation for PBF-613 is placed directly on rock (or lean concrete over rock) [36]. The measured shear wave velocity for rock typical to the INL site is 3718 ft/s [37] which corresponds to Soil Site Class B in ASCE 7 [38]. For Soil Site Class B, the Short-Period Response Acceleration Parameter (SDS) and 1 second Period Response Acceleration Parameter (SD1) are 0.13 and 0.05, respectively. These values would place PBF-613 into Seismic Design Cateogry A. Seismic Design Category A structures do not require any explicit seismic analysis and structures only need to comply with General Structural Integrity Requirements. These requirements, stipulated in Section 1.4.1-1.4.4 of ASCE 7, can be shown to be met by inspection or shown to be bound by the lateral (wind) loads to which the structure was originally designed.

Because PBF-613 is an existing structure, a different approach for seismic evaluation may be more appropriate. Section 2.1.1 of DOE-STD-1020-2016 requires evaluation of existing facilities to be done according to NIST GCR-11-917-12. According to this standard and given the low SDS and SD1 of PBF-613, the building is exempt from the evaluation standards referenced therein [39].

PBF-613 was originally designed for a wind load of 30 psf below 30 ft and 35 psf above 30 ft [40]. These loads are greater than the Main Wind Force Resisting System loads calculated to ASCE 7-22 for a 115 mph wind speed. Furthermore, the wind speeds calculated to ASCE 7-22 are to be used in conjunction with Load and Resistance Factor Design (LRFD) methodology of the more recent material-specific codes, which give a higher capacity for members and connections than the older codes. In short, the building was originally analyzed to higher wind loads and assigned lower capacities than would be done to newer versions of the code. Therefore, with regard to wind loads, the original building design would bound Wind Design Cateogry 2 (WDC-2) loading requirements.

The roof structure of PBF-613 was originally designed for a roof live or snow load of 30 psf [40]. The minimum low slope roof snow load calculated to ASCE 7-22 is 25 psf. While this load is bound by the original design roof load, the newer codes place a higher load factor (1.6 vs 1.0) on roof snow or live loads. However, capacities calculated to LRFD for the most governing limit states are only reduced by a strength reduction factor of 0.9 as opposed to using a safety factor of 1.65 as was used in the older code to which PBF-613 was originally designed. In order to compare the original design loads to the Precipitation Design Category 2 (PDC-2) loads required by ASCE 7-22, the appropriate load factors and safety factors must be applied. As shown below, even with the application of these factors, the net effect is that the original design roof loads bound those of ASCE 7-22.

$$\frac{1.6}{0.9} \times 25\,psf = 44.44\,psf < 1.65 \times 30\,psf = 49.5\,psf$$

In conclusion, the structure appears to satisfy the latest NDC-2 loading requirements without additional structural or seismic analysis of the building. The evaluation for seismic loads is exempted based on very low level of seismicity, and wind and snow loads are bound by the original design loading for the building. No retrofits appear to be required for PBF-613. An on-site investigation and condition assessment is recommended to inspect the condition of the applicable structural components identified in Table 4-1 of ASCE 41-23 [41].

3.3. Reactor Control Room and Trip System

Establishing a reactor control console and trip system will be one of the most significant efforts in the scope of this project. As discussed previously, dose rate estimates showed that a new modular building will be needed to house the Human-Machine Interface (HMI) for the reactor control system and attendant personnel during critical operations. A prefabricated modular building will be placed on the grounds near the PBF-613 building. This new building will be large enough to house the control console equipment and attendees during criticality safety classes. This modular building will be situated at a distance far enough from the reactor building so that additional radiation shielding is not required while minimizing the distance/cost of routing utilities and data cables to this modular building. The modular building will be a single room with power and climate control but will not contain plumbing services. Power to this building will be routed from the electrical panels in PBF-613. The modular building will not remain transportable as it will be placed on the site and categorized as real property.

The reactor control and trip systems will utilize two independent systems to ensure safety and functionality. The overall system will include a reactor trip system and a neutron monitoring system which will use stationary mounted detectors in the PBF-613 facility. This system will be designed to trip at adjustable flux levels which can vary depending on the experiment. The control system will consist of a programmable logic controller and a HMI system based on Rockwell Automation which is a common system used across INL. A separate data acquisition system will be used for experiment data collection, allowing for a flexible number of inputs from various detectors, including thermocouples, in-core fission chambers, self-powered neutron detectors, fiber optic scintillators, and external traditional ^{10}B ion and fission chambers. The system will also be designed to accommodate thermocouple instrumentation and fiber optic temperature measurements throughout the core. Additional data will be collected using traditional reactor dosimetry through wire and foil activation leveraging INL's extensive experience with reactor dosimetry in existing reactors such as ATR and TREAT measured in two dedicated counting labs at the site.

Conduit will be installed between the basement and upper wing for control signal routing, with additional conduit left open for potential future installation of fiber optic cables for instrumentation. Digital or sensitive electronics for the Instrumentation and Control (I&C) system will be placed upstairs in a separate I&C room in the support wing, away from the main hall, to reduce radiation exposure and protect against damage with additional shielding. Sufficient spare conduit and coaxial signal cables will be installed to allow for future instrumentation and easy replacement of cabling for future facilities. In the upper wing, a minimal control system will be installed to allow for basic operation and testing of the I&C and HST, while the external control room will be designed for full system operation. All safety-related components of the system will be placed in the main building and designed to be fail-safe in the event of a loss of connection with the external control room.

The HMI design will follow DOE requirements for human performance design and standard INL practices for display layout, alarms, and functionality. The interface between the control room and the I&C room will primarily use fiber optic or network connections with a limited number of analog connections through a prefabricated trench. The cabling between the I&C room and the control room will be designed to eliminate or minimize safety signals.

3.4. Safety Basis Strategy

As a critical experiment facility, the safety strategy will be consistent with 10 CFR 830 Subpart B [42]. Regardless of siting, the experiment will follow the requirements of DOE-STD-1189-2016 [44] for integration of safety-in-design and will produce safety basis documentation accordingly. The safety basis will follow DOE-STD-1237-2021 [45] Section 3.3, “Critical Assemblies,” as the safe harbor methodology per 10 CFR 830 Subpart B [42]. SPARC will be categorized as a Hazard Category 2 Nuclear Facility per DOE-STD-1027-2018 [43]. Critical experiments will be operated at very low power and the fission product accumulation will remain low; therefore, the reactor will be categorized as an NDC-2 facility per DOE-STD-1020-2016 [35]. Per Section 3.3 of DOE-STD-1237-2021 [45], SPARC will use DOE O 420.1C Ch. 1 augmented with ANSI/ANS-1-2000 [32] for the specification of the reactor specific design criteria. The safety basis will follow DOE-STD-3009-2014 [46] for format and content. The methodologies outlined in DOE-STD-3009-2014 [46] will be employed for hazard identification and evaluation; design basis accident identification; accident consequence analysis; hazard control selection; structures, systems, and components classification hierarchy; and a defense-in-depth evaluation.

4. PLANNING ESTIMATES

4.1. Minimum Viable Product Strategy

It is important to outline the minimum criteria for success by highlighting the features needed to conduct useful critical experiments in SPARC. It is also important to identify features, options, and enhancements that would increase the facility’s utilization and impact. Naturally, these enhancements can be included in the initial package if resources are available to support, but these upgrades can also be deployed later depending on need and available resources. The strategy for the proposed project plan is to prioritize resources toward the initial minimum viable product while retaining awareness for enhancements if opportunities arise, especially if such enhancements are highly desired and can be accomplished more efficiently during the initial facility modifications. Table 8 below provides a breakdown between the minimum needs and enhanced capabilities alongside some discussion and rationale. The project estimates described later represent the minimum viable product needs. Further work is needed to estimate resources required for enhanced capability features.

Table 8. Comparison of minimum viable product and enhanced capability features

Minimum Viable Product	Enhanced Capability
A facility that meets the security requirements as a property protection area for storage of HALEU materials (enrichment of less than 20%).	A facility that meets the security requirements as a limited area for storing significant quantities of beryllium. Note that a facility able to handle greater than 20% enrichment and/or TRU bearing fuels is beyond scope in PBF-613. The present strategy is to defer such capabilities into a potential Phase 2 deployment in the former ZPPR building when it becomes available in the further future.
An approved documented safety analysis to operate critical experiments in a Hazard Category 2 nuclear facility with appropriate processes to authorize new experiment designs.	n/a
A HST which is able to do the following: • Be installed into its intended location without requiring major modification to the existing facility overhead handling capabilities. • Support at least 2.0-meter cube and 24,000 kg of core material (12,000 kg on each side) with versatility in mechanical fixturing. • Drive two core halves together with accuracy and repeatability of 0.01 mm in horizontal gap and 2 arcmin in angular alignment. • Reliably provide a credited safety function to separate the core halves in accordance with the safety basis.	n/a
Postings, barriers, and/or locks as needed to prevent personnel access to the assembly area during critical operations.	n/a
Audio systems to indicate neutron multiplication in the assembly area. Audio and video systems for monitoring and communication between the control console and assembly area.	n/a
Redundant neutron detectors and signal processing/logging equipment in the assembly area and control room suitable to make the approach to critical, log operational data, and to send reactor trip signals.	n/a
Appropriate monitoring systems and alarms for monitoring operations and subsequent radiation in the assembly area.	n/a

Minimum Viable Product	Enhanced Capability
A reactor control console that provides the HMI for operation and records important experimental data including neutron counts and HST separation distance.	n/a
The ability to place flux dosimetry into and around core materials during irradiation followed by shipping to other INL facilities for gamma spectroscopy.	On-site gamma spectroscopy capabilities for faster and more efficient counting of flux dosimeters.
Reactor trip system, which responds to automatic and manual trip signals in accordance with the safety basis and which has hardware/software inputs to accept trip signals from experiment specific systems (e.g., loss of heater power).	A multipurpose experiment data acquisition and control system ready to enable capabilities such as electrical heating of core materials, reactivity oscillating devices, and remote controllable reactivity shim devices.
The ability to conduct experiments with uranium-bearing materials, which are clad or encapsulated.	The ability to conduct experiments on bare uranium materials (negative pressure ventilation/enclosure from the assembly area through filter and stack exhaust).
Ability to perform critical operations without requiring areas that must be habitable during operation (e.g., control room) to be considered radiation areas.	The abilities to operate at power levels which would require a heat removal system or to perform pulsed operations.
A location to stage, store, and unload enough fuel and core materials without requiring partial batches of material to be moved in and out of the facility while constructing one critical experiment.	Location(s) and storage racks in or near the building to store several fuel and core material types without requiring shipment to and from other sites at INL.
An inventory of nuclear fuel adequate to construct at least one basic critical assembly for the inaugural experiment (assumed plan is to use existing 4.81% enriched UO_2 in SST-clad fuel pins).	A library of key fuel systems stored at INL, including 19.75% enriched metallic fuel, ~10% enriched pelletized ceramic fuel, 19.75% enriched coated particle composite fuel, and 19.75% enriched U-Zr-H fuel.
The ability to use the existing facility crane (for heavy items) and stairs (for hand-carriable items) to transport core materials into the assembly area.	Handling efficiency improvements such as a vertical conveyor to move drums between building levels, forklift, and/or rolling gantry crane in the basement.
A building that meets the minimum requirements for electrical service and environment control for the reactor system, safety, and comfort of human occupants.	Convenience/cosmetic improvements to the facility, such as improved lighting, new paint/coverings, exterior signs, break/office areas, and removal of legacy asbestos materials, so that future emergent maintenance items, which may affect those areas, can be performed more efficiently.

4.2. Project Scope

This project will provide management, analysis, safety basis, design, fabrication, construction, and readiness activities necessary to perform the scope as currently identified in Table 9 below.

Table 9. Scope Summary

Activity	Description
Management and Integration	This activity includes the necessary tasks to support the overall project's planning, execution, monitoring, reporting, stakeholder interface, and closeout. This activity consists of project and engineering management resources.
Horizontal Split Table	This activity includes the development of a functional and operational requirements (F&OR) document, performance specification, procurement documentation, award of a design/build subcontract, detailed design, fabrication, delivery, and installation.
PBF-613 Modifications	This activity includes the development of an F&OR document, detailed final design (e.g., construction specifications, drawings, vendor data schedule, etc.), procurement documentation, award of a construction subcontract, and the resources necessary for execution of the construction subcontract. It also includes the Modular Control Room located outside the building and the necessary I&C equipment for the interior Reactor Instrumentation Room.
NEPA Process	This activity assumes the project will be subjected to an environmental assessment, as has been the path for similar projects at INL. This would encompass the initial environmental review process, DOE submittal, public comment process (if needed), etc.
Safety Basis	This activity includes the activities and documentation consistent with 10 CFR 830, DOE-STD-1189, and MCP-18121, and generally includes the Safety Design Strategy, Conceptual Safety Design Report, Preliminary Documented Safety Analysis, and Documented Safety Analysis with the supporting analyses of hazard and accident analysis, dose consequence analysis, facility hazard categorization, etc., with one of the outcomes being a Hazard category-2 nuclear facility. This activity also includes deliverables related to facility safety consistent with DOE O 420.1C to document the nuclear safety design criteria, fire protection, nuclear criticality safety, natural phenomena hazards, design criteria for safety structures, systems, and components, etc.
Experiment Design	This activity includes the necessary steps to derive experiment requirements, develop processes, document processes, design an initial experiment, review and approve an initial experiment, and conduct an inaugural experiment on the HST. This activity will be used to train nuclear facility operators and engineering staff, develop operating procedures, and generally ensure documentation and tasks are completed, which are needed for successful approval of the ORR.

Operational Readiness	This activity includes the activities and tasks necessary to conduct an ORR so that critical experiments can be safely and successfully operated. This activity includes a review of safety documentation, environmental documentation, training records, compliance with DOE Orders, and ensures procedures, personnel, equipment, and systems support the approved requirements for nuclear facility operations.
Turnover/Closeout	This activity includes the activities necessary to turn the project over to the nuclear facility operations management team. This activity also includes the tasks necessary to closeout the project, such as closing of charge numbers.

Preliminary cost and schedule estimates have been developed based on the minimum viable product attributes and project scope described above. These estimates were derived from historical data of similar projects conducted at INL, initial vendor quotes, current market conditions, and subject matter expertise. The cost range to execute the project scope is estimated at \$30-\$45M with a schedule range of 3-5 years. Based on the information available and planning work performed to date the "realistically optimistic" point estimate for this scope is \$34M and with just over three years of time to perform. As designs mature the cost and schedule estimates will be updated accordingly, and consistent with DOE O 413.3 guidelines, a performance measurement baseline will be developed.

4.3. DOE O 413.3B Tailoring Strategy

The requirements identified in DOE Order 413.3B are mandatory for capital asset projects having a total project cost greater than a threshold value. This value was \$50M when the present work began and was later revised to \$300M by the U.S. Secretary of Energy prior to the closure of this report. This project has ample margin to the threshold so a tailored approach will be used in lieu of the full 413.3B-mandated process. This project will use this tailored approach and will take into consideration the size, complexity, costs, schedule, and associated risks to determine the best path for efficient project completion.

4.4. Assumptions and Risks

This project will follow MCP-7350, "Project Risk Management," which provides proven, effective processes to assess and manage risk. Risk management is a necessary and ongoing management process. This project will continuously evaluate risks, they will be retired or realized, and actions taken as necessary. Rather than develop a standalone risk management plan, the risk management processes for SPARC will be documented in the project execution plan. Table 10 lists the high-level project risks currently identified in the early stages of this project that will be updated each fiscal year or when significant changes are made to the project.

Table 10. Preliminary identified risks

Risk #	Risk Title	Risk Description	Probability Level	Impact Level	Priority Level	Mitigation Notes
1	Seismic Upgrades	Structural upgrades to PBF-613 are required due to seismic analysis results	1	2	2	Initial results of the seismic analysis indicate no upgrades will be necessary, and if they are necessary, these modifications would be relatively minor.

2	NEPA Process	An EIS is required instead of an EA	1	4	4	Based on recent projects at INL of similar scope, an EA has been performed
3	Safety Analysis	External reviews/approvals of safety basis documentation are not completed as planned	3	3	9	Communicate with external reviewers frequently to promote efficiency
4	HST Weight	The HST cannot be designed and fabricated at a weight within the existing overhead crane capacity	2	5	10	Ensure this maximum weight is identified in procurement documentation as early as possible with the HST Subcontractor
5	ORR	ORR external reviews/approvals take longer than planned	2	2	4	Ensure an experienced ORR Lead is assigned to assist with these reviews
6	Labor Resources	The labor required to complete the work scope far exceeds what was planned	3	4	12	Develop task baseline agreements with respective resource/management, monitor labor charges bi-weekly
7	Funding	Adequate funding cannot be obtained, and activities either need stopped or cannot be started.	2	8	16	Work with DOE and stakeholders to identify potential funding sources and/or obtain additional federal funding
8	Competing Priorities	In anticipation of increased funding for other reactor type projects at INL, pursuant to recent executive orders, specialized labor resources are unavailable.	4	4	16	Because of the recent executive orders, it's imperative that SPARC receive the necessary funding to gain commitments from those organizations providing the specialized labor resources, particularly nuclear safety analysts.
9	I&C Equipment	Unanticipated I&C equipment in support of the Reactor Instrumentation Room are identified during detailed design.	3	3	9	Identify any unknown I&C equipment as early in the design process as possible and reflect in the project baseline.

5. NUCLEAR FUEL OPTIONS

Approximately 600 fuel rods, which were once produced for but never used in the SPERT program, were furnished to the Rensselaer Polytechnic Institute and were used there for decades in the Walthousen reactor critical facility [47]. These rods consist of 4.81% enriched UO_2 pellets in stainless steel cladding to form 1.2m long fuel rods. The Walthousen facility is now being decommissioned and its fuel rods, which are still contact handleable at a mere 0.8 mrem/hr on contact, are planned to be returned to INL under DOE's University Fuel Services (UFS) program. These rods have a rich history and are well characterized in the criticality experiment community and will make a key asset in the inaugural SPARC experiments, in supporting criticality safety training courses, and in supporting numerous general purpose criticality experiments.

However, SPARC will maximize its long-term impact by obtaining a dedicated inventory of key HALEU fuel systems. At present nuclear fuels are not readily available in any form except light water reactor fuel with ~5% enriched UO_2 pellets in ~3.5 m long zirconium alloy tubes. Commercially available HALEU fuels are a nascent development but are expected to become more available as this project progresses. It will be important for the project to stay in sync with the advanced nuclear fuels community to develop plans to populate its library with HALEU options for metallic fuel, TRISO fuel, and high-density ceramic fuels. Some possible near-term options for procuring or manufacturing fuel systems of interest are listed below:

- Encapsule HALEU rectangular ingots of the U-Mo alloy that were previously produced for the U.S. High Power Research Reactor program and/or produce new ingots using the same capabilities which were established for this program.
- Recast electro-refined HALEU metal at MFC from reguli geometries into more amenable shapes for critical configurations (e.g., rods or plates) or acquisition of fresh HALEU metallic fuel pins as fuels begin to be produced for commercial SFRs.
- Procurement of TRISO compacts and/or pebbles from commercial TRISO suppliers as production is expected to increase to support several advanced reactor projects.
- Procure LWR fuel at the highest enrichment available (up to 10% ^{235}U) in the form of UO_2 pellets as commercial light water reactor fuel vendors increase production of this product toward increasing fuel burnup.

Another near-term opportunity to obtain fuel is to partner with reactor developers seeking to demonstrate their reactor design and borrow from their early batches of fuel to conduct a critical experiment prior to its being loaded into an end use core. This approach is viable, especially when the end use reactor is to be tested on INL property, because critical experiments can return fuel in like-new condition with negligible radioisotope inventories. This approach would require careful alignment of schedules to be workable and would not result in a standard library of fuel being retained for future experiments. Examples of this strategy may include the following:

- Use of TRISO compacts or pebbles slated for installation into reactor demonstration projects to be conducted at INL or elsewhere, many of which are being coordinated by NRIC.
- Use of UZrH elements procured by UFS who retains a stock of this fuel type at INL so that university research reactors can obtain replacement elements when needed.
- Use of U-10Zr fuel pins to be made at MFC prior to their use in the Oklo Aurora demonstration reactor.
- Use of fuel molten salts, which could be synthesized at MFC using the capabilities being stood up for MCRE.

As illustrated in this section, there are several envisioned opportunities to obtain suitable fuel. Since obtaining appropriate nuclear fuels is central to the longevity of SPARC's mission, engagement from DOE's Advanced Fuels Campaign will need to play a crucial part in coordinating the design and acquisition of nuclear fuels.

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