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Options for Subscale Maturation of Advanced Reactor Technologies Testing for Nuclear Thermal Propulsion

January 2022

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EXECUTIVE SUMMARY

Several options could be implemented to establish an irradiation testing capability suitable for investigation of the performance of multiple nuclear thermal propulsion (NTP) fuel elements at prototypic conditions. The prototypic conditions of interest are based on the current needs of the National Aeronautics and Space Administration's Space Nuclear Power (SNP) Program. The results of such testing are also intended to reduce the risks currently seen for any future subscale or full-scale ground testing of an engine-reactor system. The optimal solution is dependent upon several factors such as performance, cost, availability, schedule, technology readiness level (TRL), and plans for future testing in the SNP Program.

Three options are considered in this report. An exact cost comparison of these options is not detailed in the report. At best, the relative cost of options can be considered balanced against the ability of a facility to obtain performance data at prototypic temperatures and conditions, in conjunction with other goals of the SNP program, including the desired launch schedule and the advancement of TRLs.

The option that would allow the most-rapid deployment, with the least reduction in risk, is the use of existing facilities and development of multi-physics codes only. Of the options considered here, existing government and university reactors, potentially with limited upgrades, could support separate effects testing at less than prototypic conditions (lower than desired temperatures, pressures, and/or power densities). There would be no or only very limited data to support core neutronics models, reactor controls, or manufacturability. This is also the option most likely to be able to meet the current schedule.

A more robust option that would allow for more extensive testing, greater risk reduction, but added cost, is construction of a new 14-MW TRIGA research reactor. This reactor would be based on an existing Nuclear Regulatory Commission (NRC) licensed design and certified, available fuel with some modifications. The available flux and power density would be less than desired, but limited data could be collected to support some scaling studies and some predictions of startup and transient behavior as well as providing benchmarks for neutronic and multi-physics codes. Such a facility could also be used to demonstrate an exhaust treatment system needed for subsequent ground testing. The facility could be licensed as a non-power research reactor by the Department of Energy (DOE) or as a university research reactor by the NRC.

The most comprehensive solution considered here would be the construction of a Subscale Maturation of Advanced Reactor Technologies (SMART) facility, based on existing fuel elements from the Advanced Test Reactor (ATR). This configuration would provide the closest integrated test capability to meet NTP design parameters of power densities, neutron flux, chamber pressures, temperature gradients, and hydrogen flow rates. This would allow the testing of full-length fuel and moderator elements, including prototypic startup and shutdown testing of reactor components, as well as safety margin testing and exhaust treatment demonstration. This is also the most expensive option, although the exact cost is not determined here. The existing ATR fuel is highly enriched uranium and would be licensed as a non-power research reactor by DOE.

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ACRONYMS

ACRR	Annular Core Research Reactor
AEC	Atomic Energy Commission
ANL	Argonne National Laboratory
ATR	Advanced Test Reactor
B&W	Babcock and Wilcox
BNL	Brookhaven National Laboratory
CEA	Commissariat à l’énergie atomique et aux énergies alternatives (French Alternative Energies and Atomic Energy Commission)
CERCER	ceramic-ceramic matrix fuel
CERMET	ceramic-metal matrix fuel
CF	cold flow
CX	Critical eXperiment
DOE	Department of Energy
DRACO	Demonstration Rocket for Agile Cislunar Operations
DUFF	demonstration using flattop fission
ENDF	Evaluated Nuclear Data File
GH_2	gaseous hydrogen
GTA	ground-test article
HALEU	high-assay low-enriched uranium
HEU	highly enriched uranium
HFIR	High Flux Isotope Reactor
INL	Idaho National Laboratory
INR	Institute for Nuclear Research
IRSN	Institut de Radioprotection et de Surete Nucleaire
Isp	specific impulse
KPP	key performance parameters
LEU	low enriched uranium
LWR	light-water reactor
MARVEL	Microreactor Applications Research Validation and EvaLuation
MCNP	Monte Carlo N-Particle Transport code
MITR	Massachusetts Institute of Technology Reactor
MURR	University of Missouri Research Reactor
NASA	National Aeronautics and Space Administration

NCERC	National Critical Experiments Research Center
NERVA	Nuclear Engine for Rocket Vehicle Application
NET	Nuclear Element Test
NETL	Nuclear Engineering Teaching Laboratory
NNSS	Nevada National Security Site
NR	nuclear rocket
NRC	Nuclear Regulatory Commission
NRTS	National Reactor Test Station
NTP	nuclear thermal propulsion
NTREES	Nuclear Thermal Rocket Engine Environmental Simulator
NTRS	NASA Technical Reports Server
OD	outer diameter
OSTR	Oregon State Training Reactor
PBR	particle bed reactor
PIE	post-irradiation examination
PIPE	Pulse Irradiation of a Particle Bed Fuel Element
PIPET	Particle Bed Reactor Integral Performance Tester
PRIME	Prototypic reactor irradiation for multicomponent evaluation
PSBR	Pennsylvania State Breazeale Reactor
RFP	request for proposals
RIFT	Reactor In-Flight Test
SERC	Science and Engineering Research Center
SMART	Subscale Maturation of Advanced Reactor Technologies
SNL	Sandia National Laboratory
SNP	space nuclear propulsion
SNRE	small nuclear rocket engine
SNPO	Space Nuclear Propulsion Office
SNTP	Space Nuclear Thermal Propulsion
TRD	testing reference design
TREAT	Transient Reactor Test Facility
TRIGA	Training, Research, Isotopes General Atomics
UNWR	University of Wisconsin Nuclear Reactor
VTR	Versatile Test Reactor

Options for Subscale Maturation of Advanced Reactor Technologies Testing for Nuclear Thermal Propulsion

1. INTRODUCTION

The National Aeronautics and Space Administration (NASA) Space Nuclear Propulsion (SNP) Program is actively pursuing technologies that would enable crewed missions to Mars. Nuclear thermal propulsion (NTP) is one of the options for space travel that could support such a mission, and NASA is actively supporting efforts to mature technologies needed for the reactors to drive an NTP device. Much of these research efforts require testing in facilities that can support high temperatures and hydrogen environments, including testing in a reactor facility for simultaneous exposure to irradiation, prototypic power densities, and flowing hydrogen. No facilities that are currently available can support irradiation testing to the mission operating times of either multiple or single NTP fuel elements or some assembly of fuel, insulator, and moderator materials under prototypic or near-prototypic conditions. This concept study gives a preliminary analysis of options that could be implemented to establish an irradiation testing capability that would be suitable for investigation of multiple NTP fuel element performance characteristics at prototypic conditions. The irradiation capability is referred to as the Subscale Maturation of Advanced Reactor Technologies (SMART) in the remainder of this report.

To develop and characterize promising NTP designs, testing of subscale nuclear and non-nuclear systems and components under prototypic conditions will be needed to demonstrate performance and design margin and to develop data to benchmark/validate the modeling and simulation (M&S) tools used in design and safety analyses. Since the NTP reactor serves to heat a hydrogen propellant to high temperature to enable high thrust and specific impulse (Isp), NTP system performance is fundamentally determined by fuel performance. NTP fuel is required to operate at significantly higher operating temperature and power density than exhibited in traditional terrestrial reactor designs. Performance of an NTP engine depends on the ability to demonstrate that the fuel can reliably operate under extremely high temperatures, high power densities, and a flowing hydrogen environment, for multiple burns (thermal cycles).

Historic programs have tested reactor components under prototypic conditions via ground testing of fuels in individual NTP test reactors or in a specialized nuclear furnace, but these specialized facilities have since been decommissioned, and fuel types have evolved. A SMART facility would bridge the gap between current NTP testing capabilities and future full-scale integrated engine-reactor ground testing. Prototypic testing of full-scale fuel elements and other components would enable a significant decrease of the risks associated with fuel performance and NTP reactor technology development. This facility may have the capability to benefit other use cases of space nuclear propulsion technologies, such as nuclear electric propulsion. The Department of Defense is also investigating NTP technology development through the Demonstration Rocket for Agile Cislunar Operations (DRACO) Program. This demonstration will likely have lower initial performance requirements than the NASA SNP system, but it is possible that a SMART test capability would also be useful for validating the DRACO design.

2. NUCLEAR THERMAL PROPULSION REACTOR TESTING NEEDS

Testing in a SMART facility is intended to do the following:

- Obtain fuel and moderator performance data at temperatures and conditions prototypic to NTP.
  - SNP key performance parameters (KPPs): an exhaust exit temperature of >2700 K, 10,000–15,000 lbft thrust, 900s Isp, reactor mass <3500 kg
  - SNP Key System Attributes: cumulative burn duration of > 2 hours, and > 2 restarts.

- Obtain additional data needed to benchmark neutronic and multiphysics codes and reduce the risk of a full-scale or subscale integrated engine-reactor ground test not meeting the SNP KPPs for exit temperature and reactor mass.
- Provide data to benchmark/validate M&S tools.
- Obtain fabrication and assembly experience to build the capability to manufacture NTP reactor components to scale (i.e., at full-size, of sufficient quantity and quality).

Table 1 summarizes the desirable parameters of a SMART facility based on the current SNP Program KPPs and key system attributes and identified programmatic facility bounding conditions.

Table 1. Summary of desirable basic facility parameters.

Basic Facility or Test Article Parameter	**Description**
Maximum Power Density/Deposition	Average of 5 MW/L, peak of 10 MW/L Alternate maximum neutron flux requirement: 10E15 n/cm^2 thermal
Facility Operating Conditions	Goals of longer duration irradiations, continuous operation, capable of restarts (quickly enough to avoid delayed restart) Flowing hydrogen test article coolant Time to achieve full power from initial startup conditions: <1 min Time to achieve steady state: <2 min Time from full power to shutdown: <1 min
Facility Operational Duration	Minimum of 2 minutes steady state hold time Up to 240 minutes of total hold time
Ramp Rates	Minimum of 95 K/sec in the test article fuel for both heat-up and cooldown Determined by test design more than facility design
Peak Temperatures Allowed in the Test Article During Testing	Peak coolant (GH_2) exit temperature: 2700 K Peak fuel temperature: 3000 K Peak moderator temperature: 1200 K Determined by test design more than facility design
GH_2 Supply Conditions to the Test Article	Inlet temperature: 250–400 K Max pressure: up to 10–11 MPa Mass flow rate per channel: ~2 g/sec
Effluent Treatment	Total effluent mass flow rates will be governed by reactor design and test article size. Allowable loss of fission products and uranium from the fuel – implies the ability for total exhaust capture
Allowable Test Article Size	50 to 100 cm long, ~5 cm outer diameter (OD) for one fuel element Room for multiple fuel elements
Allowable Test Article Enrichment	Up to high-assay low-enriched uranium (HALEU)

Basic Facility or Test Article Parameter	Description
Facility or Driver Core Enrichment	HALEU and above
Instrumentation	Reactor design and test independent • Test article inlet and exit fluid T, P, mass flow rate per channel. • Facility ramp rate and power trace • Radiography onsite • Capable of additional inputs from test article Test Dependent • Test article instrumentation to measure temperatures, dpa, expansion, pressure, neutron fluence, etc.
Other Features of Interest	• Cost to build and/or use • Post-irradiation examination (PIE) needs • Schedule to build or availability for new work in existing schedule; testing needs to occur in the next 5 years • Licensing needs for the proposed usage (new license, modifications for highly enriched uranium [HEU], HALEU, etc.)

2.1 Nuclear Thermal Propulsion Reactor Technology Risks

Ultimately, a successful engine-reactor ground qualification test campaign will demonstrate complete risk reduction of the engine-reactor and its subcomponents. However, if prior to this test campaign, technology development tasks include testing to demonstrate reactor component performance and to validate reactor subsystem models, significant risks can be reduced before the ground qualification testing.

Additionally, improving technology readiness level through development and testing reduces technical risk in overall system performance and reliability by providing test data for validating computational models, or verifying that the system is capable of meeting requirements. Furthermore, reduced technical risks associated with the manufacture, performance, and functionality of the reactor (and overall integrated system) improves programmatic confidence in projected cost and schedule estimates for system development.

2.1.1 Specific Risks for NTP Reactor Technology Development

In discussions with NASA SNP stakeholders [Van Dyke et al. 2021], some specific gaps were identified in the current knowledge base that lead to risks that should be reduced prior to engine-reactor ground testing. These include:

- Hydrogen reactions and erosion with reactor and fuel materials (e.g., mid-band corrosion, etc.)
- Fuel and reactor component integrity under maximum temperatures
  - Fuel behavior near the melting point
  - Hydride moderator stability
  - Fuel and moderator response under combined thermal cycling/nuclear transients
  - Fuel/moderation assembly integration and interelement effects

- Fuel and reactor component manufacture and assembly
- Heat transfer from fuel to propellant
- Reactor operation and control
  - Startup of the reactor and controllability under varying hydrogen flow rates
  - Instrumentation functionality and survivability
  - Verification of state points and available heat to the engine
  - Validation of reactor physics and predicted temperature coefficients.

In addition, there are specific nuclear data needs that a SMART testing capability should address [Palomares 2021]:

- High-Assay Low-Enriched Uranium (HALEU) fuel enrichment
  - Space Policy Directive-6 directed the use of HALEU fuel (<20 w/o enriched in U235) for space reactors unless technically infeasible
  - Use of HALEU may drive the desire to thermalize the neutron spectrum compared to historic fast and epithermal designs and may require new nuclear benchmarks to validate nuclear physics models.
- Unique operating regime
  - NTP operating temperatures vary from cryogenic to ultrahigh temperature (>2750 K)
  - Cross-section behavior over the entire temperature regime must be understood for all materials used in the moderator, fuel, and core regions.
- High temperature moderators and materials
  - Space reactors may benefit from high temperature moderator candidates (metallic hydrides and beryllium compounds) and refractory metals or ceramics
  - Previous benchmarks and historic testing do exist for reference, but the neutron spectrum may differ. This may also result in the need for resolution of unresolved resonances in the epithermal energy range.
- Unique working fluids
  - Scattering in hydrogen plays a role in reactor control and reactivity.
  - Scattering in hydrogen is a non-negligible contribution to overall reactor reactivity and will impact transient reactor response.
- Uncertainty in data: updated covariance data will allow for more accurate characterization of reactor uncertainty sensitivities.

There are additional factors to consider with respect to the progression from fuel selection to final fuel production qualification to development of a full-scale engine-reactor system. [Werner 2019] That progression includes the following steps:

- Development of normal and transient operational requirements and fuel characteristics that will satisfy operational requirements (e.g., maximum fuel temperature limit, heat conduction properties, etc.).
- Identification of laboratory-scale procedures, processes, and tolerances that produce the required fuel material characteristics followed by tests that confirm the performance of fuels.
- Identification of engineering-scale production procedures, processes, and tolerances that produce the required fuel material characteristics.
- Establishment and qualification of full-scale production fuel line.

- Testing of fuel and moderator segments or full-scale fuel and moderator elements to demonstrate their ability to meet operational requirements. This testing should match or exceed prototypic environmental conditions expected in the engine-reactor system.
- Development of fuel performance models that represent and predict how the fuel performs under normal, transient and accident conditions. This development effort runs in parallel with the fuel fabrication and testing effort

As the candidate fuel fabrication technology matures, the risk levels are reduced through mechanical and thermal testing on small samples and separate effects testing through a combination of non-nuclear and nuclear test facilities, followed by tests of moderator fuel assemblies and sub-scale systems.

2.1.2 Risk Categories for Nuclear Thermal Propulsion Reactor Technology Development

In general, the risks and development needs identified by previous efforts can be largely grouped into one of four risk categories regardless of proposed reactor design. This discussion is based primarily on the results of the NTP Industry Flight Demonstration Study. [AMA 2019, Palomares 2020] These categories are detailed in the following sections. A summary of the risk factors discussed here is also shown in Table 2.

1. Lack of fuel and moderator performance data at prototypic temperatures/conditions

   Current SNP KPPs for the reactor require operating temperatures and power densities above that demonstrated for any historic integrated NTP system or nuclear furnace conditions. Therefore, the historic performance database does not substantiate target design conditions. This results in risk in meeting performance goals of exit temperature and reactor mass. Data on reactor component performance under prototypic conditions—i.e., proposed temperature, environment (H_2), transients (nuclear and thermal), burn durations and number of burns, and power densities and fluence—is desired to validate the proposed design approach.

   This risk can be reduced substantially through testing fuel and moderator elements under prototypic conditions. This type of testing would require exposure of fuel and moderator elements under combined hot hydrogen environment (flow rates, pressures, temperatures) and nuclear environment (power density, fluence) conditions. Testing should be performed to verify full-scale component integrity to capture prototypic stress and temperature gradients expected during operation.

2. Lack of fuel manufacturing experience for NTP designs

   Because NTP is a unique application much different from existing terrestrial reactors, nuclear fuel forms needed to enable NTP designs have not yet been demonstrated at manufacturing scale (size, quantity, quality) needed for current modern NTP reactor concepts. Lack of proven manufacturing methods results in risk related to the manufacturability of the reactor. Designs that lack identified processing methods, infrastructure, or established quality control processes are associated with higher levels of risk. Demonstration of manufacturing methods that can enable the manufacture of fuel elements from the materials, in prototypic geometries, and of the quality desired for an NTP reactor is desired to reduce risk associated with lack of qualified manufacture methods.

   This risk can be fully addressed by demonstrating a pilot manufacture line capable of fabricating full-scale reactor components at the production quantities and with quality control required for test reactors. Manufacture specification for the fuel will require that the fuels fabricated through selected processes are capable of withstanding proposed operating conditions. Therefore, this risk category should be coordinated with Risk Category 1 activities.

3. Reactor operation database under NTP nominal and designed off-nominal conditions

Because NTP is a unique application and current mission planning calls for aggressive performance goals and new constraints (namely the use of HALEU fuel enrichments), reactor designs proposed to enable modern NTP missions have not yet been comprehensively analyzed or validated for their response under required startup, shutdown and cooldown, steady-state operation, or other critical nominal and designed off-nominal conditions. Lack of validated reactor physics models and control approaches results in risk in meeting performance goals and in the demonstration of desirable reactor responses for planned steady-state and transient conditions. Nuclear demonstration of reactor components to validate reactor physics and transient models is desired to increase confidence prior to an integrated NTP engine-reactor demonstration.

Integrated engine-reactor testing is needed to verify that the proposed engine-reactor design is capable of satisfying all requirements and to provide the data necessary to validate thermodynamic cycle model predictions. However, prior to an integrated engine-reactor test, three types of reactor related testing can be performed to reduce technical risk expected due to reactor operations. These tests include critical, cold- or hot-flow, and control system testing. Critical testing can be performed at multiple length scales (such as single fuel assemblies, multiple fuel assemblies, or full reactor cores) to gather nuclear data or benchmark reactor physics models. This testing can be performed at an existing nuclear facility, including existing reactors or critical assemblies, to verify reactor power distributions, reactivity worths, or temperature coefficients. These data are important to predict reactor state points during nominal operation and to design the control system to be able to enable control under all reference operational modes or design basis accidents. Critical testing needs for space reactors have previously been reviewed [Bragg-Sitton et al. 2011]. Cold- or hot-flow testing has been performed in historical development programs to investigate fluid dynamics and structural interactions due to a flowing hydrogen environment. These tests may use a mockup reactor core with unfueled elements as long as the fluid-structural interactions are representative of the reactor. This type of testing could also be performed on the component level, but multiple assemblies could capture more complex behavior, including interelement interactions. Last, control system testing to demonstrate the control approach (with simulated engine-reactor response) and control system hardware functionality reduces associated risk with ensuring control of the engine-reactor throughout the duration of the demonstration and qualification programs.

4. Testing parameters beyond established facilities limits

   New manufacturing and testing capabilities are needed to enable the SNP program to meet project milestones. NTP reactors require operating conditions that are much different from those obtainable in current test reactors. Lack of existing national infrastructure to enable fuel element or reactor component testing under prototypic or near-prototypic NTP conditions is a barrier to minimizing risk in any of the other categories.

A SMART concept can enable risk reduction in all four categories of risk; the amount of risk reduction will vary based on the SMART implementation. A new reactor facility can greatly reduce risk prior to the engine-reactor ground test by allowing for testing of NTP fuel and moderator components under prototypic conditions (combined nuclear and non-nuclear environment) and generation of nuclear data to support M&S under a relevant nuclear environment (prototypic spectrum, power density, flux). Testing of full-scale components under prototypic conditions would result in a Technology Readiness Level (TRL) sufficient to end technology development prior to an engine-reactor system ground test. (See Appendix A for a discussion of TRL definitions and NTP.)

Table 2. Summary of risk factors of primary interest for a SMART program.

1. Lack of Fuel and Moderator Performance Data at Prototypic Temperatures/Conditions
• Performance data is required under prototypic conditions, i.e., proposed temperature, environment (H_2), transients (nuclear and thermal), burn durations, and number of burns, or power densities and fluence. • High-temperature moderators and materials candidates include metallic hydrides, beryllium compounds, and refractory metals or ceramics. There is limited historic data for reference. • This results in an overall system risk in meeting performance goals of exit temperature and reactor mass.
• Key Risks of Interest ▪ Hydrogen reactions and erosion with reactor and fuel materials (mid-band corrosion, etc.) ▪ Fuel and reactor component integrity under maximum temperatures ▪ Fuel behavior near the melting point ▪ Hydride moderator stability.
2. Lack of Fuel Manufacturing Experience for NTP design
• Nuclear fuel forms needed to enable NTP designs have not yet been manufactured on the scale (size, quantity, quality) needed for full-scale NTP reactor concepts • Results in risk to success in manufacturability of the reactor.
• Key Risks of Interest ▪ Fuel and reactor component manufacture and assembly ▪ Integration of fuel and reactor components with non-nuclear engine parts
3. Limited Reactor Operation Database under NTP Nominal / Designed Off-Nominal Conditions
• NTP reactor designs have not yet been comprehensively analyzed or validated for their response under required startup, shutdown and cooldown, steady state operation, or other critical nominal and designed off-nominal conditions. • The use of HALEU may require data related to a thermal neutron spectrum, as opposed to the historic data on fast and epithermal neutron behavior • Results in risk in meeting performance goals due to reliance on unvalidated and unbenchmarked models for design and test plans.

• Key Risks of Interest
  ▪ Fuel and moderator response under combined thermal cycling/nuclear transient
  ▪ Interelement effects
  ▪ Heat transfer from fuel to propellant
  ▪ Reactor operation and control
    1. Startup of the reactor and controllability under varying H_2 flow rates
    2. Instrumentation functionality and survivability
    3. Verify state points and available heat to the engine
    4. Validate reactor physics and predicted temperature coefficients.

4. Testing Parameters Beyond Established Facilities Limits

• Nuclear fuel forms proposed for NTP designs will require operating conditions that are much different than obtainable in current test reactors
• Results in risk to the testing program in the case of lack of identified facilities infrastructure which can simulate prototypic or near-prototypic NTP conditions.
  ▪ SMART study is intended to propose options based on the limitations of existing facilities
  ▪ Risk levels are reduced through testing on small samples and separate effects testing through a combination of non-nuclear and nuclear test facilities, followed by tests of moderator fuel assemblies and sub-scale systems.

3. PREVIOUS PROGRAMS AND HISTORIC FACILITIES

Several historic programs have developed reactors and fuels for NTP. The earliest space reactor programs (NERVA/Rover, Pluto) relied heavily on ground test reactors, but later historic programs transitioned to incremental development approaches, including subscale and full-scale separate effects testing (ANL, GE-710), in-pile testing of materials under combined non-nuclear and nuclear environments through static (non-flowing) or dynamic (flowing) capsule testing (ANL, GE-710, and SNTP), and nuclear furnace testing of multiple fuel element bundles under flowing hydrogen conditions (NERVA/Rover and SNTP). All programs sought the demonstration of reactor fuel materials under prototypic operating conditions as a prerequisite to transitioning to engine-reactor qualification. Any future efforts should be informed by the valuable experience of the work already done in this arena. A summary of past NTP related programs is presented in this section while each program is described in more detail in Appendix B.

Table 3. Summary of past NTP-related programs.

Program Name	Program Duration	System Description and Target Key Performance Parameters	Program Cost (2021 USD)	Major Achievements	Readiness at Cancellation
Rover/NERVA	1955–1973	**HEU** (U, Zr)C, Graphite Moderator (CERCER) **Target:** 200,000/75,000 lbf, 2361 K chamber temperature, 3.103 MPa chamber pressure, 825 s Specific Impulse	**$8.82B** (assumes total budget spent during 1973, $11B if median year, 1967, used instead)	200,000 lbf (4100 MW_{th}, Pewee) 55,000 lbf (XE-Prime) 6540 s at full power (NF-1) 2750 K chamber temperature (Pewee) 3.8 MPa chamber pressure (XE-Prime) 31.8 kg/s flow rate (Kiwi-B4E) 848 s specific impulse (Pewee)	Reactor readiness: TRL 6 (multiple reactor ground tests, near prototypic) Fuel readiness: TRL 6 (extensive fuel ground tests, near prototypic) Key remaining challenges: demonstrating operation and corrosion resistance with multiple restarts at full power, pressure and flow rate
Pluto Program	1957–1964	HEU UO_2 homogenously mixed with BeO Target: 485 MW_{th}, 35,890 lbf 11,542 s mission operation 1642 pps airflow rate (Mach 2.8, 10,000 ft above sea level conditions) 1550 K chamber temperature	**$2.28B**	485 MW_{th}, 37,900 lbf, 303 s at full power, 1660 pps, 1566 K chamber temperature (Mission operation length assumes 11,000 km trip to missile target at constant Mach 2.8)	Reactor readiness: TRL 6 (reactor ground test, near prototypic) Fuel Readiness: TRL 6 (fuel tested at prototypic conditions) Key Remaining Challenges: Demonstrating rapid startup, automatic control rod operation at operating conditions, demonstrating operation for entire mission duration
GE-710 Program	1962–1968	HEU W-UO_2 ceramic-metallic (CERMET) elements, Ta-W-Hf cladding Propulsion system target: 3073 K fluid temperature	**$117M**	0.872 MW_{th}, 121.67 hr at full power, 1811 K chamber temperature Static out of pile: 12,072 hr, 1922 K	Reactor readiness: TRL 4 (full reactor never tested, but constituent elements tested)

		33,000 W/cm^3 fuel power density 10 hour operating lifetime Brayton power system target: 1473 K fluid temperature 17,000 W/cm^3 fuel power density 10,000 hour operating lifetime		Flowing (dynamic) out of pile: 8,005 hr, 1922 K In-pile: 8,500 hr, 1811 K	Fuel readiness: TRL 5 (fuel elements extensively tested in near prototypic environment) Key remaining challenges: demonstration of fuel bundles at prototypic conditions in-pile
ANL's Nuclear Rocket (NR) program	1963–1966	HEU W-UO_2 CERMET, Be reflector, Inconel pressure vessel Target: 200 MW_{th} reactor 10,530 lbf engine, fission-product loss (in clad) <1% 821 s specific impulse 2507 K chamber temperature	Unknown	Out of pile: 49 hr at full power, 2973 K chamber temperature Flowing (dynamic) out of pile: 12 hr, 40.3 kW, 2723 K; 6 min, 613 kW, 2543 K (heater failed in both small and large loop) In pile: 180 MW_{th}, 3 s at full power, 3023 K chamber temperature	Reactor readiness: TRL 3 (reactor designed but not tested) Fuel readiness: TRL 5 (fuel elements tested in a relevant, near prototypic environment) Key remaining challenges: testing of full fuel elements at prototypic conditions, testing fuel bundles at prototypic conditions
Space NTP program	1987–1994	HEU UC_2 particles coated with pyrolytic C and ZrC (particle bed reactor) Target: 40,000 lbf 1000 MW_{th} 40 MW/L 3000 K chamber temperature 600 s at full power, three starts	**$373M**	In pile: 3000 K, 600 s at full power (particles failed) 1.5 MW/L, 2300 K outlet temperature, 2 cycles (hot frit cracked)	Reactor readiness: TRL 3 (design finished but no full-scale experiments finished) Fuel readiness: TRL 5 (fuel concept evaluated, prototypic conditions elusive) Key remaining challenges: successful prototypic testing of partial and full fuel elements, fuel assemblies and engine subsystems

3.1 Rover, NERVA and the Nuclear Furnace (NF-1) 1955–1972

The Rover and Nuclear Engine for Rocket Vehicle Application (NERVA) Programs were the most extensive effort thus far to develop an NTP rocket engine. Project Rover started in 1955 to develop a nuclear engine for either space or terrestrial applications. The program was started under the Department of Defense and later moved to NASA, jointly with what was then the Atomic Energy Commission (AEC). Throughout the Rover/NERVA Program, significant strides in the development of NTP reactor components and model validation were made. Several successful ground-based reactor tests were performed as cost estimates for a full flight system escalated. Plans for an in-flight test reactor were never realized primarily due to the predicted cost of a full-scale ground test. The program was cancelled in 1972. The NERVA program demonstrated reactors and fuels for NTP engines at a TRL above anything else ever attempted. [Finseth 1991]

3.2 Pluto Program 1957–1964

The Pluto Program was led by what is now the Lawrence Livermore National Laboratory. The goal was to produce a reactor (the Tory series) that could power a ramjet cruise missile capable of flying at Mach 2.8 at 10,000 feet above sea level. [Reynolds 1961] Similar to Rover/NERVA, the Pluto Program ground tested nuclear reactors to demonstrate subsystem functionality and component survivability. The main reactors built and tested during the program were the Tory IIA and the Tory IIC. The Tory II-A reactor completed successful tests in 1961 at full power (i.e., 150 MW) and design point air flow rate (~650 lb/s), and a thermal stress test with lowered flow rate (thermal stresses twice that of nominal). The Tory IIC reactor completed three successful tests in 1964 at low, intermediate, and full-power, just after the program was officially cancelled.

3.3 GE-710 Gas Reactor Program 1962–1968

The GE-710 high temperature gas cooled reactor program focused on developing a fast-spectrum refractory metal reactor with hexagonal CERMET fuel elements for submarine, aircraft, and rocket propulsion applications. [710 Report Vols. I and III 1967] The reactor was demonstrated with a closed loop of neon coolant and an open loop of hydrogen coolant, with the open loop discontinued in 1963 to focus on the closed loop. The program was shifted to focus on space propulsion fuel elements in 1966, with detailed studies performed on CERMET fuel fabrication, testing, and qualification. The GE-710 Program was discontinued in 1968 due to budgetary constraints and to focus on the NERVA program. The major accomplishments of the GE-710 Program were the demonstration of the production of full-scale CERMET fuel elements and cladding and successful in-reactor and non-nuclear testing of subscale fuel components. The program's final focus before cancellation was on the design of a 200 kW_e Brayton cycle space power unit (for crewed space stations) and supporting an in-pile test loop capable of testing full-scale fuel elements under dynamic coolant flow conditions.

3.4 Argonne Nuclear Laboratory Nuclear Rocket Program 1963–1966

Beginning in January 1963, Argonne National Laboratory embarked on a preliminary design study of a nuclear rocket powered by a refractory metal-based fast neutron spectrum reactor. [Nuclear Rocket 1966] This program was undertaken with the guidance of the AEC's Space Nuclear Propulsion Office (SNPO). The design study focused on a 2000 MW_{th} (100,000 lbf) and a 200 MW_{th} (10,000 lbf) reference system, with a core of tungsten-based cladding and CERMET fuel with a beryllium oxide reflector. The program successfully produced tungsten clad CERMET fuel that could withstand high temperatures (2700 K) for tens of hours and dozens of thermal cycles, and test specimens exposed to high pressure and high flow rate hydrogen performed well. Development of fuel element fabrication was cancelled in July 1966, prior to completion, due to budgetary constraints, but the program provided confidence in the feasibility of a full engine.

3.5 Space Nuclear Thermal Propulsion Fuel Testing 1988–1995

The SNTP program ran from 1988 to 1995 and was focused on particle bed reactor (PBR) technology. The program had the goal to develop a high-performance rocket engine that would more than double the performance of the best conventional chemical rockets at the time. The SNTP program was born during the Cold War under the Strategic Defense Initiative. Despite the time pressures implied by that context, the program planned a comprehensive analysis and supporting experimental program to mature the design and minimize the risks of potential use. These spanned the spectrum of simple blowdown experiments to validate the potential power density capabilities of PBR fuel, critical experiments to validate the computational tools that were used in the design to confirm predictions of performance, and limited experiments on subscale fuel elements, constrained by the availability of appropriate facilities. A planned reactor to test PBR fuel elements (and potentially those of other concepts) underwent a successful design review but did not proceed further. The SNTP program approach for testing and validating the performance of PBR fuel elements (and likely relevant for other concepts) is described in Appendix B.

3.6 Summary of Historic Testing Programs

Historic testing programs have shown that many pathways exist for testing nuclear fuels and other critical reactor components. Overall, these programs demonstrated that an incremental development approach led to more rapid feedback between testing and manufacture activities and reduced overall test program complexity for performing fuel qualification. This approach can also be applied to the development of moderator or other in-reactor components. In the following section, facilities in operation today that may support incremental development activities are identified. For each facility, advantages or disadvantages of the facility in achieving prototypic testing conditions as defined in Table 1 are identified.

4. EXISTING FACILITIES

4.1.1 NTREES

The Nuclear Thermal Rocket Engine Environmental Simulator (NTREES) test facility (Figure 1) at NASA Marshall Space Flight Center simulates the non-nuclear conditions of a nuclear thermal rocket engine (pressurized, hot, flowing hydrogen) on full scale NTP fuel elements. NTREES uses induction heating to raise fuel specimens to prototypic NTP temperatures (up to 3700 K, demonstrated) and is capable of gaseous hydrogen flow rates of up to 250 g/s (60 L/s H_2, pressure dependent) and pressures up to 1000 psi (7 MPa), respectively. Allowable test article sizes are up to 2.5 m in length and 0.3 m in diameter. Test articles should exhibit some electrical conductivity to allow for induction heating. The induction heater can deliver up to 1.2 MW of power. This facility is currently in use for experiments in support of the SNP Program and can handle enriched fuel elements.

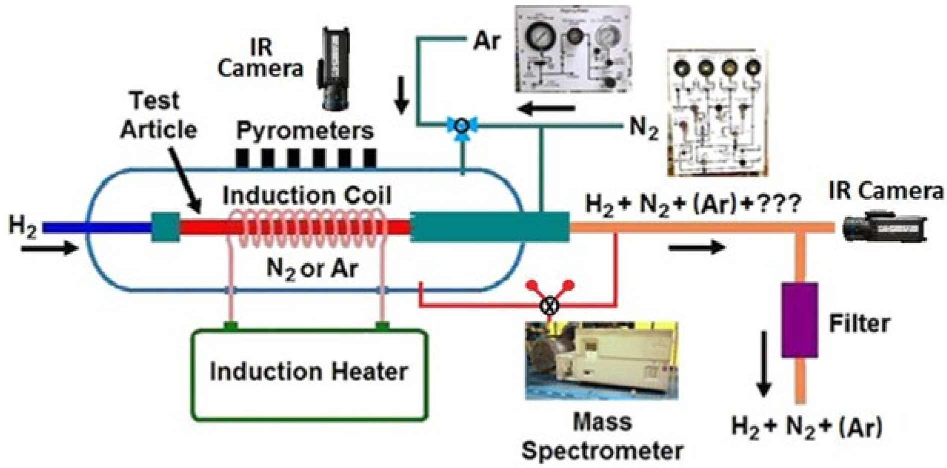


Figure 1. NTREES Facility overview and description.

Advantages for this option include an existing test facility that is hydrogen capable and allows for testing of fuel elements under the full range on non-nuclear conditions expected for most reactor designs with achievable:

- Test article temperatures of at least 3800 K
- Hydrogen pressures up to 1000 psi
- Total hydrogen flow rates up to 250 g/s per test article
- Steady-state or thermal cycling experiments
- Testing full-scale fuel elements
- Able to test with propellants other hydrogen
- Startup and shutdown transients can exceed ~10^2 K/s
- Low temperature and high hydrogen pressure conditions required for moderator element testing easily achievable

Disadvantages for this option include that it does not simulate nuclear heating or irradiation effects expected during NTP operation. This results in slightly different in element temperature gradients and thermal stress distributions than expected during operational conditions.

4.1.2 TREAT

The Transient Reactor Test Facility (TREAT) is an air-cooled, graphite-moderated, thermal spectrum test reactor. The fuel is HEU, dispersed in a graphite matrix. The reactor was built by Argonne National Laboratory in 1958 as a transient reactor to test nuclear reactor fuels under extreme conditions, primarily simulating conditions from small transients to reactor malfunctions. The reactor was in full operation until 1994, when it was shut down and put on stand-by due to declining interest in commercial nuclear reactors and associated research. As interest in the research of advanced nuclear fuel technology increased after the events at Fukushima in 2011, the decision was made to restart the TREAT reactor. The reactor restarted in 2018 at a cost of $75 million. [US DOE 2017] The reactor is located at what is now Idaho National Laboratory (INL) located outside of Idaho Falls, Idaho.

Facility changes underway at TREAT in support of the SNP Program are the capability to flow gaseous hydrogen through experiments during irradiation and a larger test vehicle for experiments. Current experiment OD is limited to approximately 2.54 cm. The larger test vehicle will easily accommodate 5 cm OD samples. The active core length in TREAT is 1.22 m long, but the flux profile drops off rapidly outside of the center 0.61 m resulting in an effective experiment length of 0.61 m (Figure 2). Preparations are also underway for TREAT testing of a single NTP fuel element in combination with moderator and structural materials in a series of experiments known as the prototypic reactor irradiation for multicomponent evaluation (PRIME) tests. The PRIME tests will include exposure to flowing hydrogen during irradiation. TREAT was also considered as an option to test PBR fuel elements under the SNTP program, but not pursued beyond a preliminary assessment [Todosow et al. 1993].

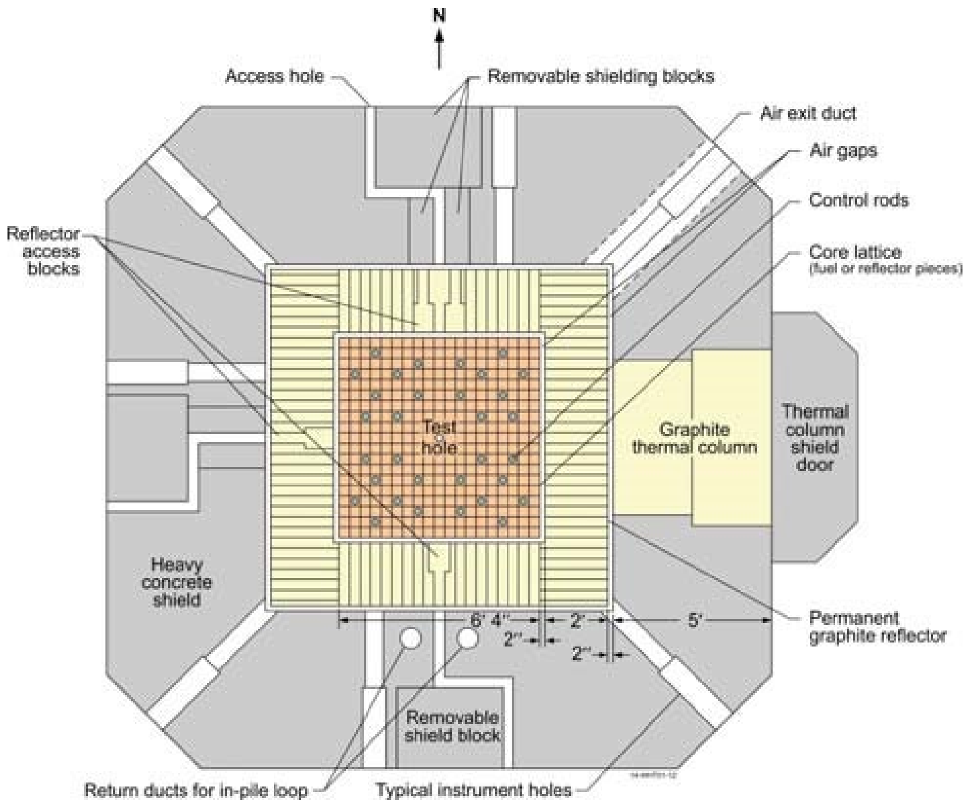


Figure 2. Cross-section of TREAT.

Advantages for this option include that TREAT is

- An existing reactor that is currently in use for some SNP testing
- Capable of performing a variety of fuel and material transient power experiments
- Able to test experiments at NTP power-density design parameters
- Capable of incorporating flowing hydrogen gas into experiments during irradiation (starting in 2023)
- Able to create experimental transient that exceed startup times.
- Provides some separate effects data for neutronic and multi-physics computational codes

Disadvantages for this option are that TREAT

- Will not be able to match startup and shutdown times
- Has operational cycles that are much shorter than NTP cycles (i.e., seconds vs. hours)
- Cannot achieve time at temperature or total fluence for the test article

- Is not able to test prototypic segment or full-length fuel elements or a multiple unit cell fuel / moderator segments

4.1.3 ATR

The Advanced Test Reactor (ATR) was built at INL in 1967 with the main purpose of studying the effects of long-term exposure on materials to a high thermal neutron flux. ATR is a pressurized light-water reactor with HEU fuel and a beryllium reflector that operates at temperatures and pressures much lower than those in commercial reactors. Typical operating conditions in ATR are 360 psia and 250 MW_{th} (producing coolant temperatures of 160°F). This reactor has a unique cloverleaf-shaped core design that allows each corner, or lobe, to be operated at different power levels (Figure 3). This design allows the reactor to facilitate several isolated experiments, providing a more efficient means of research. The reactor is currently undergoing its regularly scheduled core internals changeout overhaul, which will prepare the reactor for its next decade of use. The experimental facilities do not currently accommodate flowing hydrogen, and the current overhaul will not add that capability. Experimental facilities in ATR are in constant demand; adding new experiments to the ATR schedule is challenging.

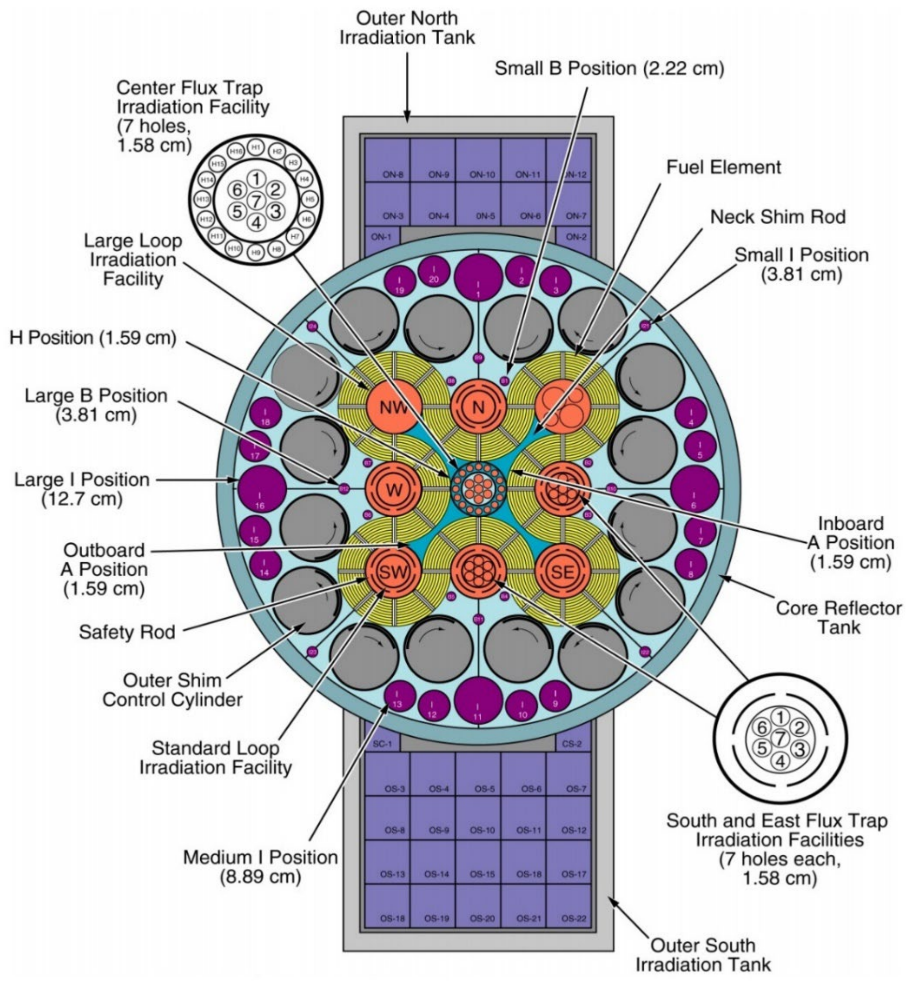


Figure 3. Cross-section of ATR.

Advantages for this option include that it is an existing reactor capable of performing a variety of fuel and material experiments. Additionally, ATR:

- Can achieve NTP temperature and power-density design parameters for test articles

- May be able to incorporate hydrogen gas into an instrumented-lead irradiation test
- Can test full-length NTP elements in certain flux traps within the reactor.

Disadvantages for this option include that ATR

- Will not be able to match startup and shutdown times
- Has operational cycles that are much longer than NTP burn cycles (days vs. hours)
- Provides total fluence and dpa on the test that would far exceed NTP lifetimes
- Starts up over several days before peak flux levels are achieved
- Requires an extensive redesign and update of the reactor safety basis before a hydrogen flow loop that meets prototypic pressure and flow rates could be incorporated
- Provides limited integrated or separate effects data for neutronic and multi-physics computational codes needed for NTR designs
- Currently experiences high customer demand for available irradiation positions.

4.1.4 HFIR

The High Flux Isotope Reactor (HFIR) was built in 1965 with a focus on isotope research and production. Today, this 85 MW reactor is known worldwide as a source of the highest neutron flux for research. The reactor is a light-water-cooled and water-moderated reactor with HEU fuel and a beryllium reflector. The reactor design supports irradiation using multiple irradiation locations in the central flux trap and beryllium reflector and a hydraulic tube for short-term irradiations (Figure 4). This reactor is located at Oak Ridge National Laboratory in Oak Ridge, Tennessee.

Advantages for this option is that it is an existing reactor, capable of performing a variety of fuel and material experiments. Additionally, HFIR:

- Can achieve NTP temperature and power density design parameters for the experiments required
- May be able to incorporate hydrogen gas into an instrumented lead type irradiation test
- Is capable of using small sample test leads for targeted separate-effects testing.

Disadvantages for this option include that HFIR

- Will not be able to match startup and shutdown times
- Has operational cycles that are much longer than NTP burn cycles (days vs. hours)
- Creates total fluence and dpa that would far exceed NTP lifetimes on the test
- Would not be able to test full-length NTP elements
- Would require extensive safety analysis and possibly an update of the reactor safety basis to add hydrogen flow capability, and still might not achieve prototypic pressures
- Currently experiences high customer demand for available irradiation positions.

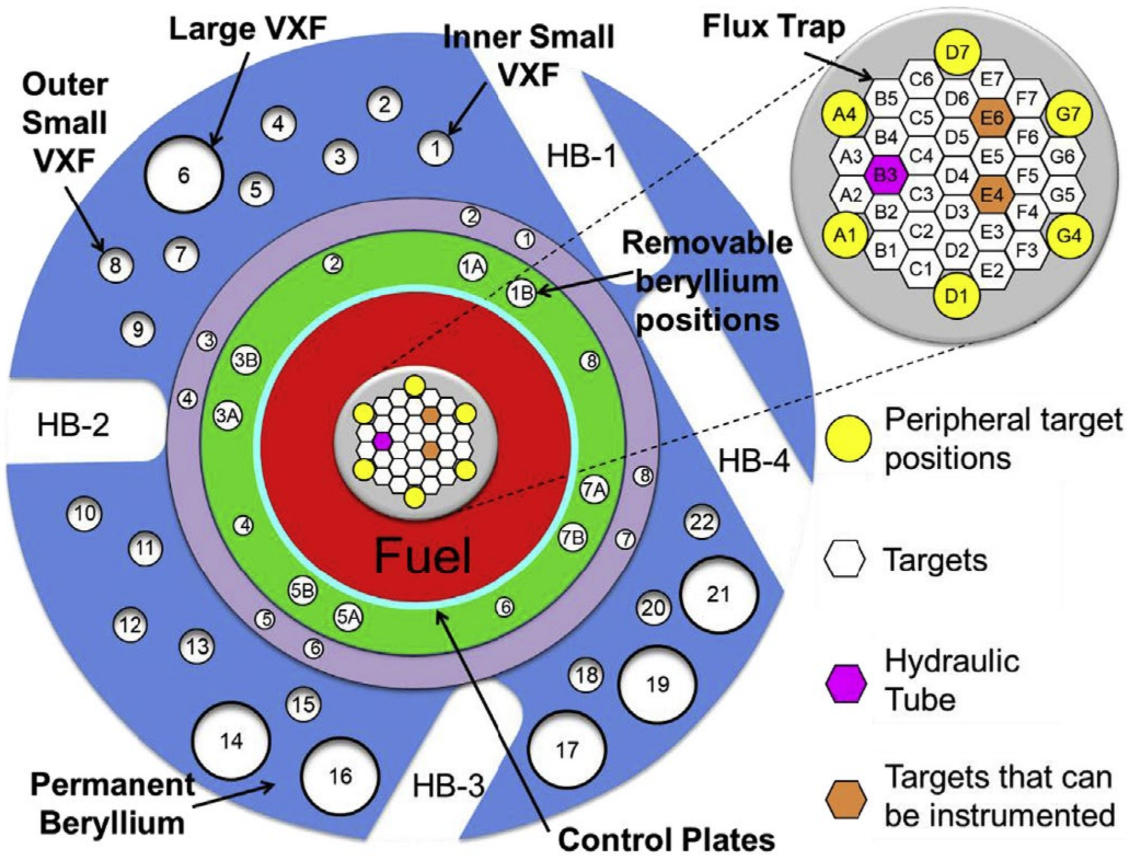
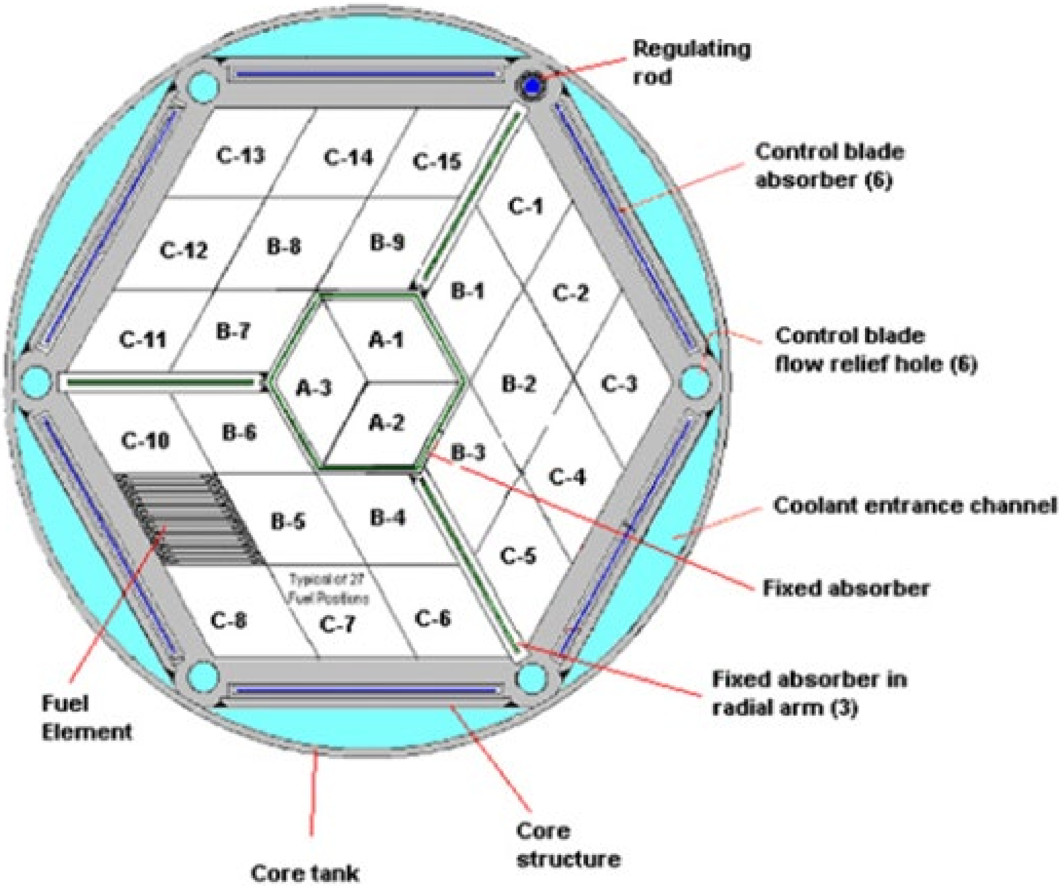


Figure 4. Cross-section of HFIR.

4.1.5 MITR

The Massachusetts Institute of Technology Reactor (MITR) is a university test reactor operated by MIT. The MITR is light water cooled, and moderated with heavy water, with a graphite reflector. It is capable of long duration irradiation of subscale fuel or non-fuel material coupons. MITR fuel elements are composed of a uranium-aluminum alloy fuel plates with aluminum clad which are assembled into rhomboid-shaped assemblies. This allows for MITR to achieve an average power density of 70 kW/L with a peak thermal and fast neutron flux of 6E13 and 1.2E14 n/cm^{2}s respectively. The MITR reactor has several different in-reactor irradiation positions available that average a maximum test article length of 0.61 m and 0.05 m diameter (Figure 5). The reactor positions are capable of operation at up to 1173 K, and one port is capable of being adapted for a custom irradiation loop. This custom port currently is equipped to allow for testing of high temperature gas reactor materials at 1273–1873 K under helium and nitrogen flows. The reactor is capable of long or short duration exposures using a pneumatic tube transfer system. Pneumatic tubes are capable of testing specimens that are between 2.5 and 3.49 cm in diameter in polyethylene (for short duration irradiation) or titanium (long duration irradiation) rabbits. MITR has demonstrated use of its pneumatic tube transfer system for irradiation of NTP moderator and fuel materials under Ar-H_2 mixtures through recent SNP test activities.



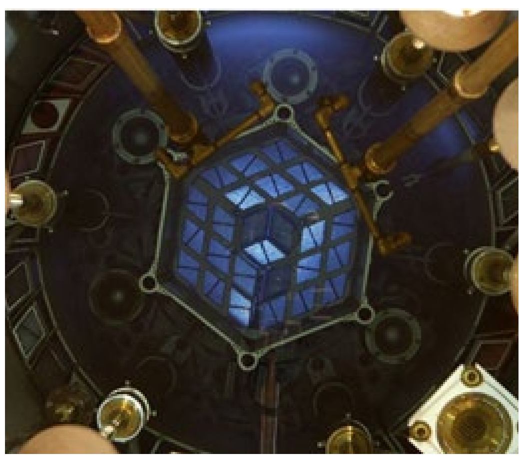


Figure 5. Top: MITR cross section with experimental positions indicated. Bottom: top view of reactor following operation.

Advantages for this option include that it is an existing reactor and currently in use for some SNP testing. Additionally, MITR offers

- An irradiation port that could be customized for specific NTP testing needs
- Demonstrated ability to test fuel or non-fuel materials in static hydrogen

- Capability of achieving higher power densities, neutron flux, and temperatures than most alternative university test reactors
- A pneumatic transfer system that can allow for test duration and total fluence to be better controlled together with a limited ability to mimic startup and shutdown times.
- Provides some separate effects data and benchmarking for neutronic and multi-physics computational codes

Disadvantages for this option include:

- Irradiation port sizes that will not allow for testing of full-scale NTP samples
- No existing flowing hydrogen gas loop
- Demonstrated test temperature that are limited to 1873 K
- Static hydrogen capsules only, with no flowing hydrogen capability at any temperature.

4.1.6 Other University Reactor Facilities

Alternative university test reactors exist which can enable irradiation testing of NTP materials, but they are less capable due to lower flux, with the exception of the University of Missouri Research Reactor (MURR), which currently has no availability for NTP experiments. These existing facilities will be limited in total power, test article dimensions, as well as achievable total power, power density, neutron flux, irradiation duration, and irradiation temperature. Included in Table 4 is a summary of available university reactors with the highest available flux. Table 4 also lists information about available irradiation port sizes and other unique capabilities of the reactors.

Table 4. University reactor facilities with the highest available flux.

Reactor Name	University Name (Location)	Reactor Thermal Power [MW]	Peak Thermal Flux [n/(cm^{2}s)]	Peak Fast Flux [n/(cm^{2}s)]	Neutron Beam Port Size(s)	Other Unique Capabilities
MURR	University of Missouri, Columbia (Columbia, MO)	10	6E14	1E14	6-in. diameter Helium-filled tubes	
MITR	MIT, (Cambridge, MA)	6	7E13	1.7E14	3 1.25-in. dry tubes	Lazy Susan, Thermal Column, Pneumatic transfer system (Rabbit system, limited to 0.9-in. diameter by 4-in. length)
Oregon State Training Reactor (OSTR)	Oregon State University (Corvallis, OR)	1.1	1E13	1E13	1 cold, 3 7.5-in. diameter, 1 3.9-in. diameter, central thimble 1.33-in. diameter	2 dry irradiation tubes, 1.25-in. diameter, fast flux tube and fast neutron irradiator tubes air-filled, 6.25-in. diameter, and 10-in. diameter respectively

Reactor Name	University Name (Location)	Reactor Thermal Power [MW]	Peak Thermal Flux [n/(cm^{2}s)]	Peak Fast Flux [n/(cm^{2}s)]	Neutron Beam Port Size(s)	Other Unique Capabilities
Nuclear Engineering Teaching Laboratory (NETL) TRIGA	University of Texas at Austin (Austin, TX)	1.1	2.7E13	4.8E13		
Penn State Breazeale Reactor (PSBR)	The Pennsylvania State University (University Park, PA)	1	3.3E13	3E13		
University of Wisconsin Nuclear Reactor (UWNR)	University of Wisconsin, Madison (Madison, WI)	1	3.2E13	3E13		

Advantages for using university facilities are that they represent existing test facilities that have access to graduate students and are well suited to separate-effects testing and material assessments on a small scale, with shorter time scale between concept, irradiation, and post-irradiation examination (PIE).

Disadvantages of using university facilities include that these facilities are

- Limited as to the types of material that can be tested, although most can handle HALEU
- Limited in the sample sizes they can handle
- Their employment of static hydrogen capsules only, with no flowing hydrogen capability (at any temperature) with some facilities needing to develop procedures for handling even static hydrogen.

5. CURRENT PLANS FOR NEW REACTOR TEST FACILITIES

5.1.1 VTR

The Versatile Test Reactor (VTR) is a sodium-cooled fast neutron spectrum reactor design, supported by DOE, that has released a draft environmental impact statement and is awaiting federal funding for future development. [Cating 2021, US DOE 2019] After the last domestic fast reactor, the Fast Flux Test Facility, was decommissioned in 1992, the DOE Office of Nuclear Energy and others cited a continued need for a domestic fast-neutron nuclear testing facility. [Assessment 2017] Fast reactors are a vital component in nuclear fuel research and are essential for studying and creating the new types of nuclear fuel needed for advanced reactor designs and fuel cycles. This reactor is modeled after the Experimental Breeder Reactor-II and GE-Hitachi's Power Reactor Innovative Small Module (PRISM) sodium cooled reactor. The anticipated power levels are around 300 MWth, with a proposed first fuel of metallic U-10Zr, followed by a U-Pu-Zr alloy to increase performance. The earliest that VTR could be operational is 2026, provided construction begins in 2023. Specifications and capabilities for testing will continue to evolve through the approval process. The VTR is currently awaiting funding for future development.

5.1.2 MARVEL

DOE supports the development of the Microreactor Applications Research Validation and EvaLuation (MARVEL), to be housed within the TREAT facility. [US DOE 2021]This project will test and establish how microreactors, including designs from private industry, can integrate within our current electrical grid or with other technologies. While this is not likely to be suitable for SMART testing, the presence of MARVEL will impact experiment availability at TREAT (Figure 6). The two reactors will be located in the same building and will not be allowed to operate simultaneously.

Figure 6. The MARVEL reactor concept, illustrated in a TREAT storage pit.

5.1.3 DOME and LOTUS

The National Reactor Innovation Center, led by INL, is working on 2 reactor test facilities, the Demonstration and Operation of Microreactor Experiments (DOME) and the Laboratory for Operation and Testing in the U.S. (LOTUS). Plans are to have these facilities available for use in the 2024-2025 timeframe. Pre-conceptual design reports for both facilities were released in 2020. [Balsmeier and Burnett 2020, Balsmeier and Core 2020]. The DOME is intended to house demonstration reactor systems that operate at less than 10 MWt while LOTUS is intended for systems that operate at less than 500 kWth. These facilities are not intended to house a permanent test reactor. Their mission is to accelerate the demonstration of advanced reactors for commercial applications.

5.1.4 IRIS

The Los Alamos National Laboratory microreactor program has developed a conceptual test bed, called IRIS, for sub-scale demonstration of microreactor cores [Blood et. al. 2021].

A unit cell from a prototypic reactor is nested inside a reflector and/or high-enriched uranium fuel and is designed to closely replicate the neutron environment of the full-scale reactor (Figure 7). Achievable power limits are determined by details of the test of interest, but dose restrictions of the National Critical Experiments Research Center (NCERC) restrict demonstrations to around 1–2 kW_{th} over a couple of days of operation. Facility restrictions on power levels will likely require electrical heater elements within the system to achieve the high temperatures needed for NTP fuel testing. The IRIS system, consisting of neutronic and thermal drivers within nested prototypic unit cell, can be insulated and/or placed under vacuum. Customization of the neutron energy spectrum and heater elements leads to a versatile test bed for a range of reactor designs.

The system is lifted into a neutron reflector for reactivity control on the top of Comet, a vertical critical assembly machine (see Figure 8). Heat transfer elements, such as heat pipes, may extend through the reflector and attach to a prototypic power conversion system. Comet has the capacity to hold up to

20,000 lbs on the stationary platform and 2,000 lbs on the moveable platform, with dimensions not exceeding a height of 71 cm and width of approximately 53 cm. Systems that meet the volume and weight restrictions can be operated and tested as a whole.

Thermocouples will be integrated into the IRIS system, and reactivity measurements will be made with various radiation detectors surrounding the system to record thermal and reactivity behavior continuously for the extent of operation.

While the plans for IRIS at this time do not include the development of a hydrogen supply system, there are no restrictions on the use of hydrogen with IRIS.

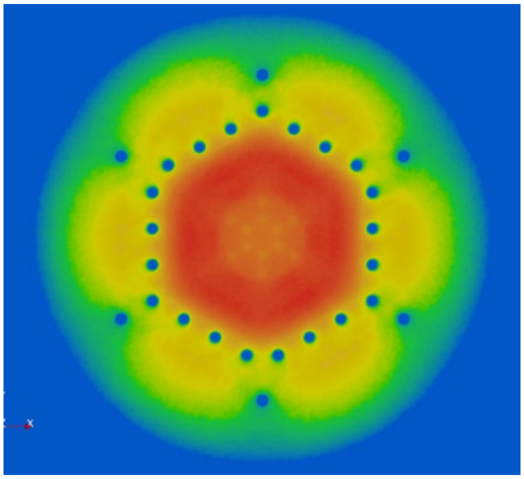


Figure 7. Cross-section visualization of neutron flux generated by the IRIS source component.

Figure 8. Photo of the Comet critical assembly machine at NCERC.

5.1.5 NEXT Lab

Abilene Christian University, located in Abilene, Texas, is developing infrastructure for the Nuclear Energy eXperimental Testing Laboratory (NEXT Lab) to support education, training, development, and deployment of advanced nuclear systems in an easy to access academic setting. The Science and Engineering Research Center (SERC) is a new multiuse facility designed to house a number of radiation producing systems, with ample space for construction, operations, and research. The SERC is located on a recently acquired 12.5-acre property adjacent to the campus that has direct access for road shipments, with airport and rail services nearby.

The SERC contains office, laboratory, construction, and auxiliary systems spaces to support the large, flexible, Cat II safeguards reactor bay. The 50-ft × 120-ft ground-level reactor bay is serviced by a 40-ton NOG-1 crane and surrounds a subterranean shielded trench that is 15 ft wide, 25 ft deep, and 80 ft long. The trench is designed to be segmented using moveable shielding to neutronically isolate a number of drop-in reactor systems. The reactor bay has assembly system support areas, a fuel storage pit, 6500 lbs/ft^2 of floor loading, and at least 250 kW of power that can be readily increased. As a test bed, the SERC can be used to house subscale engine-reactor testing or potentially some form of driver core on the level of IRIS discussed previously, or an ATR-Fuel Driver Core, which will be discussed in the next section. The facility is intended for the operation of reactors, rather than test articles, so a driver core plus a test article or some type of engine-reactor subassembly would be inserted into the trench, operated for minutes to a few hours, and then removed in its entirety in preparation for shipment to an off-site PIE facility.

Figure 9. Shown here is a cross-section of the SERC through the reactor bay and trench.

Advantages for using the planned IRIS and NEXT facilities are

- Once constructed, the use of existing facilities is the most economical and easy to access
- The facilities require Category II safeguards
- The test beds are intended to be very flexible and to accommodate a wide variety of test article shapes, sizes, and test conditions.

Disadvantages of using planned facilities include that

- Completion dates are uncertain and may not meet the SNP schedule

- Irradiation conditions and availability of flowing hydrogen are undetermined
- Both IRIS and NEXT are likely to have limitations on the total activation of test articles, which results in limits to accumulated dpa, similar to the limitations seen in low-flux facilities, and will also limit the duration of the irradiation.

6. NEW CONCEPTS

As shown in Table 5, existing facilities cannot meet a critical subset of the criteria for a SMART facility: that they be designed to improve the TRLs of reactor and engine systems needed for an NTP device such that the risks are reduced prior to a full-scale ground test. Meeting all criteria for a SMART facility is best done through a customized facility. New facilities are difficult to build for many reasons, primarily the time and effort needed for licensing or authorization to operate and cost. Both of those obstacles can be reduced by relying on existing reactor designs or components, such as fuel, that are already in use. However, once a facility such as this is established, it could be used for testing improved fuels and moderator and core materials for a variety of programs, both public and private. It also would be instrumental in developing Generation 2 and 3 NTP reactors or revolutionary new NTP concepts. Two options for a SMART facility based on existing technology are described here. The first is the reconfiguration of the classic TRIGA design. The second is a totally new facility based on existing ATR fuel, combined with well-understood design principles of a driver core.

The authors acknowledge that the use of current ATR fuel provides an additional challenge for a new facility because the existing ATR fuel is HEU. Efforts to eliminate the use of HEU in research facilities and replace them with HALEU fuel are underway worldwide. Replacement of the existing ATR HEU fuel with HALEU fuel has been discussed though such a replacement is not yet planned. The analyses in Section 6.2 consider a driver core using ATR fuel with enrichment levels of 93% and 20% in the event that the lower enriched ATR fuel is certified for use in time for construction of a SMART facility.

There are several foreign nation state facilities that offer capabilities to irradiate NTP fuels and system components under prototypical conditions, with some limitations on hydrogen flow. Under current SNP Program restrictions, primarily due to export control concerns, these facilities are not an option for testing NTP fuel. If the use of a foreign facility becomes possible, Appendix C contains brief descriptions of facilities of interest.

Table 5. Comparison of a subset of SMART and existing test facilities and capabilities.

NTP Parameter	Goal	SMART 14-MW TRIGA	SMART ATR-Fuel Driver Core	TREAT	NTREES	MITR	MURR	ATR Reactor Loop	HFIR
Average Power Density (MW/L)	5	0.1	5		>2MW/L (by induction heating only)	< 1	0.303	1	1.7
Thermal Neutron Flux (n-cm^2-s)	1E15	1E14	1E15	~7E12 per MW	No nuclear heating	6E14	6E14	1E15	2.5E15
Fuel Region or Test Article Length	1 m	0.5 m	1 m	1.2 m active core length, 0.61 m for best flux profile	2.5 m test article length	0.61 m	0.61 m	1 m	0.51 m
Room for 3-Unit Cells or a Fuel/Moderator Bundle	7–13 cm	18 cm	13 cm	<6 cm	30 cm	4.572 cm	13.6 cm	16 cm	<6 cm
Peak Test Article Fuel Temperature	3000 K+			2700 K	3800 K achieved to date	2500 K		3000 K	2500 K
Fuel Temperature Ramp Rate in Test Article	20–100 k/sec			100 K/s is in use for current experiments	>100 K/s				
Lifetime	240 minutes							>24 hours	>24 hours
Operation Time	20 min. operation with cooldown. Repeat five times			< 40 Sec			Main function is isotope production so not run for short durations	>24 hours	>24 hours
Hydrogen Pressure through Core	7–11 Mpa			Peak pressure of 1000 psig (~7 Mpa)	Peak pressure of 1000 psig (~7 Mpa)	Static Pressure (has used GH_2 in rabbits)	Static pressure (has not used GH_2)		Has flow capabilities but has not used GH_2
Hydrogen Mass-Flow Rate	~2 g/s per channel			Up to 200 g/s total (in progress for 2023)	Max of 250 g/s (by regulation)				

LEGEND:
Green: Goal achievable
Yellow: Can obtain useful data or modify to achieve goal
Red: Goal not likely

6.1 SMART Option: Use a 14-MW TRIGA Reactor

A 14-MW TRIGA research reactor is a viable alternative for a SMART driver core. TRIGA reactors at this and similar power levels have been built and operated as discussed previously. A 14-MW TRIGA reactor can provide in-core experiments with thermal fluxes in excess of 2E14 n/cm^2•s and fast fluxes (>0.1 MeV) of 2E14 n/cm^2•s (Table 6). The average core power density is about 118 kW/liter (INR TRIGA).

Table 6. Performance characteristics of high-power TRIGA reactors.

Parameters	Standard 10-MW TRIGA	INR TRIGA
Power (MW)	10	14
No. of fuel clusters	30	27
Fuel rods/clusters	4 × 4	5 × 5
No. of fuel rods	480	675
Length of fuel (mm)	559	559
Diam. of fuel rod (mm)	13.7	13.7
Avg. power density (kW/liter)	87	118
Reflector type	H_2O	Be/H_2O
Flux (thermal) central core region (n/cm^2/s)	2.9 (14)	2.8 (14)
Flux (fast), central core region (n/cm^2/s)	2 (14)	2.4 (14)

The core contains 27 hexagonal clusters, each of which contains 25 TRIGA 13.7-mm-diameter fuel rods. The vertical height of the fuel is 559 mm. The fuel assembly contains 25 fuel rods fit together in a 5 × 5 square lattice. The assembly uses Al 6061 tubes with Inconel 600 spacers. Cladding is either 304 stainless steel or Incoloy 800. The fuel assembly has cast aluminum alloy fittings on the top and bottom ends. The top fittings allow handling of the assembly and control of the water flow. The fuel assembly and core elements are placed into lower reactor grid with a lattice pitch of 90 mm. The fuel area is surrounded by vertical blocks of beryllium reflectors and also by the reactor grid. The core also contains eight square guide tubes for reactor control rods (Figure 10).

TRIGA low enriched uranium (LEU) fuel is very robust. Its clad is rugged enough to provide a large margin of safety, including during such abnormal events as loss of forced cooling. The UZrH fuel matrix, together with the Incoloy 800 cladding, assures that no fuel damage will occur during any credible reactivity accident. A high performance 10–15 MW TRIGA reactor is particularly user friendly.

Typically, at least 2, and up to 6, in-core experimental regions are available for the thermal or fast neutron irradiations. The experimental locations have been measured for fast neutron irradiations with the assumption that each region was filled with an experiment equivalent to a metal with density half that of stainless steel. Core configuration of up to 35 fuel clusters have been examined to provide excess reactivity if needed.

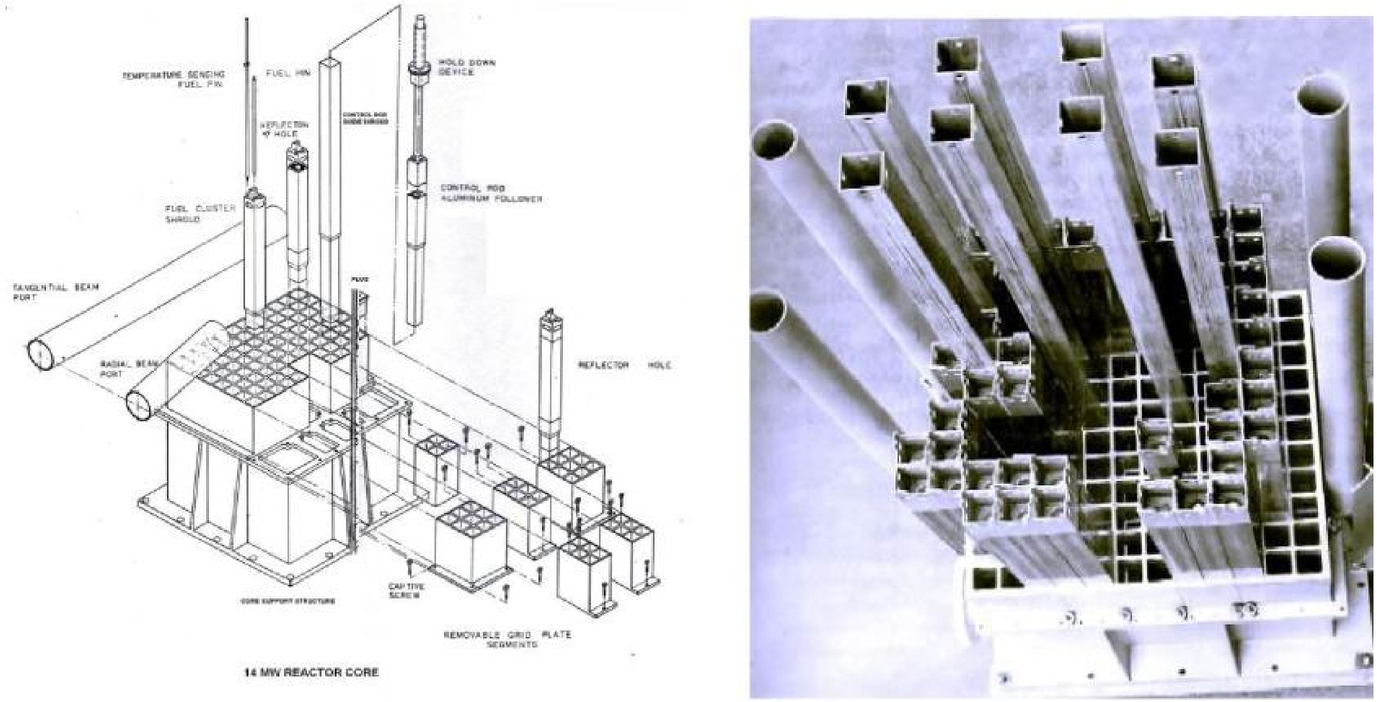


Figure 10. Schematic and picture of a 14-MW TRIGA's core structure.

D	D	D	D	D	D	D	D	D	D	D
R	R	R	R	R	R	R	R	R	D	D
R	R	F	●	R	●	F	R	R	D	D
R	R	F	F	D	F	F	F	R	D	D
R	R	●	F	●	F	D	F	R	D	D
R	R	F	F	X	F	F	F	R	D	D
R	R	F	F	F	F	D	D	R	D	D
R	R	●	F	●	F	F	R	R	Shim	D
R	R	F	F	X	F	D	R	R	D	D
D	R	F	●	F	●	F	R	Shim	D	D
R	R	R	R	R	R	R	R	R	D	D
D	D	D	D	R	R	R	D	D	D	D

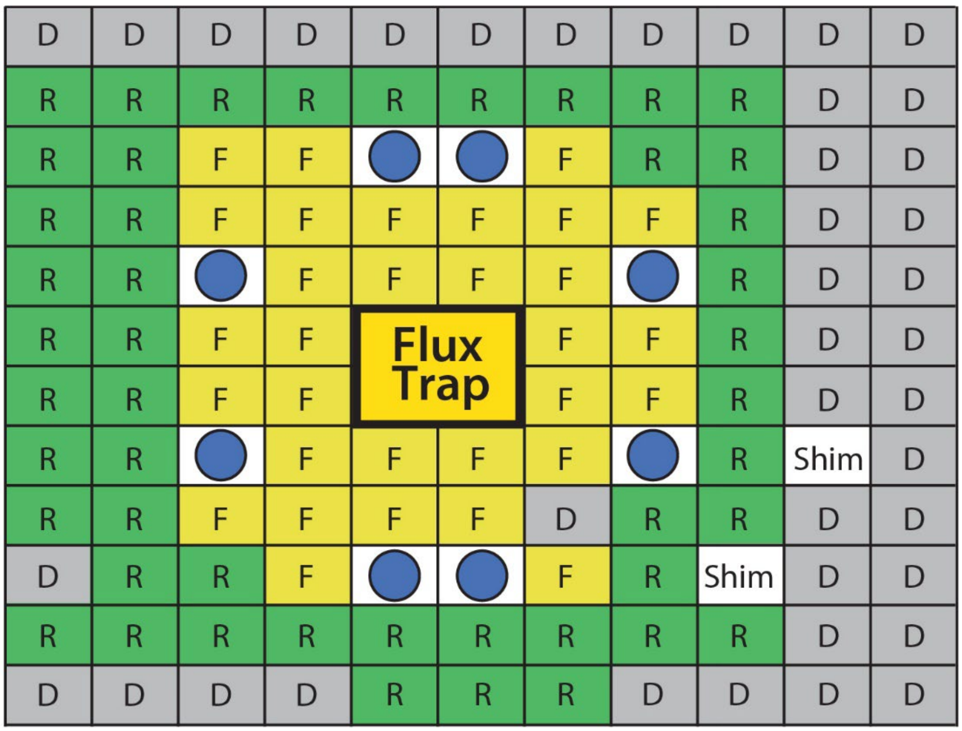
Legend:

D: plug, R: Be reflector, X: flux trap (experiment), F: fuel, ● control rod

Figure 11. Core configuration of a 14-MW TRIGA reactor.

6.1.1 Modifications to the TRIGA for SMART Applications

The major proposed modification would be to install a high temperature gas loop through the center of the core. If the lattice is unmodified, this would limit the test chamber to <9 cm. However, if the center four lattice positions were used, the test loop could be about 18 cm × 18 cm. The maximum irradiation test chamber length would be approximately 50 cm, which has an impact on temperature distributions in test articles. All current NTP conceptual designs have fuel element lengths of approximately 100 cm. Fuel temperatures will range from 400 K at the cold end of the element to peak temperatures approaching 3000 K. Temperature gradients of the fuel must be limited to less than 200 K/10 cm to maintain prototypic temperature profiles. Because only segments of the fuel elements (<50 cm long) can be irradiated in the flux trap, the tests will have to be designed to the temperature regimes and power densities in separate tests (e.g., 400–1400 K, 1200–2000 K, and 1800–3000 K).



Legend: D: plug, R: Be reflector, Flux trap (experiment), F: fuel, control rod

Figure 12. Core configuration of a SMART 14-MW TRIGA.

6.1.2 Advantages and Disadvantages of the New SMART TRIGA Concept

Advantages for this option are that the new facility is based on an existing reactor design, licensed by the NRC as a non-power reactor. Additionally, a new facility:

- Would be able to achieve NTP time at temperature at certain power densities
- Is believed to be a reduced cost option over a new ATR-Fuel driver core
  - Approximately $200M for construction of a new SMART TRIGA
  - Approximately $150 million for design and installation of a gas loop in an existing facility
  - Existing fuel supply (HALEU)
- Has shorter time to operation for a new facility versus a new design
- Could be used to test and prototype effluent cleanup systems and processes before being installed in an NTP demonstration test facility.

Disadvantages include that the new facility:

- May not be able to achieve prototypic power densities, but instead, may be able to install driver fuel around test element to boost power density in the flux trap
- Will not be able to match startup and shutdown times
- May not achieve prototypic temperature, pressure, and coolant flow parameters
- Has a test element fuel length limited to less than 50 cm, requiring separate tests to cover the desired temperature regimes as described above.

6.2 SMART Option: An ATR-Fuel Driver Core Reactor

This option considers a SMART facility configuration based on the use of fuel elements from the ATR as it exists today and ATR fuel with a lower enrichment of 20wt%. These fuel elements are 45-degree segments, as shown in Figure 13. Eight of these elements can be formed into a ring surrounding a test chamber (see Figure 14) with a diameter of approximately 9 cm and a length of approximately 100 cm. The expected power should be on the order of 50 MW_{th}, with a neutron flux on the order of 10^{14} n/cm^2s. The reflector region could be beryllium, as in ATR, or preferably less-expensive materials like stainless steel, light or heavy water, or BeO. Thermal insulation and variations in coolant gases could be used to drive up temperatures in the test cavity.

To get a quantitative sense of whether a configuration based on ATR fuel elements to support SMART was worth exploring further, some initial scoping calculations were performed to model a simplified configuration.

A simplified Monte Carlo N-Particle (MCNP) model was developed based on the MCNP input file for ATR, listed in the “International Handbook of Evaluated Criticality Safety Benchmark Experiments” supplied by INL. The initial model assumed room temperature nuclear data from the Evaluated Nuclear Data File (ENDF/B), which is representative of zero-power criticality, and modeled the ATR fuel as a ring, without the discrete boundaries of the actual ATR fuel elements, which are 45-degree segments.

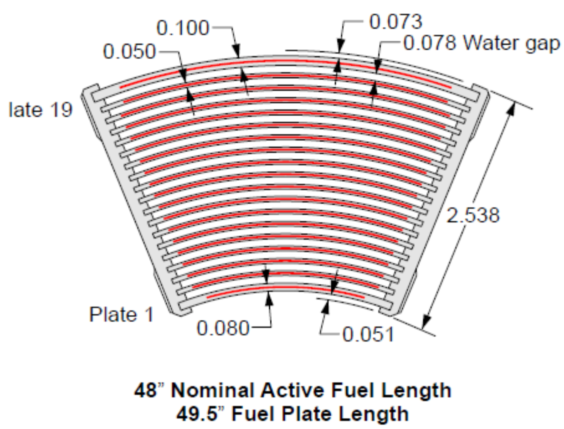


Figure 13. Diagram of an existing ATR fuel element segment.

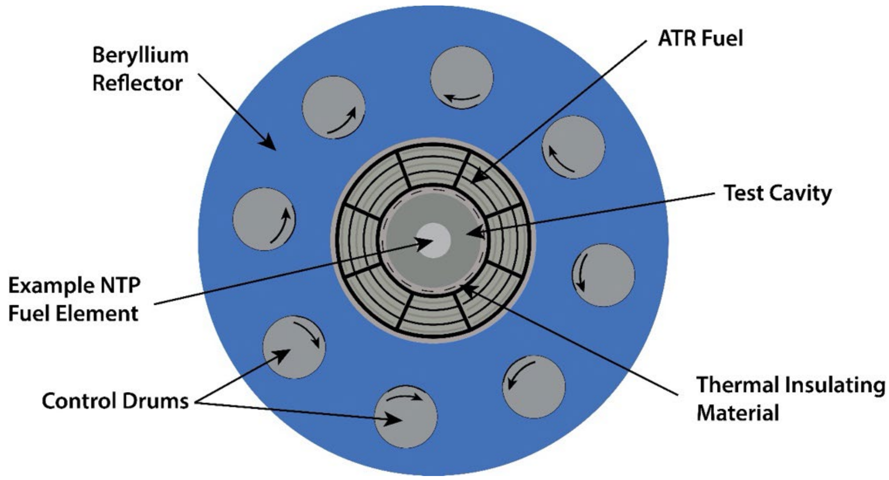


Figure 14. Conceptual cross-section of a ATR-fuel driver core SMART facility.

The simplified MCNP model was two-dimensional and preserved the discrete structure of fuel cladding, fuel, and coolant gaps in an annular model of a conceptual driver core reactor. All the fuel was assumed to have the same composition (UAl from Plate 5 in the benchmark document) and did not contain boron, which was present in some of the original outer plates. The details of the ATR fuel elements dictated a test cavity with an inner ring of 6.83 cm and an annular driver core surrounded by a 20-cm beryllium reflector for a total OD of 68 cm, as shown in Figure 15.

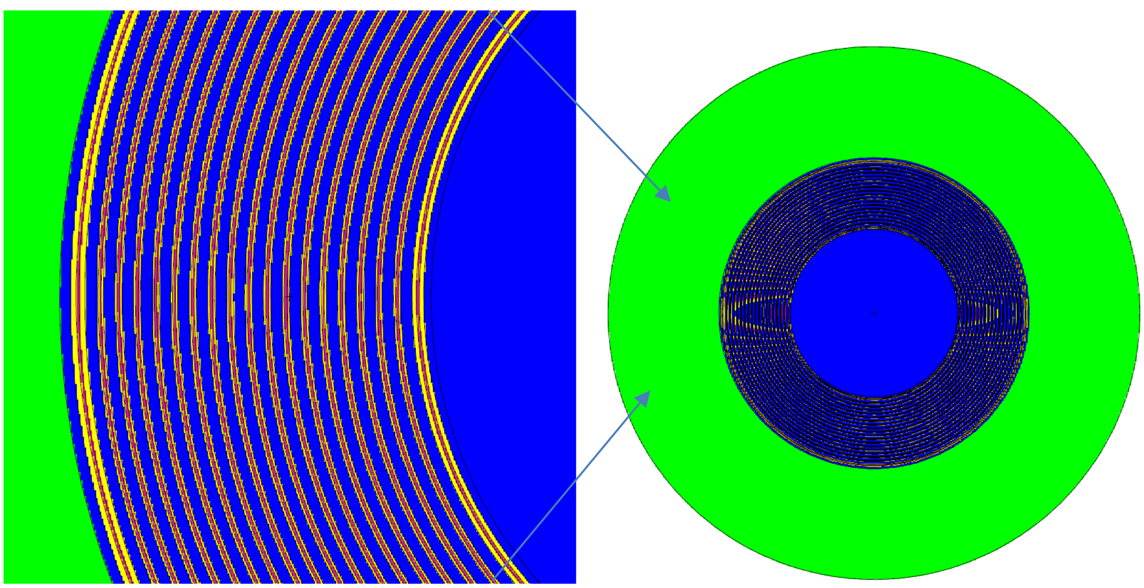


Figure 15. Cross-section of a SMART ATR driver core with a close-up of the fueled region.

The cross sections in the original model were then updated to ENDF/B-VII (except for some trace elements in the Al 6061), and a 0.635-cm thick Al 6061 baffle was added on the outer surface of the test cavity to separate it from the fuel plates.

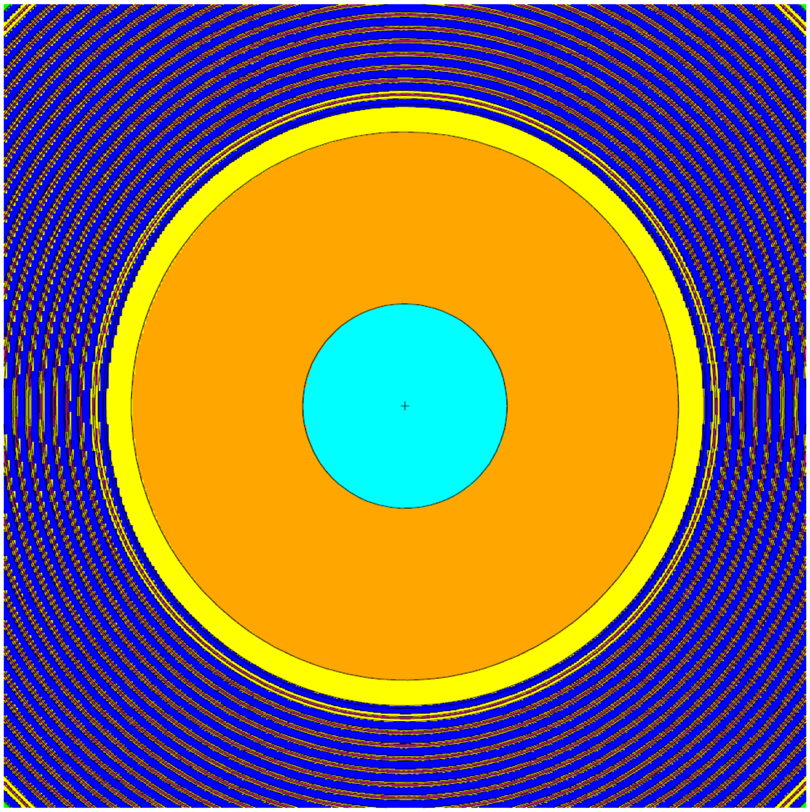
Three initial sets of calculations were performed, based on MCNP models supplied by Kelsa Palomares, William Searight, and Isabella Rieco for potential NTP test articles. The driver core was the same for all calculations, and the various candidate test articles were all contained in the test cavity (radius = 6.83 cm). The k-eff and the energy-deposition tallies for the driver core and the NTP fuels were calculated to obtain coupling factors.

6.2.1 SET 1: CERMET and CERCER NTP Test Articles

Homogenized HALEU CERMET and ceramic-ceramic (CERCER) NTP test articles were evaluated assuming both 93 wt% enriched and 20 wt% enriched fuel for the driver core based on the UAl fuel from the original ATR MCNP model (Table 7). Test article geometries were preserved to be representative of the geometries used in the current SNP testing reference design (TRD) [Gustafson 2021]. The test articles were represented by two zones, shown as the light blue and orange regions of Figure 16: 1) light blue regions represent the homogenized fuel, clad, and insulator and 2) orange represents the homogenized fuel assembly structural housing and moderator (r = 6.83 cm). To be consistent with the 2020 SNP TRD, two alternate configurations were surveyed—a CERMET fueled region in a zirconium hydride moderator block and a CERCER fuel in a zirconium hydride moderator block. The moderator was assumed to be an epsilon phase zirconium hydride, the SiC structural housing separating the fuel and moderator, and all moderator coolant channel passages were included in the homogenized material calculation. The CERMET fuel was modeled to have a MoW structural matrix and channel claddings with dispersed UN particles. The CERCER fuel was modeled to have a ZrC structural matrix and channel coatings with dispersed coated UN particles. Particle loading was assumed to be the average volume loading of the range specified for the TRD. All hydrogen coolant channels, and the porous zirconium carbide insulator, were included in the homogenized fuel models.

Table 7. ATR MCNP driver-core results from Set 1.

NTP Test Article	93-wt% Driver		20-wt% Driver	
	k-eff	coupling	k-eff	coupling
CERMET	1.368	0.0915	1.069	0.158
CERCER	1.404	0.0479	1.095	0.097



Legend:

Pale Blue: NTP Test Article – homogenized "fuel" (fuel & clad); r=2.55)
Orange: NTP Test Article – homogenized "moderator" (insulator & moderator); r=6.83
Yellow: Baffle – Al-6061; radii (inner/outer) = 6.83/7.6

Figure 16. MCNP model cross-section for Set 1.

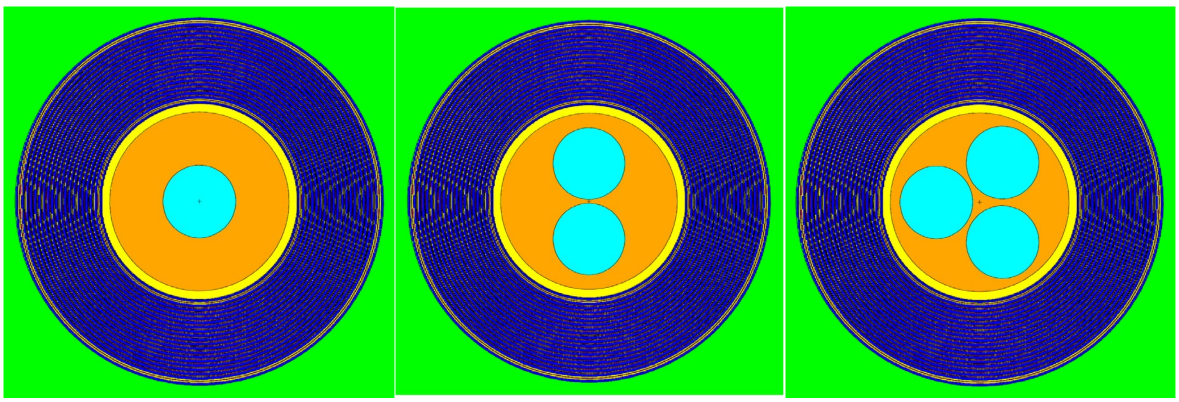
6.2.2 SET 2: Be_Block X NTP Test Articles

In addition to the SNP TRD fuel assembly cases, alternative HALEU NTP reactor configurations, representative of possible alternative designs, were also surveyed (Table 8). These analyses only considered 20 wt% enriched fuel for the driver core. For the first set of alternate reactor configuration cases, a CERCER fuel form in a beryllium moderator block was surveyed. These test articles differed from the testing reference design case due to the change in the moderator element material and overall geometries of the test article. In this case, the reference reactor design called for a reduced fuel element diameter, which allowed for multiple fuel elements to fit within the ATR-SMART test zone. For each case—one fuel element, two fuel elements, and three fuel elements—the material for the fuel and moderator remains constant, and the minimum pitch between fuel elements is equal to or greater than the minimum pitch defined in the TRD.

Similar to the SNP TRD case, the test articles were represented by two zones shown as the light blue and orange regions of Figure 17. Light blue regions represent the homogenized fuel, clad, and insulator, and orange regions represent the homogenized fuel assembly structural housing and moderator. The moderator was assumed to be pure Be metal (i.e., no impurities representative of a specific Be alloy were included) with a SiC structural housing, all coolant channel passages were included in the homogenized material calculation. The CERCER fuel was modeled to have a ZrC structural matrix with dispersed coated UN particles. Particle loading was assumed to be the average volume loading of the range specified for the reference reactor design. All hydrogen coolant channels and the porous zirconium carbide insulator were included in the homogenized Region 1.

Table 8. ATR driver core MCNP results from Test Article Set 2.

NTP Test Article	20-wt% Driver	
	k-eff	coupling
1 test article	1.097	5.88E-04
2 test articles	1.090	1.07E-03
3 test articles	1.082	1.51E-03



Legend:

Pale Blue: NTP Test Articles – Fuel

Orange: NTP Test Article – Reflector/Moderator; r=6.83

Yellow: Baffle – Al-6061; radii (inner/outer) = 6.83/7.46

Figure 17. MCNP model cross-sections for Set 2, from r-l: one, two, and three test articles.

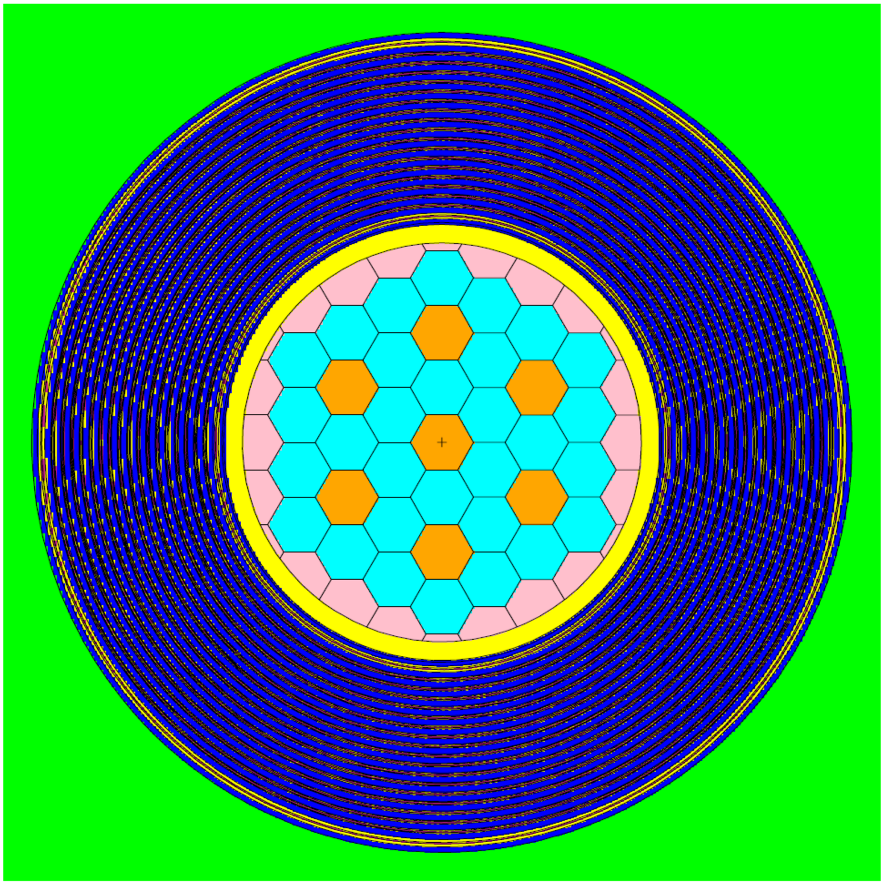
6.2.3 SET 3: NERVA NTP Test Articles

In the last set of alternate reactor configuration cases, HALEU fuel forms based on legacy NERVA/Rover designs were surveyed (Table 9). These test articles significantly differed from the TRD case due to a change in fuel element material, moderator incorporated into a tie-tube element instead of a moderator block geometry, and overall shift in geometry from cylindrical to hexagonal elements (Figure 18). This set also only considered 20 wt% enriched fuel for the driver core.

Consistent with the NERVA / Rover design, the fuel element was modeled to be a graphite matrix element with ZrC claddings and dispersed coated UC_2 and UN fuel particles. The homogenized MCNP fuel cells (light blue) were assumed to be the same geometry as the NERVA small nuclear rocket engine (SNRE) fuel element and included all relevant claddings and coatings [Durham 1972]. The tie-tube moderator element geometries and materials were consistent with those defined in the SNRE preliminary design and were also modeled as a homogeneous region (orange). Fuel and moderator elements were arranged within the available ATR-SMART core volume in a 3:1 fuel element to moderator element ratio and with unfueled graphite elements (outer ring, pink elements) lining the periphery, as shown in Table 9 and Figure 18.

Table 9. ATR driver core MCNP results for Test Article Set 3.

NTP Test Article	20-wt% Driver	
	k-eff	coupling
NERVA-UC2	1.140	0.267
NERVA-UN	1.131	0.267



Legend:

Pale Blue: NTP Test Articles – Fuel

Orange: NTP Test Article – ZrH_x Moderator;

Pink: Unfueled graphite r=6.83

Yellow: Baffle – Al-6061; radii (inner/outer) = 6.83/7.46

Figure 18. MCNP model cross-section for Set 3.

6.2.4 Follow-on Studies

The team had several discussions related to what enrichment should be considered as the driver for subsequent analyses, recognizing that the conversion of ATR fuel to HALEU was likely beyond the time desired for a SMART test facility. Therefore, it was decided that subsequent analyses only consider the current 93wt% enriched fuel for the driver.

To date, only a limited number of additional calculations have been performed beyond the three sets described above:

- Sensitivity to nuclear data
- Considering the effect of elevated temperature for the fuel core of the test article
- Varying composition and/or dimensions of the reflector.

The sensitivity to nuclear data considered ENDF/B-VII vs. ENDF/B-VII.1 and discrete and continuous scattering kernels for bound nuclei (e.g., light water, Be); only minor impacts were observed.

To avoid generating temperature-dependent cross-sections for MCNP, the available temperature-dependent data in the distributed cross-section libraries were considered for these studies. The available data included temperatures of 0.1, 250, 293.6, 600, 900, 1200, and 2500 K.

Temperature-dependent data, along with an adjusted density to reflect different temperatures, were used for the fuel in a CERCER test article similar to that considered in Set 1 above. The results are shown in Table 10, for room temperature and 2500 K, and within the statistics are very close.

Table 10. ATR driver core MCNP results with temperature dependent data.

NTP Test Article	**93-wt% Driver**	
	K-eff	Coupling
CERCER 293.5K	1.402	0.0457
CERCER 2500K	1.400	0.0424

The final scoping calculations considered different reflector dimensions and/or materials with the CERCER test article described above. All calculations assumed room-temperature ENDF/B-VII.1 cross-sections and scattering kernels and the 93wt% enriched driver core.

Table 11. ATR driver core MCNP results with various reflectors.

Reflector Material/Thickness	**k-eff**
Be/20 cm	1.402
Be/10 cm	1.249
BeO/10 cm	1.261
Light water/15 cm	1.041
Heavy water/15 cm	1.163
SS-316/15 cm	1.113

It is important to recognize that the MCNP calculations for the ATR-based SMART configurations described above are only for a preliminary scoping assessment of the potential for using this option. More detailed evaluation of an ATR-based driver core concept for SMART is needed to assess its viability and desirability for satisfying programmatic requirements. However, the results of these initial scoping analyses suggest that from the neutronics and criticality perspective, various NTP test articles could be successfully accommodated. Based on the range of coupling factors and the design power of ATR—i.e., 250 MW and 40-fuel elements, with an 8-element driver operating at the average ATR power per assembly—up to ~12 MW could be achieved in the "homogenized fuel" depending on driver and test article characteristics. In addition, there is flexibility in the materials and thicknesses for the reflector with 93 w/t% driver fuel.

The cost of an ATR-based SMART facility has yet to be determined.

6.2.5 Advantages and Disadvantages of an ATR-fuel driver core SMART Facility Concept

Advantages for this new-facility option are that it:

- Is a dedicated test reactor with the flexibility to address all the SMART testing needs as well as alternative NTP concepts, such as
  - Flux trap and test cavity that can be sized to accommodate full scale fuel elements or clusters of fuel elements (for some designs).
  - Driver fuel that could enable prototypic or near prototypic fuel element power densities, fluxes, and temperature distributions
- Is based on existing, demonstrated ATR fuel; this should expedite design, safety analysis, licensing and cost estimating
  - In the event that ATR fuel is converted from HEU to HALEU prior to the design of a SMART facility, the scoping analyses that consider a 20wt% enriched version of ATR fuel indicate that such a driver core would be feasible for SMART testing.
- Employs driver core concepts that have been studied earlier to support NTP fuel testing
- Could be used to test and prototype effluent cleanup systems and processes before being installed in an NTP demonstration test facility.

The main disadvantage is that, at this point, this is only a paper reactor, supported by limited scoping analyses. Time and resources would be required for a detailed assessment of the viability/desirability of the concept before proceeding. A detailed design is needed to address the safety, licensing, and cost of the concept. Additionally, a new facility:

- Will be the most expensive option.
- Will have a flux trap/test cavity OD that is limited and constrained to the curvature of the existing ATR fuel design. This limits the number of fuel elements that could be tested at a time.

7. SUMMARY

To minimize risk prior to an engine-reactor ground tests, the proposed technology development program should aim to demonstrate fuel and moderator material performance under prototypic conditions, together with full-scale fuel element manufacture and assembly approaches, and gather nuclear data needed to fill knowledge gaps related to proposed reactor operations. Historic programs have shown that this can be achieved through either full-scale reactor ground testing or through incremental technology development activities that progress from subscale separate effects testing to full-scale combined effects testing under simulated prototypic conditions. In this latter approach, existing infrastructure can be used to perform separate effects (i.e., non-nuclear, hot hydrogen testing or in-reactor testing) of sub-scale and full-scale fuel elements. However, no existing facility is capable of simulating the combined non-nuclear and nuclear environment of the NTP. A new reactor facility that allows for testing of full-scale NTP fuel and moderator element components under prototypic conditions (combined nuclear and non-nuclear environment) and generation of nuclear data under a relevant nuclear environment (prototypic spectrum, power density, flux) has the potential to most significantly reduce technical risk prior to the engine-reactor ground test.

Several options could be implemented to establish an irradiation testing capability suitable for investigation of multiple NTP fuel element performance characteristics at prototypic conditions. The best solution is dependent on several factors, such as performance, cost, availability, schedule, and plans for future testing in the SNP Program.

We believe sufficient testing capabilities are currently available to meet small-sample testing for both nuclear and non-nuclear testing. However, nuclear test facilities do not exist that could fill the gap between initial small sample testing and testing at prototypic conditions needed for fuel elements or fuel segments. We see three pathways that could be followed to obtain NTP fuel qualification

1. Rely extensively on fuel model development supported by non-nuclear testing and specific separate effect small sample nuclear testing.
2. Construct specific nuclear test facilities in which fuel and moderator segments or elements are tested to prototypic conditions.
3. Construct an engine-reactor test facility in which the entire core and engine components can be tested starting out with cold flow and short startup transients building to full operation tests.

The three options to fill the current testing gap are discussed below. The highest risk for the program is associated with the first pathway. Investment into the second pathway will further reduce program risk as more data are obtained to support fuel qualification and performance under operating environments are characterized to support and validate modelling efforts. Investment into the third pathway would provide the most risk reduction for the program, but also requires the most/greatest investment.

An exact cost comparison of options is not discussed here. At best, the relative cost of options can be considered in light of the ability of a facility to obtain fuel and moderator performance data at prototypic temperatures and conditions, in conjunction with other goals of the SNP Program, including meeting a desired launch schedule and the advancement of TRLs.

7.1 Most Rapid Deployment

Use of existing government and university reactors and test facilities with limited upgrades would provide an option for the most rapid deployment for testing but would also require that the SNP Program assume the highest level of risk for the fuel, reactor, and engine due to parameters in Table 1 that would not be met. Testing would be limited to some separate effects testing (of temperature, pressure, power density) but no data would be available for demonstrating any combined or integrated effects for full scale fuel assemblies. There would be no new data to support core neutronics or reactivity worth measurements as a result of the moderator material or the effects of hydrogen on the control of the reactor, especially during the startup transient and achieving steady-state operation. The inability to test the fuel to design conditions (for either a fuel element or a fuel-moderator bundle) results in risk related to the manufacturability of the reactor and the integrity of the fuel. Without the ability to demonstrate that the manufacturing methods that can produce fuel elements in the desired prototypic geometries capable of performing under all design conditions proposed for the NTP system, there is a greater risk that the fuel will fail and not meet performance objectives prior to the engine-reactor demonstration. Lack of validated reactor physics models and control approaches results in increased risk in meeting performance goals or demonstrating desirable reactor response for planned steady-state and transient conditions. Because of this risk, for this option, fuel and reactor TRL would be limited to 3 or 4 prior to integrated engine-reactor demonstration. If facilities modifications are not capable of full-scale separate effects testing over the total range of prototypic hydrogen and power density for the fuel's proposed total lifetime (including thermal cycling effects) or range of nuclear transients (including start up and shutdown transient rates and number of transients), fuel and reactor TRL will be constrained to TRL 3. If facilities modifications are capable of full-scale separate effects testing but combined effects testing is not pursued, fuel and reactor TRL would be limited to TRL 4. This option would rely heavily on multiphysics modelling to extrapolate fuel and reactor component performance demonstrated in the laboratory to predict performance during operation. These models would not be validated until test data is available from the demonstration.

7.2 A More Robust Solution

A new facility is needed to meet all of the desirable parameters in Table 1. Of the two options considered here, construction of a new 14-MW TRIGA research reactor is a viable lower cost alternative than a SMART driver core, though the TRIGA facility would require more compromise in the test parameters. The core and facility can be built to handle most of the unique test environments and would be large enough to test both fuel bundles and fuel-moderator combinations with flowing hydrogen. Such a facility would also allow for additional insight into some important nuclear physics parameters. Nuclear data related to material temperature coefficients and worth would be able to be measured and post-irradiation examination may allow for a better understanding of expected fission product inventory and material activation which is important for source term analysis. Such a facility would also provide combined effects performance data that could be used for various scaling studies that could be applied to full reactor and engine designs. Another use of this facility would be to test and prototype various effluent cleanup systems and processes before being installed in an NTR demonstration test facility. This facility would also provide valuable information in understanding design requirements and operation of a hydrogen gas loop within a nuclear reactor. The facility could be licensed as a non-power reactor either by the NRC or by DOE.

The limitations of this facility primarily stem from its inability to match all desirable parameters for the entire prototypic range. This facility would not be able to test full length fuel and moderator elements and the flux level and power density would be less than desired. Startup data would also be limited. This option would require scaling analyses to predict the combined full-scale fuel and reactor component performance at a subsystem level. Since the full-scale, full range of combined effects parameters would not be achievable, TRL would be limited to TRL 4 since overall performance in all critical areas would not be demonstrated commensurate to TRL 5 requirements. This option would also rely heavily on multiphysics modelling to extrapolate fuel and reactor component performance demonstrated through SMART TRIGA testing to predict performance during operation. These models would not be validated until test data is available from the demonstration, but there would be reduced risk compared to the rapid deployment option in understanding combined effects and interelement interactions.

7.3 Best Comprehensive Solution

We believe that the best comprehensive solution is to consider a SMART facility configuration based on the use of existing fuel elements from the ATR, or reduced enrichment ATR fuel. Eight of these elements can be formed into a ring surrounding a test chamber, with a flux chamber diameter of approximately 9 cm and a length of approximately 100 cm. This would allow the testing of full-length fuel and moderator elements and combinations of prototypic scale assemblies to better understand both component level performance under combined effects conditions and any inter-element interactions that may impact reactor performance. Prototypic full-length elements could be tested under the full range of temperatures, pressures, hydrogen flow rates, and power densities, for the total projected lifetime, or beyond, to allow for safety margin testing. This concept would also allow for prototypic startup and shutdown testing of reactor components and fuel. This configuration would provide the closest integrated test capability to meet NTP design parameters of power densities, neutron flux, chamber pressures, temperature gradients and hydrogen flow rates. This facility could be used to test and prototype various effluent cleanup systems and processes before being installed in an NTP demonstration test facility. It would also provide valuable information toward understanding design requirements and operation of a hydrogen gas loop within a nuclear reactor.

This facility would be housed at a DOE site and licensed by DOE - using existing fuel elements that are already certified for use simplifies the licensing of a new device with that fuel. This option would allow for the most risk reduction prior to a ground or flight demonstration. Due to the ability to demonstrate fuel and reactor core material performance under representative combined effects conditions at the full scale, successful completion of the testing for this option would result in a TRL of 5 for these components. This option would provide testing data needed to validate fuel and reactor component multiphysics models used to predict reactor core component performance during the engine-reactor demonstration.

Cost estimates to create a new facility based on a TRIGA design or on an ATR-fueled driver core were not part of this study.

7.4 Recommendations

NASA is currently considering either a flight demonstration or a ground demonstration to achieve TRL 6 for an NTP engine. Our recommendations for a SMART facility to fill the testing gap, based on the previous options, are as follows:

- We believe that using advanced modeling in conjunction with existing nuclear test facilities alone will not provide sufficient risk reduction to the program to ensure that the reactor or fuel would meet mission performance requirements and may result in a system failure. High-performance modeling should be complemented by either of the following.
    - If the flight demonstration pathway is selected, we recommend that an ATR-fuel driver core reactor or equivalent be built. This option would provide more detailed essential performance data to the program with regard to fuel performance, core neutronics, thermal stability, mechanical stress, and hydrogen compatibility.
    - If the ground demonstration pathway is selected, we recommend that a 14-MW TRIGA reactor, modified with a hydrogen loop, be built. This option would provide sufficient fuel and moderator performance data to the program and allow for the design and fabrication of a ground demonstration engine to be built with an acceptable level of risk and confidence that it can meet performance goals.
- Future recommendations and planning will benefit from a comprehensive test strategy, which identifies testing objectives and requirements that are aligned with the expected results from the usage of test facilities, new or existing, and how those results can be used to identify reductions in risk and progress in TRLs.
    - Agreement on a set of TRL definitions for an engine-reactor system between NASA and DOE is needed to facilitate determination of progress in TRL advancement. (See Appendix A for more information).

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Appendix A
Considerations for NTP Reactor TRL Advancement

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Considerations for NTP Reactor TRL Advancement

NASA's SNP project is developing new reactor and fuel materials as a risk reduction activity under the SNP fuel and moderator development plan, in parallel with efforts by industry led design teams [US DOE Feb 2021]. As a part of the overall SNP strategy, the project aims to mature NTP reactor technologies to an appropriate technology readiness level (TRL) to support a fuel / reactor down select (assumed to be TRL 4 or higher) in the FY2025 timeframe prior to proceeding to an integrated engine-reactor demonstration in the FY2027 timeframe. This appendix contains a review of applicable TRL definitions to NTP reactor development and preliminary recommendations on testing conditions and important parameters to consider for NTP Reactor technology development planning for each TRL.

One of the primary challenges facing the project is identifying the range and combination of testing parameters appropriate for sufficiently improving the TRL of reactor technologies so that risk is minimized prior to demonstration of any subscale or full-scale engine-reactor system. This assessment is challenging because NTP reactors differ significantly from traditional, non-reactor in-space propulsion systems typically developed by NASA, and also differs significantly from traditional terrestrial power reactor systems tested under DOE authorization. Therefore, to support ongoing planning activities, TRL guidance from both NASA and DOE is evaluated to identify specific considerations for maturing the NTP reactor components for each readiness level. This appendix contains an overview of the NTP reactor subsystem technology considerations (including functional and performance requirements) relevant to technology maturation planning, NASA and DOE TRL definitions, and specific recommendations for NTP Reactor testing and demonstration needs for each readiness level.

A-1. NTP Reactor Technology Overview

When assessing the readiness level of a technology, the technology must be defined in context to the overall system and its functions identified. Typical NTP system key performance parameters (KPPs) and their relationship to the reactor design are listed below:

- **Thrust** – the forward force generated by the rocket which is used to accelerate the spacecraft. Thrust is dependent upon fluid inlet and exit conditions leaving the nozzle (temperature, pressure, and gas velocity) and is directly proportional to mass flow rate. Engine thrust most directly impacts the reactor by controlling the amount of propellant that must be heated by the reactor, thereby impacting the total thermal power to be generated by the reactor.

- **Specific Impulse** – a measure of the efficiency of the overall system, specific impulse corresponds to the amount of thrust generated per unit mass of propellant throughout the mission. Therefore, specific impulse is dependent upon the exit fluid conditions leaving the nozzle (temperature, pressure, gas velocity, and mass flow rate) and is maximized with increasing temperature and minimizing propellant molecular mass. To maximize specific impulse, reactor exit temperature is typically maximized and selection of a hydrogen propellant (reactor working propellant) is typically proposed.

- **Mass** – to meet requirements related to launch and overall spacecraft design, mission optimization, the overall mass of each of the NTP subsystems is constrained with a goal to minimize each subsystem's mass to enable reduced propellant requirements or increase the amount of spacecraft payload. Since total reactor thermal power, will be governed by specific impulse and thrust, the reactor design must enable criticality under proposed operating conditions (total thermal power, reactor exit conditions, and total mass flow rate) for a minimal mass size which meets design requirements. This constraint when paired with specific impulse and thrust, impacts overall reactor operating conditions by impacting maximum reactor power density which further impacts overall reactor power, temperature, and stress distributions for in-reactor components.

- **Total Lifetime and Number of Burns** – Performance of the system must be maintained for all desired operational modes and overall system lifetime. The reactor must be able to satisfy heat transfer to the propellant intermittently and for multiple re-starts to satisfy the mission trajectory. Reactor components need to be able to survive total lifetime at full power, thermal cycling needed due to multiple engine burns, and required startup and shutdown transient conditions. Total lifetime encompasses total lifetime at full operational power as well as total lifetime at idle or standby conditions. Material performance and reactor criticality under designed operating conditions are critical to ensuring this KPP can be met.

Within a nuclear thermal propulsion system, the reactor's primary function is to act as a heat exchanger, directly transferring the heat generated from fission to a propellant (reactor working fluid) which is heated to sufficiently high enough outlet temperatures to be expanded out a nozzle to provide thrust. In this manner, the NTP reactor is a critical technology element to ensure system performance is maintained over the entire mission (reactor lifetime). To ensure KPPs are met for the NTP system, the key functions of the reactor that must be considered in its design and operations include:

- Heat transfer to the propellant to meet desired engine-reactor interface conditions to close the power balance for all operating modes of the engine.
- Enable criticality and a controllable reactor response for all design conditions (including reactor transients, nominal full power, and design basis off nominal conditions). Reactor control is maintained by the engine-controller, the instrumentation, physical mechanisms, and reactor physics response must be reliable and predictable during operation.
- Maintain acceptable component performance (including acceptable thermodynamic / compositional stability and structural integrity) throughout lifetime for all design modes to ensure system safety and integrity throughout the mission.

While the reactor subsystem is a key component within the NTP system, it must be able to interact with other subsystems in a way that does not diminish the ability of any subsystem from meeting its intended functional or performance requirements. In addition to TRL, system readiness level (SRL) should also be considered when assessing overall readiness. System readiness level maturation requires subsystems to consider the impact of related subsystems which is the subject of technology maturation planning. Key reactor subsystem interfaces to consider in the NTP system include:

- **Engine-Reactor Interface**
    - Exchange of the propellant across the engine-reactor power balance enables the performance of the system (specific impulse and thrust) and may enhance the engine subsystem functionality by providing heat to the propellant to power the engine-turbopump. This interface is represented as state points (fluid temperature, pressure, and mass flow rate) in the engine-reactor power balance.
    - The engine controller will be responsible for interacting with the reactor control system and instrumentation to ensure the reactor physics response is appropriate for desired operating modes of the NTP system throughout the mission.
    - The engine-reactor physical interface, which is impacted by external reactor volumes and overall reactor mass, may transmit stresses, radiation, or heat from the reactor subsystem to the engine subsystem.
- **Reactor-Spacecraft Interface**

  - The primary reactor-spacecraft interface of concern to designers is the impact of reactor operation on the radiation fields (gamma and neutron) experienced by the crew and spacecraft components (including the cryogenic fluid management system and propellant tanks)
  - The spacecraft may indirectly impact the reactor due to its role in management and storage of the hydrogen propellant and payload. This may impact the state points at which the propellant initially enters the reactor or result in slight alterations to reactor operations including number of burns, total burn time, and rated power during each burn.
- *Note: other interfaces may exist but have not been explicitly identified or considered in this report.*

In order for the NTP reactor to perform its desired functions, technology maturation planning aims to adapt existing technologies as well as develop new and novel materials or components capable of enabling the performance goals governed by the reference NTP system. A key aspect of this verification will be to ensure acceptable function, performance, and reliability of the design under all prototypic operating environments. For the NTP reactor, the typical operating environment includes the following elements:

- **Hydrogen working fluid:** consider temperature, pressure, mass flow rate (gas velocity), and possible fluid dynamic response from engine pump operation.
- **Temperature:** consider fuel, moderator, and reactor component temperature profiles (radial and axial). Consider temperature profile and feedback due to reactor physics controlled parameters; match temperature and stress profiles.
- **Irradiation:** consider power density (and neutron flux), total fluence (or burnup), transient irradiation response of materials.
- **Lifetime:** consider thermal cycling, total lifetime at full power, total lifetime including low power or decay heating modes.

For all components that have not been previously matured for the specific NTP operating environments expected of the reference design, they must be developed and demonstrated under relevant conditions that are representative of underlying physical phenomena or technical risk areas. Specific areas of technical risk for the NTP reactor subsystem include:

- **Reactor Physics:** controls power profiles, reactor criticality, and reactor transient response
- **Thermal-Structural:** dependent on material properties, reactor power densities, and fluid flow conditions, controls maximum fuel and moderator temperature and interelement temperature and stress profiles
- **Fluid-Structural:** dependent on engine-reactor interface, controls reactor heat transfer and efficiency, may result in different vibrational modes of the reactor components during operation
- **Material Performance:** includes the high temperature stability, corrosive interactions, and material irradiation response, contributes to reliability modelling of components
- **Manufacturability:** readiness of fabrication processes and infrastructure

NTP reactor design requires modelling and analysis of many competing multi-physics phenomena. These models require validation if a pre-existing benchmark does not exist. Therefore, planned technology maturation tasks should be planned to allow for collection of appropriate test data needed for model validation.

A-2. Technology Readiness Level Definitions

A-2.1. DOE Technology Readiness Levels

The Department of Energy has released updated guidance on performing technology readiness assessments and technology maturation planning for advanced reactor systems. The DOE technology readiness level definitions reported in this report are transcribed from the Department of Energy's "Technological Assessment Guide", DOE G 413.3-4A (September 2011). The TRL definitions contained in this document are summarized in Table A-1. Compared to alternative TRL definitions, DOE definitions give more specific guidance useful for reactor systems development, which considers both waste processes and commissioning readiness.

A-2.2. NASA Technology Readiness Levels

The NASA TRL guidance is provided by (National Aeronautics and Space Administration, 2013). Table A-2 summarizes the NASA TRL definitions. NASA TRL definitions include specific guidance for the development of hardware and software systems. For an NTP reactor, hardware definitions are most applicable, however modelling and simulation maturity and integration with the testing program is considered in the recommendations provided in this assessment.

A-2.3. Other Applicable Definitions Proposed for Technology Readiness Levels

To support this assessment, a wide range of documentation and published literature from NASA, DOE, the Department of Defense [JANNAF 2019], and the U.S. Government Accountability Office were also considered. There are other aspects of readiness to consider, beyond TRL, when developing and assessing the maturity of the overall NTP system. Some additional readiness levels that were considered include:

- Manufacture readiness level – "defines current level of manufacture maturity, identifies maturity shortfalls and associated costs / risks, provides basis for manufacture maturation and risk management" [OSD Manufacturing Technology Program, 2011].
- Integration readiness level –" a metric to measure the integration maturity between two or more components. IRLs, in conjunction with TRLs, form the basis for the development of the System Readiness Level (SRL)." [U.S. Government Accountability Office, 2016].
- System readiness level – "[defines] a holistic picture of the readiness of a complex system of systems by characterizing the effect of technology and integration maturity on a systems engineering effort". SRL definitions recommended by GAO are included in [U.S. Government Accountability Office, 2016].
- Advancement degree of difficulty (AD^2) – an alternative method for evaluating system maturity which focuses on development difficulty rather than readiness. From the NASA System Engineering Handbook, AD^2 assessment is defined as "the process to develop an understanding of what is required to advance the level of system maturity" (National Aeronautics and Space Administration, 2016).
- Fuel Readiness Level – Defines readiness of a fuel form for use in a system. Fuel readiness assessment includes evaluation of fuel performance readiness and fuel manufacture readiness [Carmack et al. 2016]

Additional supplemental information considered in the recommendations in the following section included: "Technology Maturation Planning and Technology Roadmap Development Example Using the Technology and System Readiness Assessment Process", INL/L TD-19-54549 [Dixon 2019], which contains recommendations on assessing TRL and system readiness level using an "evidenced-based" response to specific questions included in Appendix A (section A-3 and A-4).

It is also noted that previous attempts to resolve NASA, DOE, and fuel TRL have been previously performed for the assessment of NTP reactor readiness [Analytical Mechanical Associates, Inc. 2019, Frerking and Beauchamp 2016]. The definitions previously proposed for assessing NTP reactor readiness have been included in Table A-3.

Table A-1. The Department of Energy definitions of Technology Readiness Levels (TRLs)

TRL	TRL Definition	Description	Relative Level of Technology Development
1	Basic principles observed and reported.	Lowest level of technology readiness. Scientific research begins to be translated into applied research and development (R&D). Examples might include paper studies of a technology's basic properties.	Basic Technology Research
2	Technology concept and/or application formulated.	Invention begins. Once basic principles are observed, practical applications can be invented. Applications are speculative, and there may be no proof or detailed analysis to support the assumptions. Examples are still limited to analytic studies.	Basic Technology Research
3	Analytical and experimental critical function and/or characteristic proof of concept.	Active research and development is initiated. This includes analytical studies and laboratory scale studies to physically validate the analytical predictions of separate elements of the technology. Examples include components that are not yet integrated or representative. Components may be tested with simulants.	Research to Prove Feasibility
4	Component and/or system validation in laboratory environment.	Basic technological components are integrated to establish that the pieces will work together. This is relatively "low fidelity" compared with the eventual system. Examples include integration of "ad hoc" hardware in a laboratory and testing with a range of simulants.	Technology Development
5	Laboratory scale, similar system validation in relevant environment.	The basic technological components are integrated so that the system configuration is similar to (matches) the final application in almost all respects. Examples include testing a high-fidelity system in a simulated environment and/or with a range of real waste and simulants.	Technology Development
6	Engineering/pilot-scale, similar (prototypical) system demonstrated in a relevant environment.	Representative engineering scale model or prototype system, which is well beyond the lab scale tested for TRL 5, is tested in a relevant environment. Represents a major step up in a technology's demonstrated readiness. Examples include testing a prototype with real waste and a range of simulants.	Technology Demonstration
7	Full-scale, similar (prototypical) system demonstrated in a relevant environment.	Prototype full-scale system. Represents a major step up from TRL 6, requiring demonstration of an actual system prototype in a relevant environment. Examples include testing the prototype in the field with a range of simulants and/or real waste and cold commissioning.	System Commissioning
8	Actual system completed and qualified through test and demonstration.	Technology has been proven to work in its final form and under expected conditions. In almost all cases, this TRL represents the end of true system development. Examples include developmental testing and evaluation of the system with real waste in hot commissioning.	System Commissioning
9	Actual system operated over the full range of expected conditions.	Actual operation of the technology in its final form, under the full range of operating conditions. Examples include using the actual system with the full range of wastes.	System Operations

Table A-2. NASA definitions of Technology Readiness Levels (TRLs)

TRL	TRL Description	Hardware Description	Software Description	Success Criteria
1	Basic principles observed and reported.	Scientific knowledge generated underpinning hardware technology concepts / applications.	Scientific knowledge generated underpinning basic properties of software architecture and mathematical formulation.	Peer reviewed documentation of research underlying the proposed concept/application.
2	Technology concept and/or application formulated.	Invention begins, practical application is identified but is speculative; no experimental proof or detailed analysis is available to support the conjecture.	Practical application is identified but is speculative; no experimental proof or detailed analysis is available to support the conjecture. Basic properties of algorithms, representations, and concepts defined. Basic principles coded. Experiments performed with synthetic data.	Documented description of the application / concept that addresses feasibility and benefit.
3	Analytical and experimental proof-of-concept of critical function and / or characteristics.	Research and development are initiated, including analytical and laboratory studies to validate predictions regarding the technology.	Development of limited functionality to validate critical properties and predictions using non-integrated software components.	Documented analytical / experimental results validating predictions of key parameters.
4	Component and/or breadboard validation in a laboratory environment.	A low fidelity system / component breadboard is built and operated to demonstrate basic functionality in a laboratory environment.	Key, functionally critical software components are integrated and functionally validated to establish interoperability and begin architecture development. Relevant environments defined and performance in the environment predicted.	Documented test performance demonstrating agreement with analytical predictions. Documented definition of potentially relevant environment.
5	Component and/or brassboard validated in a relevant environment.	A medium-fidelity component and/or brassboard, with realistic support elements, is built and operated for validation in a relevant environment so as to demonstrate overall performance in critical areas.	End-to-end software elements implemented and interfaced with existing systems / simulations conforming to target environment. End-to-end software system tested in relevant environment, meeting predicted performance. Operational environment performance predicted. Implementations.	Documented test performance demonstrating agreement with analytical predictions. Documented definition of scaling requirements. Performance predictions are made for subsequent development phases.
6	System / sub-system model or prototype demonstration in a relevant environment.	A high-fidelity prototype of the system / subsystems that adequately addresses all critical scaling issues is built and tested in a relevant environment to demonstrate performance under critical environmental conditions.	Prototype implementations of the software demonstrated on full-scale, realistic problems. Partially integrated with existing hardware / software systems. Limited documentation available. Engineering feasibility fully demonstrated.	Documented test performance demonstrating agreement with analytical predictions.

7	System prototype demonstration in an operational environment.	A high-fidelity prototype or engineering unit that adequately addresses all critical scaling issues is built and functions in the actual operational environment and platform (ground, airborne, or space).	Prototype software exists having all key functionality available for demonstration and test. Well integrated with operational hardware / software systems demonstrating operational feasibility. Most software bugs removed. Limited documentation available.	Documented test performance demonstrating agreement with analytical predictions.
8	Actual system completed and "flight qualified" through test and demonstration.	The final product in its final configuration is successfully demonstrated through test and analysis for its intended operational environment and platform (ground, airborne, or space). If necessary, life testing has been completed.	All software has been thoroughly debugged and fully integrated with all operational hardware and software systems. All user documentation, training documentation, and maintenance documentation completed. All functionality successfully demonstrated in simulated operational scenarios. Verification and Validation (V&V) completed.	Documented test performance verifying analytical predictions.
9	Actual system flight proven through successful mission operations.	The final product is successfully operated in an actual mission.	All software has been thoroughly debugged and fully integrated with all operational hardware and software systems. All documentation has been completed. Sustaining software support is in place. System has been successfully operated in the operational environment.	Documented mission operational results.

Table A-3. Previously Proposed Definitions for NTP Reactor Technology Readiness Levels (TRLs)

TRL	Engine (NASA)	Fuel (DOE)	Reactor (AMA)
1	Basic principles observed and reported	A new concept is proposed. Technical options for the concept are identified and relevant literature data reviewed. Criteria developed.	At this level everything is conceptual only. There is no historical test data for the operating regime of interest and no proven models. Concepts are solely based on the observed and studied principles behind the theoretical idea.
2	Technology concept and/or application formulated	Technical options are ranked. Performance range and fabrication process parametric ranges defined based on analyses.	At this level everything is conceptual only. Supporting information from previous studies and tests are utilized to develop designs and possible testing ideas.
3	Analytical and experimental critical function and/or characteristic proof-of-concept	Concepts are verified through laboratory-scale experiments and characterization. Fabrication process verified.	Initial development and testing occurs at this level. Testing is laboratory scale and typically consists of determining material properties and other key features of each component.
4	Component and/or breadboard validation in laboratory environment	Fabrication of samples using stockpile materials at bench-scale irradiation testing of small-samples in relevant environment. Design parameters and features established. Basic properties compiled.	Continued testing and development based on results acquired from TRL 3. At this point components of the system can be tested together or separate in simulated environments that could be similar to those experiences by the components to determine the functionality of the initial concepts.
5	Component and/or breadboard validation in relevant environment	Fabrication of fuel segments using prototypic feedstock materials at laboratory-scale. Irradiation testing at relevant environments. Primary performance parameters with representative compositions under normal operating conditions quantified. Fuel behavior models developed for use in fuel performance code(s).	At this level, full-scale components are being integrated to determine if they will function together properly.
6	System/subsystem model or prototype demonstration in a relevant environment (ground or space)	Fabrication of fuel segments / elements using prototypic feedstock materials at laboratory-scale and using prototypic fabrication processes. Irradiation testing of fuel at relevant and	This step can be noted as the step between "proof-of-principle and proof-of-performance" phases. At this step, the components are integrated into full-scale subsystems which are

		prototypic environment (steady-state and transient testing). Predictive fuel performance code(s) establishment.	tested in environments similar to those that will be experienced by the system.
7	System prototype demonstration in a space environment	Fabrication of test assemblies using prototypic feedstock materials at engineering-scale and using prototypic fabrication processes. Assembly-scale irradiation testing in prototypic environment. Predictive fuel performance code(s) validated.	At this level, it has been established that the components can be successfully fabricated and will function to meet given requirements.
8	Actual system completed and "flight qualified" through test and demonstration (ground or space)	Fabrication of a few core-loads of fuel and operation of a prototype reactor with such fuel.	The integrated system has been fabricated and tested successfully. Development and risk mitigation still play a role in this step as the system is not ready to be commercially fabricated and used.
9	Actual system "flight proven" through successful mission operations	Routine commercial-scale operations. Multiple reactors operating.	The integrated system can be routinely fabricated and utilized with low risk. There is now confidence in its capabilities.

A-3. Technology Readiness Level Tasks for Nuclear Thermal Propulsion Reactor Development

A-3.1. Technology Readiness Level 1

DOE guidance calls out TRL 1 as:

> *Lowest level of technology readiness. Scientific research begins to be translated into applied research and development (R&D). Examples might include paper studies of a technology's basic properties.*

While NASA guidance calls out TRL 1 as:

> *Hardware: Scientific knowledge generated underpinning hardware technology concepts / applications.*
>
> *Software: Scientific knowledge generated underpinning basic properties of software architecture and mathematical formulation.*

For the NTP reactor subsystems, a TRL 1 would correspond to:

- The overall system configuration is identified including propellant and mechanism for heating the propellant with the reactor.
- The trade space of technology and material candidates for the reactor subsystem are identified through literature review or equivalent survey.
- Research is limited to paper studies or observations.

A-3.2. Technology Readiness Level 2

DOE guidance calls out TRL 2 as:

> *Invention begins. Once basic principles are observed, practical applications can be invented. Applications are speculative, and there may be no proof or detailed analysis to support the assumptions. Examples are still limited to analytic studies.*

While NASA guidance calls out TRL 2 as:

> *Hardware: Invention begins. Practical application is identified but is speculative; no experimental proof or detailed analysis is available to support the conjecture.*
>
> *Software: Practical application is identified but is speculative; no experimental proof or detailed analysis is available to support the conjecture. Basic properties of algorithms, representations, and concepts defined. Basic principles coded. Experiments performed with synthetic data.*

For the NTP reactor subsystems, a TRL 2 would correspond to:

- Confirmatory benchtop studies or supporting literature data is gathered to confirm the heat transfer mechanism from the reactor to the propellant.
- The functions of the reactor and related physical parameters are identified. Modeling and simulation tools are identified and / or adapted for analyzing the technology.
- A reference mission model is developed and a range of top-level performance and functional requirements of the system identified for a range of possible reference missions.

- The design space of the reactor subsystem is identified based on system level performance and functional requirements. Reactor subsystem technology options are identified at a conceptual design level. For each option, critical components are identified and design begins including identifying material compositions, fabrication readiness, component functional requirements, and range of operating environments.
- Candidate manufacture technologies are identified, and initial fabrication studies may begin using surrogate materials.

System readiness level considerations for TRL 2 include:

- Have the reactor subsystem interfaces been defined and how do they impact the reactor subsystem?
- Does integration of the reactor within the overall system impact any of the functions of the reactor or its components? The operating environment of the reactor or its components?
- Are there any breakpoints in the reactor technology design space with respect to performance or functional requirement ranges? Can all technology options enable reference missions?

A-3.3. Technology Readiness Level 3

DOE guidance calls out TRL 3 as:

> *Active research and development is initiated. This includes analytical studies and laboratory scale studies to physically validate the analytical predictions of separate elements of the technology. Examples include components that are not yet integrated or representative. Components may be tested with simulants.*

While NASA guidance calls out TRL 3 as:

> *Hardware: Analytical studies place the technology in an appropriate context; laboratory demonstrations, modeling, and simulation validate analytical prediction.*
>
> *Software: Development of limited functionality to validate critical properties and predictions using non-integrated software components.*

For the NTP reactor subsystems, a TRL 3 would correspond to:

- Fabrication technology development begins for critical reactor components. Subscale fabrication demonstration is completed with representative feedstocks and fabrication technologies.
    - Increased risk reduction is achieved if: fabrication process scalability is explored with surrogate materials (if applicable).
- Material property measurements are initiated with representative, as-fabricated material coupons under relevant temperature ranges.
- Non-nuclear testing (separate effects testing) to assess chemical compatibility and thermodynamic stability under relevant conditions (subscale). Testing informs component level designs and reactor modelling activities to confirm target KPPs can be met or refine KPPs.
- Initial irradiation testing (transient and steady state) under representative transients, temperatures, power densities/fluxes, and fluences (subscale) is completed on critical reactor component materials.
- Nuclear data is generated through cross section measurements or integral critical experiments if data gaps or appropriate benchmark does not exist
- Component level quality control techniques are proposed.

- Reactor component level finite element analysis (FEA) and multiphysics modeling commences. Applicable modelling and simulation (M&S) tools are adapted for the technology. Data gaps are identified for underlying property / performance databases.
- Mission models are refined and an integrated system model is developed. Pre-conceptual level trade studies are performed to down select specific reactor technologies and refine the NTP engine-reactor configuration (power balance). Target KPPs are established for the system and each of the subsystem technologies.

System readiness level considerations for TRL 3 include:

- Can the reactor and component level designs under consideration satisfy the proposed target KPPs of the system?
- Do best available models and test data predict that the reactor is capable of satisfying the proposed engine-reactor power balance? How would changes to the engine side of the power balance impact the reactor and range of operating conditions for the components?

A-3.4. Technology Readiness Level 4

DOE guidance calls out TRL 4 as:

> *Basic technological components are integrated to establish that the pieces will work together. This is relatively "low fidelity" compared with the eventual system. Examples include integration of "ad hoc" hardware in a laboratory and testing with a range of simulants.*

While NASA guidance calls out TRL 4 as:

> *Hardware: A low-fidelity system / component breadboard is built and operated to demonstrate basic functionality and critical test environments, and associated performance predictions are defined relative to the final operating environment.*
>
> *Software: Key, functionally critical software components are integrated and functionally validated to establish interoperability and begin architecture development. Relevant environments defined and performance in the environment predicted.*

TRL 4 completes all reactor component level development activities. For the NTP reactor subsystems, a TRL 4 would correspond to:

- Fabrication technologies are down selected for critical reactor components. Component fabrication demonstration with representative feedstocks. Assembly and joining techniques required for assembly fabrication are assessed using surrogate materials or components (such as a fabrication demonstration of a mockup reactor core unit cell).
- Confirmatory material property measurement of representative as-fabricated material coupons is performed. Assessment of fabrication process repeatability is completed to understand impact to material properties or component reliability. Subcomponent level quality control techniques are validated. Component level quality control techniques are validated.
- Non-nuclear testing (separate effects testing) to assess chemical compatibility and thermodynamic stability under relevant conditions (subscale). Testing informs component level designs and reactor modelling activities to confirm target KPPs can be met or refine KPPs.

- Irradiation testing (transient and steady state) under representative peak nominal transients, temperatures, power densities/fluxes, and fluences (subscale) is completed on critical reactor component materials. Post irradiation examination for materials characterization is completed following irradiation (including gamma spectroscopy, i.e. fission product inventory and activation measurement, as well as material property measurements).
- Non-nuclear "prototypic" subscale mock-up structural and flow testing of proposed components is completed using cold or hot flow testing.
- Zero power critical (mockup reactor) testing completed to benchmark nuclear physics codes.
- Reactor component level finite element analysis (FEA) and multiphysics models are refined. Modeling activities establish component requirements and identify the range of operating parameters to explore through testing.
    - Note: M&S activities should be used to inform TRL 5 – 6 technology development task planning to gather the necessary test data to validate system, subsystem, or component models and reduce highest risk area / technology gaps.
- Mission models and the integrated system model are further refined. Reactor modelling demonstrates KPPs can be met.

System readiness level considerations for TRL 4 include:

- Do validated models and preliminary testing activities indicate the reactor is capable of satisfying the proposed engine-reactor power balance and mission KPPs? Do modelling activities require testing results to be extrapolated in order to meet KPPs? If so, is the extrapolation approach purely empirical or physics-based and does further testing need to be completed to confirm trends?
- How would changes to the engine side of the power balance impact the reactor and range of operating conditions for the components? Will any component level limits that have been identified be violated due to changes in the engine-reactor power balance?
- Is there any updated test data that indicates engine component technologies would impact the operating conditions of reactor components compared to what was demonstrated in the laboratory?

A-3.5. Technology Readiness Level 5

DOE guidance calls out TRL 5 as:

> *The basic technological components are integrated so that the system configuration is similar to (matches) the final application in almost all respects. Examples include testing a high-fidelity system in a simulated environment and/or with a range of real waste and simulants.*

While NASA guidance calls out TRL 5 as:

> *Hardware: A medium-fidelity system / component brassboard is built and operated to demonstrate overall performance in a simulated operational environment with realistic support elements that demonstrates overall performance in critical areas. Performance predictions are made for subsequent development phases.*

Software: End-to-end software elements implemented and interfaced with existing systems / simulations conforming to target environment. End-to-end software system tested in relevant environment and meeting predicted performance. Operational environment performance predicted. Prototype implementations developed.

TRL 5 completes all reactor subsystem level development activities including: lifetime, combined effects reactor unit cell demonstration (roughly equivalent to fuel qualification) and zero to low power critical demonstration (representative reactor, can be subscale, with prototypic working fluid). For the NTP reactor subsystems, a TRL 5 would correspond to:

- Engineering scale fabrication demonstration with prototypic feedstocks and representative unit cell assembly. Laboratory equipment may be used in the fabrication of components to a scale nearing that of the prototypic use case.
  - Note: Full risk reduction may be achieved if reactor component fabrication is demonstrated on the production scale commensurate to that required for fabricating a test reactor (pilot line demonstration).
- Non-nuclear testing (separate effects testing) to assess chemical compatibility and thermodynamic stability under relevant conditions (engineering scale components). Testing informs component level designs and reactor modelling activities to confirm target KPPs can be met or refine KPPs.
- Non-nuclear engineering scale structural and flow testing of proposed integrated reactor design (may be non-prototypic pump and heat exchanger substitute for engine interface). This task may correspond to cold or hot flow testing of a mockup (unfueled) reactor.
- Component and assembly level vibrations and loads testing is completed.
- A statistical material property database is completed and baselined for demonstration reactor design activities.
- Combined effects prototypic irradiation testing of engineering scale components (representative unit cell) under the full range temperature, flux, working fluid interface, and spectrum expected for the reactor. Post irradiation examination confirms acceptable material performance. Quality assurance / control techniques for material manufacture are established based on test program data. A fuel specification for the demonstration reactor is finalized. Component level quality control techniques are validated.
  - Note: full risk reduction may be achieved if full scale reactor components are tested under prototypic, combined effects conditions with margin exceeding that expected for the operational use case.
- Reactor component level finite element analysis (FEA) and multiphysics models are refined. Technology development tasks are used to validate system, subsystem, or component models and reduce highest risk area / technology gaps. Component performance and functional requirements are established for the demonstration system.
- Mission models and the integrated system model are further refined. Reactor testing activities confirm KPPs can be met.

System readiness level considerations for TRL 5 include:

- Is the reactor capable of satisfying the proposed engine-reactor power balance and mission KPPs? Was reactor performance observed to change due to combined effects testing and are these changes accurately captured in reactor models? Does further testing need to be completed to confirm or resolve underlying physical phenomena?

- Were reactor tests performed at the appropriate scale? How does scaling impact the reactor or reactor component performance, operating environments, or requirements? How is scaling predicted to impact interfaces of the reactor with different subsystems?
- Is there any updated test data that indicates engine component technologies would impact the operating conditions of reactor components compared to what was demonstrated?

A-3.6. Technology Readiness Level 6

DOE guidance calls out TRL 6 as:

> *Representative engineering scale model or prototype system, which is well beyond the lab scale tested for TRL 5, is tested in a relevant environment. Represents a major step up in a technology's demonstrated readiness. Examples include testing a prototype with real waste and a range of simulants.*

While NASA guidance calls out TRL 6 as:

> *Hardware: A high-fidelity system / component prototype that adequately addresses all critical scaling issues is built and operated in a relevant environment to demonstrate operations under critical environmental conditions.*
>
> *Software: Prototype implementations of the software demonstrated on full-scale, realistic problems. Partially integrated with existing hardware / software systems. Limited documentation available. Engineering feasibility fully demonstrated.*

TRL 6 is completed with a successful engine-reactor demonstration. For the NTP reactor subsystems, TRL 6 testing would require the following tasks:

- Prototypic scale reactor hot or cold flow testing
- Prototypic scale reactor vibration and mechanisms (thermal vacuum) testing
- Prototypic scale reactor zero power critical testing

System readiness level considerations for TRL 6 include:

- How was reactor performance impacted by the engine interface?
- Was the demonstration test performed at the appropriate scale? How does scaling impact the reactor and integrated engine-reactor performance, operating environments, or requirements? How is scaling predicted to impact interfaces of the reactor or integrated engine-reactor with the other full scale NTP subsystems?

SRL 6 is completed with a successful engine-reactor demonstration. For the NTP reactor subsystems, SRL 6 testing would require the following tasks:

- Engine-reactor manufacture and assembly. Reactor component fabrication is demonstrated on the production scale commensurate to that required for fabricating a prototypical scale NTP reactor (pilot line demonstration).
- Engine-reactor zero power critical testing
- Engine-reactor electromagnetic (EM) field testing
- Engine-reactor full power*, prototypic flux testing.

    - *Note: engineering scaled testing is acceptable if the test data is shown to be extensible to the operational use case with validated models and analysis. Full power refers to full nominal power of the prototype engine-reactor system. This prototype may be subscale compared to the operational use case. Example: subscale engine-reactor system with all major scaling parameters matched. All of the critical relevant physics phenomena and operating conditions identified in the introductory portion of this presentation should be demonstrated within the same regime expected of the operational system.

Some complementary activities to reduce risk related to engine-reactor scalability would include:

- Engine-reactor hot or cold flow testing
- Engine-reactor reactor vibration and mechanisms (thermal vacuum) testing

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Appendix B
Historical Programs

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B-1. ROVER, NERVA AND THE NUCLEAR FURNACE (NF-1) 1955–1972

The NERVA program was the most extensive effort thus far to develop an NTP rocket engine. [Finseth 1991, Atomic Power 2015] This program started in 1955 to develop a nuclear engine for either space or terrestrial applications. The program was started under the Department of Defense and later moved to NASA, jointly with what was then the AEC. Most experiments took place at the Nevada Test Site, now known as the Nevada National Security Site (NNSS). Several successful ground-based reactor tests were performed under the program as cost estimates for a full flight system escalated.

The NERVA/Rover program consisted of three ground test reactor campaigns, integrated engine-reactor testing, and nuclear furnace testing. Three ground test reactor campaigns: Kiwi, Phoebus, and Pewee, were undertaken to understand the reactor physics and component level performance of an NTP reactor using a graphite-matrix fuel element with dispersed uranium fuel particles (initial reactors used UO_2 particles, but eventually evolved to pyrolytic carbob [PyC]-coated UC_2 kernels) at various reactor power and thrust scales. Kiwi reactors tested NTP reactor designs over a range of 70–990 MW_{th} (~5–75 lbf) and demonstrated heating of the hydrogen propellant to exit temperatures of greater than 2361 K (>825 s Isp). The Phoebus reactors demonstrated controllable operation of larger scale reactors (3500–4082 MW_{th}) at similar operating temperatures (2444 K exit gas temperatures) while Pewee demonstrated a small scale (500 MW_{th}) reactor under higher performance conditions (2750 K exit temperature) with a two-pass, tie-tube, moderated reactor core. Integrated engine-reactor tests for NERVA began with the NRX series, based on the Kiwi B4E, in 1964 at the NNSS. The NRX-A6 was the ultimate model of this series, reaching the highest power (1120 MW_{th}) and operating time (62 minutes) after incremental improvements in operation and control of the systems before it. These successes led to the development of experimental engine (XE) cold-flow (CF) and XE-Prime tests, the most prototypic NTP engine systems developed for the program. The XE-Prime tests achieved a thrust of 55,000 lbf, with 28 engine restarts.

The program also planned a reactor-in-flight test (RIFT) campaign to develop a NERVA-powered vehicle as an upper stage for a Saturn-class launch vehicle, which aimed to culminate in an in-space engine-reactor demonstration. However, the ejection of fuel elements from the reactor core during Kiwi B4A tests led to reassessment of the program and cancellation of RIFT in December 1963.

In the final stages of the Rover Program, efforts to improve the reactor fuel elements were continued through the Nuclear Furnace Program. The nuclear furnace (NF-1) was meant to be reused, but instead was only used once before the program was cancelled in 1972. NF-1 consisted of two parts: a permanent, reusable portion that included the reflector and external structure and a temporary, removable portion that consisted of the core assembly and associated components. This reusable test device was intended to reduce both the time between reactor tests and the cost of testing. After completion of a test series the core assembly would be removed and disassembled for examination, and the permanent structure would be retained for use with a new core. [Kirk 1973]

NF-1 was used to evaluate advanced fuel elements that contained ZrC coated, (U, Zr)C graphite fuel. [Lyon 1973] The NF-1 fuel elements were tested for the longest time at full power, achieving a fuel temperature of 2450 K for 108.8 minutes.

A unique feature of these tests—of significance to future testing under SNP—is the use of an effluent cleanup system downstream of the reactor to remove fission products from the reactor effluent before release of the cleaned gas.

The NERVA program demonstrated reactors and fuels for NTP engines at a TRL above anything else ever attempted. While plans for an in-flight test reactor were never realized, the NRX and XE series tests demonstrated engine-reactor operations and ground testing of reactor designs that allowed for definitive assessment of reactor component performance to meet requirements. While the program was broadly successful, a program of its budget size is unlikely to materialize again, which means a qualification schedule of smaller, evolutionary tests from subscale assemblies to a fully integrated engine would allow a modern NTP design to proceed with experimental validation at a significantly lower cost compared to the NERVA series.

B-2. PLUTO PROGRAM 1957–1964

The Pluto Program was led by the Lawrence Radiation Laboratory, now the Lawrence Livermore National Laboratory. [Reynolds 1961] The Pluto Program's goal was to produce a reactor (the Tory series) that could power a ramjet cruise missile capable of flying at Mach 2.8 at 10,000 foot above sea level. The ramjet is a jet engine capable of supersonic air intake, with a compressor that slows air to subsonic speeds before receiving heat and being accelerated back to supersonic speeds through the nozzle. Similar to NERVA/Rover, the Pluto Program ground tested nuclear reactors to demonstrate subsystem functionality and component survivability. The main reactors built and tested during the program were the Tory IIA and the Tory IIC. The Tory IIC reactor completed three successful tests in 1964 at the Nevada Test Site (now the Nevada National Security Site, NNSS), at low, intermediate, and full-power (Tests 87–89) just after the program was officially cancelled.

The Low Power Test Campaign, (Run 87, Tory IIC), served as a final check of the nuclear control system under high air flow and provided data on the reactor's temperature coefficient of reactivity. In this test, reactor power was increased up to 1 kW, and reactor temperature was then modified solely by inlet air temperature. As airflow ramped up to 1800 lb/s (pps), position signals from actuators became noisy due to electrical shorting, but upon reset, airflow was maintained for 1 minute, and then reduced to 200 pps and held for 10 minutes as the inlet air temperature declined. Run 88, the intermediate power test, sought to simulate steady-state reactor operation at Mach 2.8, 10,000 feet above sea level. The reactor power was raised to 80 kW, held for 40 minutes to adjust exhaust chambers for coverage during high power operation, and airflow was brought up to 1260 pps. The airflow was held at 440 pps for a while to verify predicted reactor power, temperatures and flow rates from thermocouples and actuators. After reaching 1260 pps, reactor power was increased to provide a measured 1132°C in the core, and this flow rate was maintained for several minutes before air reserves were depleted. Run 89, the design power test, was intended to simulate realistic reactor operating conditions, or prototypic conditions, at a steady-state Mach 2.8 at sea level. Similar to the intermediate power test, reactor power was raised to 700 kW and held to adjust power control, then airflow was raised from 410 pps and the reactor power with it to achieve predicted values of 1660 pps and a core temperature of 1293°C. Although the airflow rate was inconsistent due to heating system limitations, this high-power state was maintained for 5 minutes as planned, and then the test shut down as normal with low pressure blowers cooling the air and the reactor. The high-power test concluded with no loss of reactivity detected, and no difficulties in reactor operation. Planned tests before program cancellation included a fast, 1 minute engine startup from 1 kW to high power, and a fast startup with a constant shim-rod removal, demonstrating control of the reactor using aerodynamic measured parameters like airspeed instead of reactor parameters like temperature. These final and planned tests indicated that ground testing of a reactor is critical in determining the feasibility of the integrated system working as intended.

B-3. GE-710 GAS REACTOR PROGRAM 1962–1968

The GE-710 high temperature gas cooled reactor program focused on developing a fast-spectrum refractory metal reactor with hexagonal CERMET fuel elements for submarine, aircraft and rocket propulsion applications. [710 Report Vols. I and III 1967] The reactor was designed to operate with a closed loop configuration (neon coolant, 1963-1966) and an open loop configuration (hydrogen propellant, 1962–1963). The 710 program was shifted to focus on space propulsion fuel elements in 1966, with detailed studies performed on CERMET fuel form fabrication, testing, and qualification until its discontinuation in 1968. The qualification program consisted of non-nuclear testing in static and dynamic test loops, in-pile reactor testing, and critical testing.

The 710 testing program consisted of three major objectives. The first objective (May 1962–October 1963) aimed to demonstrate a small, high-performance fast-spectrum NTP reactor. Experimental static closed loop hydrogen tests culminated in the demonstration of a Ta-clad W-60 vol% UO_2 sample for 10 hours at 2600°C, and a W-25 atom% Re sample for at 2600°C for 50 hours. For the program's second objective (October 1963 - July 1965), a full power demonstration of fuel elements with the closed loop, partial core length (full length approx. 1/3 of a meter), 91 and 37 coolant channel, Ta-clad fuel elements and a single W-Re-Mo-clad fuel element were fabricated and tested at 2150°C for 100 hours and 30 cycles. In both test campaigns, CERMETs exhibited excellent dimensional stability, thereby satisfying initial testing goals. This success led to a series of mockup critical experiments of the 710 reactor in 1965.

For the 710 program's third objective (July 1965–October 1966), fuel element performance was evaluated for a 200 kWe Brayton cycle space power reactor. Fuel qualification test program goals were to demonstrate fuel performance at an operating temperature of 1650°C for 10,000 hours. Failure mode analysis was conducted via non-nuclear testing static and dynamic tests of single and multiple subscale fuel elements, as well as static in-pile tests of single partial-length elements. A long-term demonstrator plan was drawn up for the AEC, with a reference design testing by 1971, reactor demonstration in 1975, and service, possibly starting in 1980. The need to test 710 fuel elements in a fast-flux environment was identified, with a fast-flux filter determined to be feasible for static in-pile tests using either the Engineering Test Reactor at the National Reactor Test Station (NRTS, now Idaho National Laboratory) or the Oak Ridge Research Reactor. This plan indicated that both in-pile and out-of-pile testing needed to be complete by 1969. To this end, T-111-clad partial length fuel elements were fabricated and tested for 1000 hours in August 1966, and a dynamic hydrogen loop for full-size elements was made operational and a full-size T-111-clad element was tested in the same timeframe. Two static in-pile tests at Oak Ridge National Laboratory's Low Intensity Test Reactor were designed and conducted in June and August of 1966. Dynamic in-pile tests would have been the next phase of testing. Design of a pilot loop for in-pile dynamic tests at the NRTS was completed but cancelled in FY 1967. Experimental results indicated that fuel qualification was on schedule for June 1969 completion.

The major accomplishment of the GE-710 Program was in its fabricating prototypic geometry CERMET fuel elements and cladding and identifying fabrication processes necessary for the production scale. All fuel elements fabricated and tested passed qualification tests, demonstrating strong stability and fission product retention under extended cyclic exposure to high temperatures. Early 710 fuel element testing focused on qualification for 100 hours at 2150°C and 30 thermal cycles, and fuel elements continued to be tested to meet these standards until the program scope changed to fuel development only. The 710 program's dynamic loop non-nuclear tests indicated that gas-stream impurities and radial temperature differentials are not the limiting cause of fuel element failure at operating conditions, information that static tests could not have provided. The in-pile tests demonstrated that the CERMET fuel elements could retain fission products and maintain performance for the 10,000 hours at low temperatures (<2150°C). The critical experiments and analytical correlation methods developed for the 710's design demonstrated an ability to accurately predict the neutron energy spectrum, critical mass, and control system reactivity worth. Unlike the NERVA/Rover program, future maturation of fuel elements aimed to evaluate fuel reliability, not through ground reactor testing, but through in-reactor dynamic loop testing and separate effects tests.

B-4. ARGONNE NUCLEAR LABORATORY NUCLEAR ROCKET PROGRAM 1963–1966

Beginning in January 1963, Argonne National Laboratory (ANL) embarked on a preliminary design study of a nuclear rocket, powered by a refractory metal-based fast neutron spectrum reactor. [Nuclear Rocket 1966] This program was undertaken with the guidance of the AEC's Space Nuclear Propulsion Office. The design study focused on a 2000 MW_{th} (100,000 lbf) and a 200 MW_{th} (10,000 lbf) reference system, with a core of tungsten-based cladding and CERMET fuel with a beryllium oxide reflector. The program successfully produced CERMET fuel clad with tungsten that could withstand high temperatures (2500°C) for tens of hours and dozens of thermal cycles, and test specimens exposed to high pressure and high speed hydrogen performed well. Development of fuel element fabrication was cancelled in July 1966, prior to completion, due to budgetary constraints, but the program provided confidence in the feasibility of a full engine.

The goals of ANL's experimental work were to capture neutronic information (using critical assemblies) and calculating the properties of their proposed design by developing simplified analytical techniques. Materials studied in the critical assemblies included tungsten-rhenium, tungsten cores with uranium metal and simulated UO_2, and reflectors of alumina (Al_2O_3) and BeO. At the time of the program's cancellation, a multipurpose vacuum furnace facility was being developed to evaluate material structural behavior at high temperatures. ANL also developed two flowing hydrogen loop experiments to test fuel elements—one small loop (50 kW) and one large loop (1000 kW). The smaller loop could fit smaller specimens with seven cooling channel holes (smaller than the full element cross-section). The goals of the hydrogen tests were to conduct steady-state hot hydrogen runs for 1–2 hours, conduct thermal cycling hot hydrogen runs on the order of one hour, and conduct transient nuclear testing of subscale fuel element specimens. Both loop heaters experienced multiple failures as the Re end plug thermally expanded, leading to limited data collection. The large loop was dissembled for storage. The last large loop run lasted 6 minutes, providing a sample temperature of 2270°C, with no evidence of creep-related failure in the sample. For in-reactor testing, the program developed an evacuated capsule, with tantalum-sheathed W-Re thermocouples and ThO_2 insulation, for irradiation of partial fuel elements with seven representatively sized cooling channels in TREAT. Due to uneven vapor coating of the seven-hole specimens, there is considerable temperature variation in the recorded surface temperatures of the fuel elements. However, none of the eight specimens cracked or spalled during tests that ranged from 0.2–3 seconds in total.

The results of the static, dynamic, and in-pile tests on wafers and other partial fuel element specimens demonstrate initial confidence in the selected materials' performance at high temperature cyclic operation. The ANL program's methods indicate that methods development and calculations can be performed in parallel experimental testing of fuel elements and sub-elements and that obtaining experimental verification of fuel material and fuel element performance before proceeding to fuel bundles/assemblies testing is useful in reducing test complexity and optimizing fuel element design.

B-5. SPACE NUCLEAR THERMAL PROPULSION FUEL TESTING 1988–1995

The following description of the fuel testing program for the Space Nuclear Thermal Propulsion (SNTP) program is excerpted from various documents developed by the program, including work started under Project Timber Wind. [Office of the Inspector General 1992] The final report was published by the Air Force's Phillips Laboratory in 1995 [Haslett]. This program had a goal to develop a high performing rocket engine that would more than double the performance of the best conventional chemical rockets at the time. The program was focused on PBR technology.

Testing of fuel for the SNTP originally started out as a focused single use facility for testing SNTP fuel elements and engines, expanded to the National Propulsion Test Facility, which would also be used to test other nuclear propulsion concepts. Finally, with the realization that a "do everything" facility was not affordable with the available funding sources, it was refocused and simplified to reduce its cost. The test facility was the major program cost driver. The planned program is shown in Figure B-1.

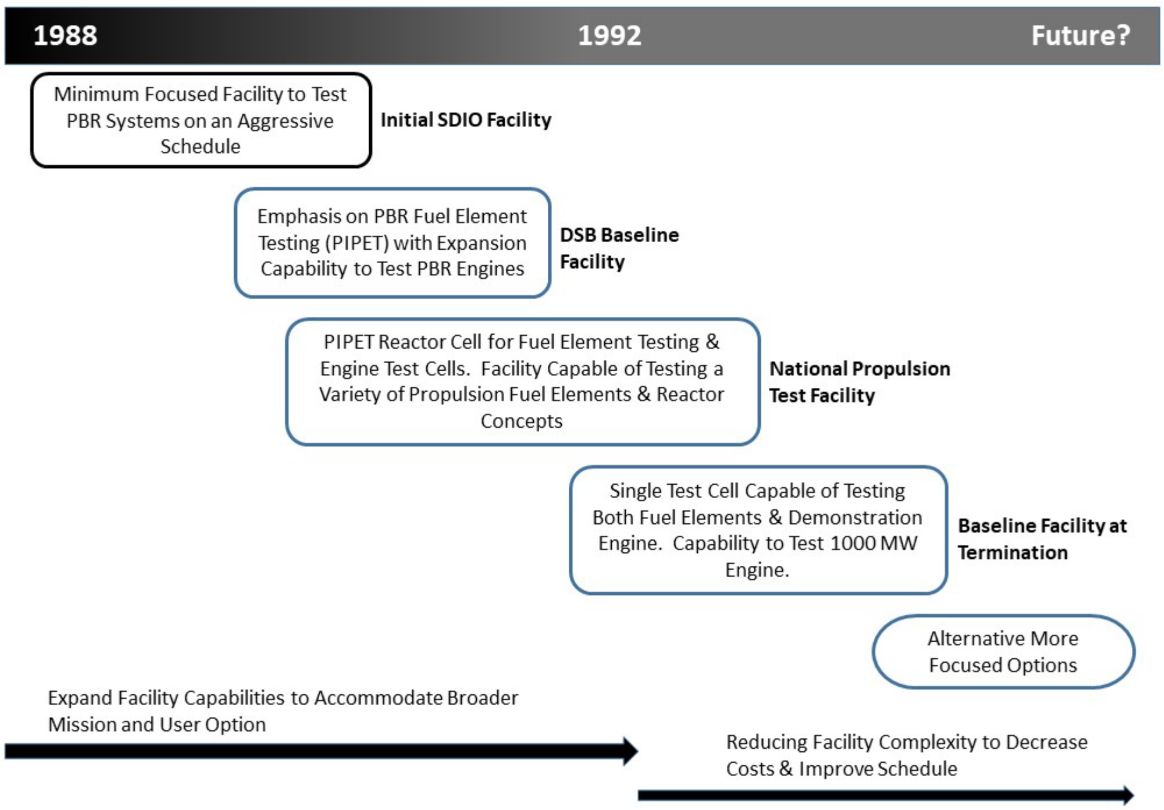

Figure B-1. Timeline for a ground test facility.

The ground test facility was initially divided into an element tester called the PBR Integral Performance Element Tester (PIPET) and a full-scale area with five engine test stands. Development of the SNTP engine also required non-nuclear test facilities capable of testing with cryogenic and high temperature (3000 K) hydrogen. This facility was required to develop many of the engine components—e.g., the turbopump, feed valves, (subscale) nozzles, and internal reactor components (i.e., hot frit). It was located in the remote San Tan Test Site, operated by Allied Signal on land leased from the Gila River Indian Community. San Tan was well along in construction when the program was terminated.

The following sections focus on the key tests that were performed in the SNTP program to qualify the concept, including nuclear design and fuel performance.

B-5.1 Blowdown Experiment, Performed at BNL

In the area of fluid dynamics and heat transfer, initial experiments concentrated on verifying the very high power densities possible in particle beds. The high power density capability results from the very high heat transfer area per unit volume, compared to other fuel geometries. This capability was verified by blowdown experiments (Horn 1992) carried out on cylindrical particle beds that were preheated to high temperatures. After reaching the desired temperature, high-pressure coolant (helium and hydrogen) at room temperature was allowed to rapidly flow through the bed (Figure B-2). Temperatures and pressures were measured as a function of time. From this data, the temporal variation of the power density in the bed were assessed.

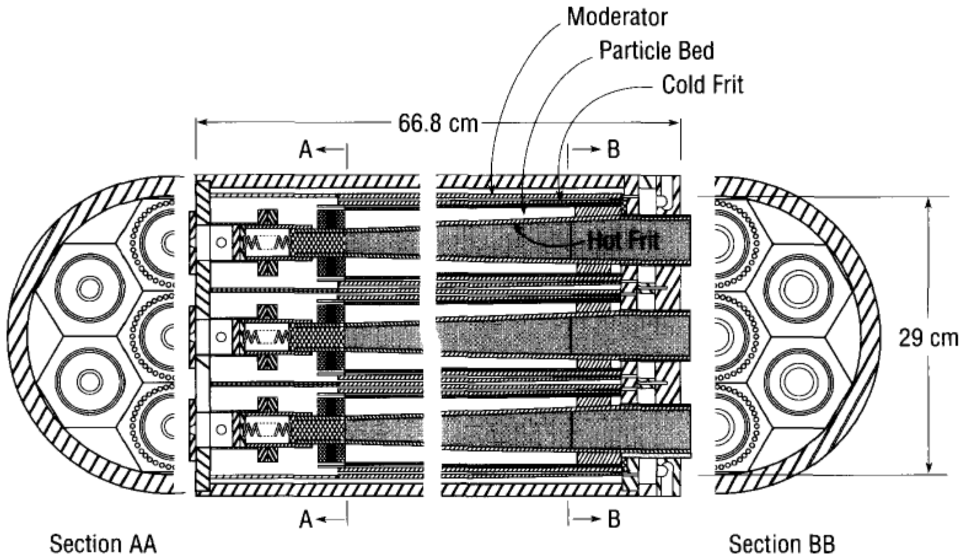

Figure B-2. Schematic representation of the seven-element blowdown apparatus.

These experiments indicated that a multi-element array of full-size PBR elements can be operated stably and predictably. Furthermore, the many thermal cycles imposed on the same fuel-element array confirmed its inherent robustness to thermal cycling. Finally, these experiments demonstrated that a PBR fuel element and associated end fittings can operate repeatedly without losing mechanical integrity. The design of these experiments were used as the basis for either more complex in-core experiments or a full-size reactor.

B-5.2 Pulse Irradiation of a Particle-Bed Fuel Element

In the latter part of 1988 and into 1989, The pulse irradiation of a particle-bed fuel element (PIPE) project was conducted to establish the feasibility of the fixed PBR fuel element assembly design for space power and propulsion applications. This first integrated testing of a PBR fuel element—designed and fabricated by Babcock and Wilcox (B&W) and tested in the Annular Core Research Reactor (ACRR) at Sandia National Laboratory (SNL)—was conducted in two series of tests: PIPE-I and PIPE-II. Although it was not possible, due to the ACRR operating capabilities, to achieve the full power densities and flow rates that would exist in an actual PBR, the tests did prove the feasibility of the PBR concept. Exhaust temperatures approached those necessary for an actual PBR rocket, and the particle fuel performed as expected. Problems experienced during the PIPE-II series of experiments caused by carbon contamination of the test loop and manufacturing of the test element did not alter the overall success of the experiments.

B-5.3 CRITICAL EXPERIMENTS (CX)

The CX critical experiments were performed at SNL to validate the MCNP models for the unique characteristics of the PBR. The results for key neutronics-related parameters (e.g., criticality, control worth, prompt-neutron generation time) are excerpted from Ludewig, et al., "Design of a Particle Bed Reactors for the Space Nuclear Thermal Propulsion Program," and are summarized below:

1. The predicted value of the multiplication factor (k-eff) is extremely close to the experimental value, i.e., 1.016 vs. 1.015. Furthermore, the slope of the boric-acid-worth curve is predicted accurately.

2. The predicted prompt-neutron generation time agrees with the experimental value to within approximately 5%. This parameter is important in predicting the transient response of the reactor.
3. To be confident that an analysis technique can accurately predict the performance of a reactor, it must not only predict correctly parameters that reflect a core-wide average (i.e., k-eff), but also parameters that result from a non-symmetric perturbation. In this case, the test was the worth of a polyethylene plug that was inserted in an outer fuel element position. The Monte Carlo method based on the MCNP code predicted this worth with exceptional accuracy.
4. An important parameter in the transient analysis of any reactor is the moderator's temperature coefficient. The results show that MCNP accurately predicts the coefficient.
5. Nuclear rocket reactors usually have to start rapidly, and during initial phases of the startup, the moderator's temperature can drop to cryogenic levels. Hence, it is important to accurately predict the variation of moderator worth in the cryogenic temperature range. Analytically determined scattering kernels were created for this analysis. The accuracy of these kernels could not be determined until a validating experiment was performed by modifying the critical experiment core, so that it allowed for a cryogenically cooled central volume. The worth of polyethylene was measured as a function of temperature, starting at 70 K up to room temperature. The experimentally measured worth and the MCNP predictions agreed closely, again confirming the MCNP code, and the analytical method used to determine the scattering kernels.
6. The control system consisted of four coaxial aluminum cylinders with six narrow neutron-absorbing strips symmetrically attached to them and surrounding a fuel element, similar to a modern control drum arrangement. By rotating the cylinders, the absorbing strips at one extreme line-up and, at the other extreme, form the equivalent of a continuous layer, thus progressively isolating the contained fuel element. This rotation results in the reactivity input used to control the reactor. A comparison of the measured and computed worths of the shutters in the open and closed positions showed that the MCNP method accurately predicts the worth of such a control device.

B-5.4 NUCLEAR ELEMENT TESTS

After preliminary confidence was established in the reactor element design, the testing progressed to higher levels of assembly. Single fuel element testing, the Nuclear Element Test (NET), was performed at SNL’s ACRR under a range of conditions to characterize and validate the design. The first several NETs were designed to validate the PBR fuel element concept, to obtain engineering data for design purposes, and to benchmark codes. The fuel, size, and materials configurations in each NET series were not to be the same as the demonstration engine element, but were to address specific design issues. The final series of NETs were going to test a fuel element that was as close as possible to the demonstration fuel element. A typical NET test article is shown in Figure B-3.

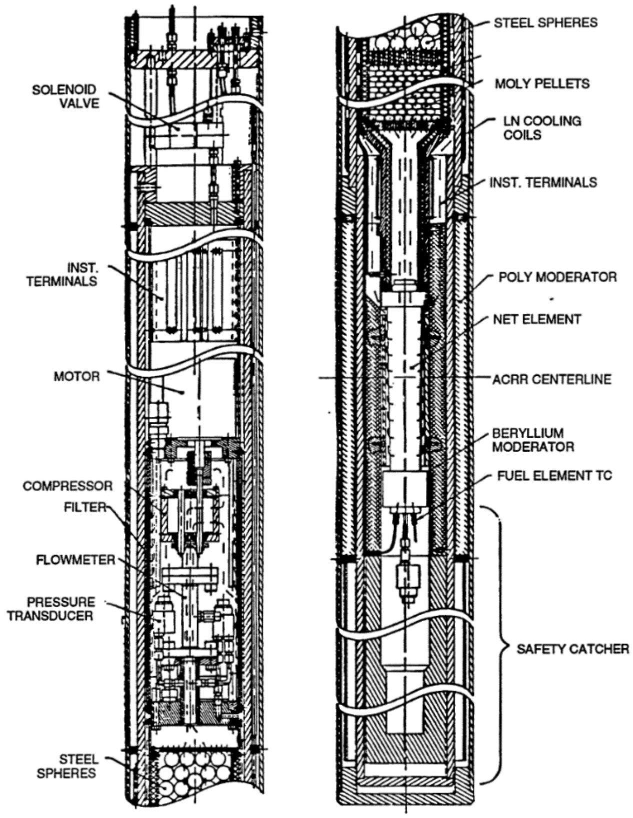

Figure B-3. Typical NET experiment capsule.

The initial NET test runs were completed satisfactorily, with the fuel element exhibiting stable flow characteristics. On the second run to a moderate temperature (= 1700 K) ACRR power irregularities halted testing. It was concluded (and later verified by an x-radiograph) that the ACRR power anomalies were indicative of potential fuel movement within the fuel element. The program office then authorized SNL and B&W to perform minimal PIE, which confirmed the existence of circumferential breaks in the hot frit.

The postulated failure from analysis of the PIE results was excessive thermal stress on the hot frit. This conclusion was based on the appearance and location of the breaks. Post-test analysis revealed that the breaks were located in the regions of maximum stresses and that the fuel bed thermal expansion had been incorrectly calculated, probably contributing to the overstress condition. With designs already in development to improve the performance of the compliant layer, and with improved pretest analysis, there was a high degree of confidence that future NET experiments with the same basic fuel element concept would have been successful.

B-5.5 PARTICLE BED REACTOR INTEGRAL PERFORMANCE TESTER (PIPET)

Options to use ACRR or TREAT to drive a full-scale individual fuel-element module under near-prototypic conditions were explored, but it was decided that due to the limitations of either option, a reactor that could test PBR fuel elements under the conditions required to obtain increased confidence in the design and its performance would be required.

Because of the very high temperatures and power densities of the PBR, using any existing national test facilities for the ground test of a full PBR engine system was precluded. As noted, several ground test facility design options were considered, and it was finally concluded that a new facility located at the Nevada Test Site (now NNSS) would be required. This new facility would allow safe and environmentally clean nuclear testing of a limited quantity of restricted size engine systems.

PIPET was a stand-alone test reactor for PBR fuel elements. The reactor design was critical with eight fuel elements and used a beryllium moderator, a graphite reflector, and hydrogen coolant, and it was controlled by drums in the reflector. The reactor exhaust would flow through a special effluent treatment system that would scrub out all particulate fission products. Noble-gas fission products would be captured in a cryogenic trap. The test would validate the fluid-dynamic and heat-transfer design of the prototype PBR fuel elements operating at high power densities for characteristic mission times. In addition, material compatibility behavior could be measured, along with damage due to the intense radiation fields. PIPET would be the first test with the combination of parameters characteristic of an operating PBR rocket. A formal preliminary design review of PIPET was held at SNL [Vernon 1992]. This review was chaired by the Air Force, with other government agencies and consultants taking part in the review and submitting formal action items, which were successfully dispositioned. With this very detailed design as a baseline, efforts were concentrated on reducing the projected costs.

Several PIPET campaigns were planned. The tests would have started with bare, prototypic fuel elements, tested at operating conditions in the PIPET driver assembly, and would have progressed to moderated fuel elements, and then to a core assembly. These test articles would have demonstrated the reactor core configuration and design. These tests were to be performed at a Nuclear Ground Test Facility at the Saddle Mountain Test Site located within the Nevada Test Site.

Following a successful PIPET campaign, reactor testing would have stepped up to the engine system level. In support of the engine ground test, the demonstration engine would have integrated selective, concurrently developed, prototypical engine system hardware components designed for the ground test environment. The demonstration engine was to be tested at the Nuclear Ground Test Facility, and if successful, would have achieved the Phase II objective of ground test demonstrating and validating the PBR engine technology and capabilities. Throughout the program, traceability to flight engine designs was to be maintained. Successful completion of Phase II of the program would have enabled Phase III for ground flight engine qualification, a flight demonstration, production, and deployment.

In spite of all efforts to design and construct a low cost test facility, the cost of environmental protection systems eventually increased the facility cost to about $500 million (in 1990 dollars), which represented about half of the total program cost. During the final year of the SNTP Program, a significantly lower cost test facility approach was considered that made use of underground weapons test tunnels, located at the Nevada Test Site, that might have become available due to nuclear weapons test ban treaty provisions. Use of these existing facilities never received a full design review; however, preliminary estimates indicated this approach could have reduced test program costs to approximately half of the baseline approach.

A summary of testing used in historic NTP fuel-qualification comprises Table B-1. Table B-2 provides a similar overview of the history of reactor testing.

Table B-1. Summary of testing types used in historic NTP fuel qualification efforts.

Test Type	Test Description	Test Objectives	Related Program(s)	Variable Range Explored	Test Article Description	Performed or just Proposed?
Thermophysical and Mechanical Property Measurement	Measure temperature dependent thermal and physical properties. Measure material response under torsional, compressive, and tensile loading.	Gather thermal, physical, and mechanical properties for use in fuel design. Evaluate impact of environmental exposure to material property degradation	ANL NR Pluto	<16,000 lbf applied at the free end (ANL NR) 863–1073 K, 2–23 hr. (Pluto Tory II-C)	Aluminum (6061-T6) plate with SS304 (and Inconel-X) clamping studs (ANL NR) BeO-10 wt% UO_2 (3/4 in. OD, 1/4 in. ID, 3 in. length) (Pluto Tory II-C)	Performed (ANL NR, Pluto)
Vibrational Testing	Fuel elements are vibrated at high frequencies representative of reactor launch or other operations.	Determine vibrational modes, evaluate dynamical stability of fuel elements	ANL NR	5–3000 cps, 1–163 “fuel” elements (interelement interactions)	Steel rods (ANL NR)	Performed (ANL NR)
Static Thermal Cycling (Non-Nuclear)	Fuel elements are placed in a heater and exposed to high temperatures or heat fluxes in a repeated (cyclic) pattern, with some amount of downtime in between heating cycles. The furnace may contain a gaseous environment (cover gas), but there is no gas flow.	Verify structural integrity of reactor materials under high temperatures / heat fluxes Verify material resilience to fatigue under repeated high heating and cooldown rates Investigate fuel element interactions with cover gas	GE-710 ANL NR Pluto	672–1922 K, 98,685 total hours → 368 cycles (GE-710) 20,000 hour life demonstration (GE-710) <3278 K vacuum furnace (ANL NR) 1305–1700 K, 550–1614 W/in to failure (Pluto Tory II-C)	Partial length fuel element, W-based or Ta-based clad (GE-710) Partial length, fuel elements (ANL NR) BeO-10 wt% UO_2 (3/4 in. OD, 1/4 in. ID, 3 in. length) (Pluto Tory II-C)	Performed (GE-710, Pluto) Proposed (GE-710, ANL NR)
Dynamic Thermal Cycling (Non-Nuclear)	Fuel elements are placed in a heater and exposed to high temperatures or heat fluxes in a repeated (cyclic) pattern, with some amount of downtime in between heating cycles. The furnace design allows for fuel elements to be exposed to a flowing working fluid, referred to as a test loop.	Verify structural integrity of reactor materials under high heat fluxes and temperatures Investigate fluid interactions and chemical compatibility between fuel and working flux	GE-710 ANL NR	1089–1922 K, up to 15,699 hr, up to 48 cycles (GE-710) 0.555 standard m^3 H_2/minute, 293–2723 K, 40.3–613 kW, 12 hours and 16 cycles–6 minutes (ANL NR small & large loops)	Full length fuel element, Ta-based clad (GE-710) Partial length, partial cross-section (subscale) 7-hole fuel elements (ANL NR)	Performed (GE-710, ANL NR)

In-pile Static testing (Nuclear)	Fuel elements are placed in a reactor (in-pile) and exposed to high neutron fluxes in order to cause fission. Fission heating is used to heat the fuel to prototypic temperatures. The reactor may contain a gaseous environment (cover gas), but there is no gas flow.	Verify structural integrity of reactor materials under fission reactor operation Verify fission product retention (e.g. below 1% release) Identify irradiation-induced dimensional changes	GE-710 SNTP ANL NR	<1811 K, 121,066 hr → 231 cycles (GE-710) 1800–3000 K, 100–600 s (SNTP Particle Nuclear Tests 1-5) 1073–3023 K, 0.2–3 s (ANL NR TREAT)	Partial length fuel element, W-based or Ta-based clad (GE-710) CERCER fuel particles (SNTP PNT 1-5) Partial length, partial cross-section (7 coolant hole) CERMET elements (ANL NR)	Performed (GE-710) Performed (SNTP PNT 1-5) Performed (ANL NR)
In-pile dynamic Cycling (Nuclear)	In-Pile testing in a repeated (cyclic) pattern, with working fluid, downtime between operating cycles.	Verify mechanical integrity of reactor materials under fission reactor operation Verify fission product retention (e.g. below 1% release) Verify material resilience to fatigue under repeated fission heating and cooldown	NERVA/Rover (Nuclear Furnace) SNTP (PIPE 1-7, NET 1.2, TREAT)	<2444 K exit temperature, 44 MW_{th}, 108.8 minutes total → 4 cycles (NF-1) 3–60 s, 0.09–1.6 MW/L (SNTP PIPE 150 K–2300 K H_2 temperature, 10 s/2 cycles (SNTP NET 1.2) 10–20 MW/L, 5-10 s (SNTP TREAT)	NERVA/Rover: 47 (UC-ZrC)C-carbon composite elements & a seven-element cluster of single-hole pure (U,Zr)C carbide (NF-1) Unknown, information at SNL (PIPE 1-7) CERCER fuel particles contained in a graphite/NbC hot frit and a stainless steel cold frit (NET 1.2)	Performed (SNTP PIPE 1-7, NET 1.2) Proposed (SNTP TREAT) Proposed (GE-710)

Table B-2. Summary of testing types used in historic NTP reactor qualification efforts.

Test Type	Test Description	Test Objectives	Variable Range Explored	Test Article Description	Related Program(s)	Performed or just Proposed?
Cold Flow Testing	Cryogenic / room temperature fluid is pumped through fueled or mock up reactor core	Evaluate erosive effects and fluid interactions with components	<342 pps (Tory-IIA)	un-fueled or as built reactor	Rover / NERVA Pluto (Tory-IIA)	Performed (Rover/NERVA, Tory-IIA)
Hot Flow Testing	Hot fluid is pumped through fueled or mock up reactor core	Evaluate corrosion effects and fluid interactions with components, evaluate thermal response of reactor materials	300–1514 K, 40 MW_{th}, 103 pps (Tory-IIA)	un-fueled or as built reactor	Pluto (Tory-IIA)	Proposed (Tory-IIA)
Zero Power Critical Experiments	The reactor or representative assembly is demonstrated to reach criticality while minimizing fission heat generation	Evaluate neutron population (spectrum), power distribution, reactor transient response, temperature feedback; control; kinetics parameters	200 -1800 pps, 1 kW_{th}, 1.33 hours (Tory-IIC Run 87) 107–283 L core volume, 192–494 kg critical mass (ANL NR Critical Experiments 1-9)	mock up, as built reactor, or subscale reactor U-W CERMET with Al/BeO/Alumina reflector (CX 1-9)	Rover / NERVA Pluto (Tory-IIC Run 87) GE-710 ANL NR CX 1-9 SNTP	Performed (Rover / NERVA) Performed (Tory-IIC Run 87) Performed (ANL NR CX 1-9) Performed (SNTP-CX)
Reactor Ground Tests	Full power critical experiments at various power / thrust levels, no engine hardware		548–1096 MW_{th}, 6–46 minutes → 4 cycles (Rover Kiwi, Phoebus, Pewee, NERVA NRX-A2, A3) <1120 MW_{th}, 14.5, 15.5 and 62 minutes (NRX-A5, A6) 200–1260 pps, 313 MW_{th}, 750–1405 K chamber temperature, 1.75 hours (Tory-IIC Run 88) 200–1660 pps, 485 MW_{th}, 745–1566 K chamber temperature, 1 hour (Tory-IIC Run 89) Fast startup: 1644 K, 1800 pps, 480 MW_{th} in 1 minute, hold for several (Tory-IIC)	Mock up, as built reactor, or subscale reactor	NERVA/Rover Pluto (Tory-IIC Run 88, 89) SNTP	NERVA/Rover Performed (Tory-IIC Run 88, 89) Proposed (TORY-IIC, SNTP)

			200–2000 s, 1–75 MW/L (SNTP Particle Bed Reactor Integral Performance Tester)			
Integrated (Engine-Reactor) System Testing	Full power critical experiments with engine hardware at nominal power / thrust levels	Demonstrate repeatable, controllable operation of reactor with desired performance	1140 MW_{th}, 2272 K chamber temperature, 105 minutes total → 24 cycles (XE-Prime) 100–1000 s, 20–60 MW/L (SNTP Ground Test Article Engine Test)	As built reactor, or subscale reactor with engine integration	NERVA/Rover (XE Prime) SNTP (ground-test article [GTA])	Performed (NERVA/Rover XE-Prime) Proposed (SNTP GTA)

Appendix C
Foreign Facilities

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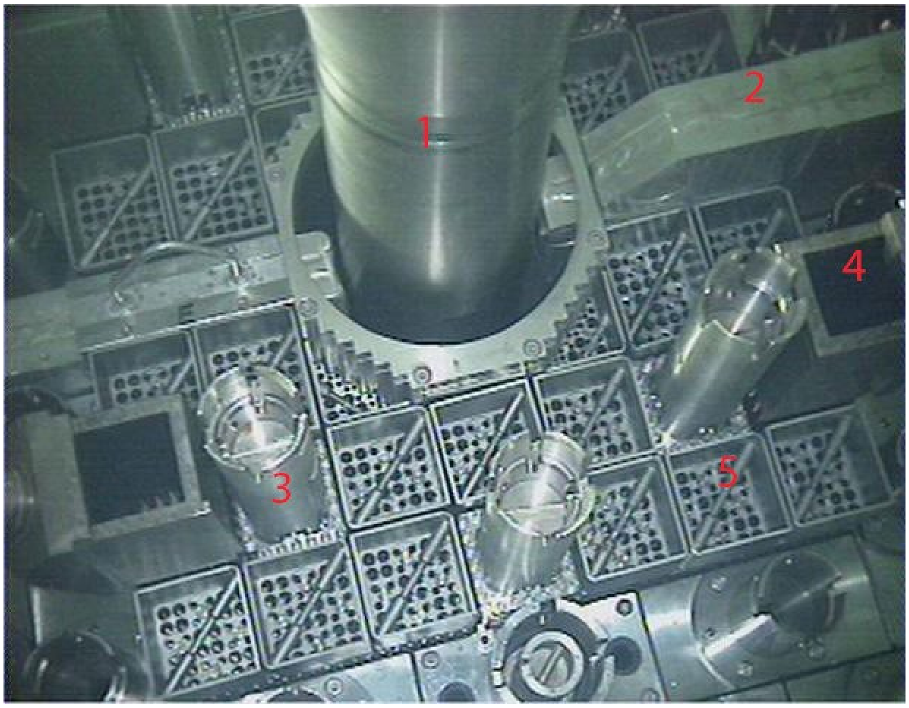
There are several foreign nation state facilities that offer capabilities to irradiate NTP fuels and system components under prototypical conditions, with some limitations on hydrogen flow. Use of foreign nation state facilities often has some undesirable complications associated with cost, schedule, quality, physical security, and export compliance. The terms and conditions associated with conducting nuclear related irradiation tests vary among nations for the facility being considered. However, there may be opportunities where the use of some international facilities may enable a rapid demonstration or provide a cost advantage for a highly complex tests that would be performed domestically at a later date. Some nations, such as Russia, are explicitly prohibited in collaborative testing, especially for nuclear systems of this nature. For this reason, facilities such as Rosatom's 100 MW High-Flux Reactor or the BOR-60 Reactor, while technically capable of accommodating hydrogen irradiation loops, are currently disqualified from supporting NTP fuels irradiation testing for the U.S. government or its contractors.

The lead time for experiment execution should be anticipated to be of the order 2 years or more, regardless of the location when accounting for export licensing, engineering, shipping, and installation. Furthermore, some facilities have limited capacity for new irradiation projects due to the volume of preexisting workload at the facilities.

Return shipment of experiments should be considered during the early stages of experiment design. International cask shipments are a known process but are costly and require long-range planning.

C-1. CABRI

The CABRI reactor, in Cadarache, France, offers access to a 23.7-25 MW steady-state or multiple GW pulsed reactor operations using a ^{3}He power shaping control system. The reactor operation of CABRI is very similar to that of TREAT, with the significant addition of steady-state operating power. Most recently, CABRI was reactivated in October 2015 to support the continuation of the CABRI International Program's LWR safety testing series following facility seismic and operational safety upgrades. The facility houses a pool reactor and separate experiment loop conditioning systems (see Figure C-1). CABRI is able to perform irradiation testing on commercial LWR fuels and has more than sufficient length to accommodate a full prototype NTP fuel assembly or bundle of fuel/moderator assemblies. LWR fuel experiments have included self-contained loops with instrumentation. Similar design approaches can be applied to a flowing hydrogen loop approximately 6 inches in diameter with effluent containment. Of significance for consideration, the U.S. DOE has an active collaboration with Institut de Radioprotection et de Surete Nucleaire (IRSN) and the Commissariat à l'énergie atomique et aux énergies alternatives (CEA), which operate both reactor experiments and the reactor facility to perform reactor safety testing



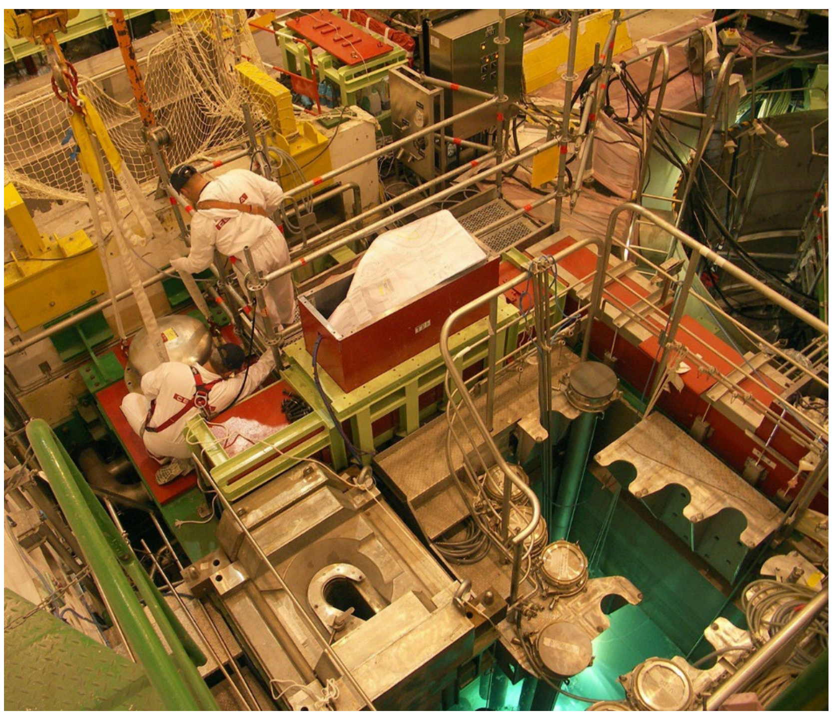


Figure C-1. The open pool of the CABRI reactor is shown on the top with the plant working floors shown on the botttom. The numbers represent the self-contained irradiation loop.

C-2. Institute for Nuclear Research-Pitesti 14-MW TRIGA

The Romanian 14-MW TRIGA reactor is an open pool-type reactor that is highly adaptable to installation of gaseous hydrogen flow irradiation loops (Figure C-2 and Figure C-3). Originally designed to operate at 28 MW, but licensed to operate at 14-MW, the Institute for Nuclear Research (INR) TRIGA is larger than any domestically operated U.S. TRIGA facility. Power coupling experiments may be performed as a pathfinder for a domestic facility that may engage in the new construction of a 14MW TRIGA. A loop in-core test section that is from 1.5 to 4 inches in diameter, with a specimen length of up to 15 inches, may be considered for testing at the facility [Preda et al. 2004].

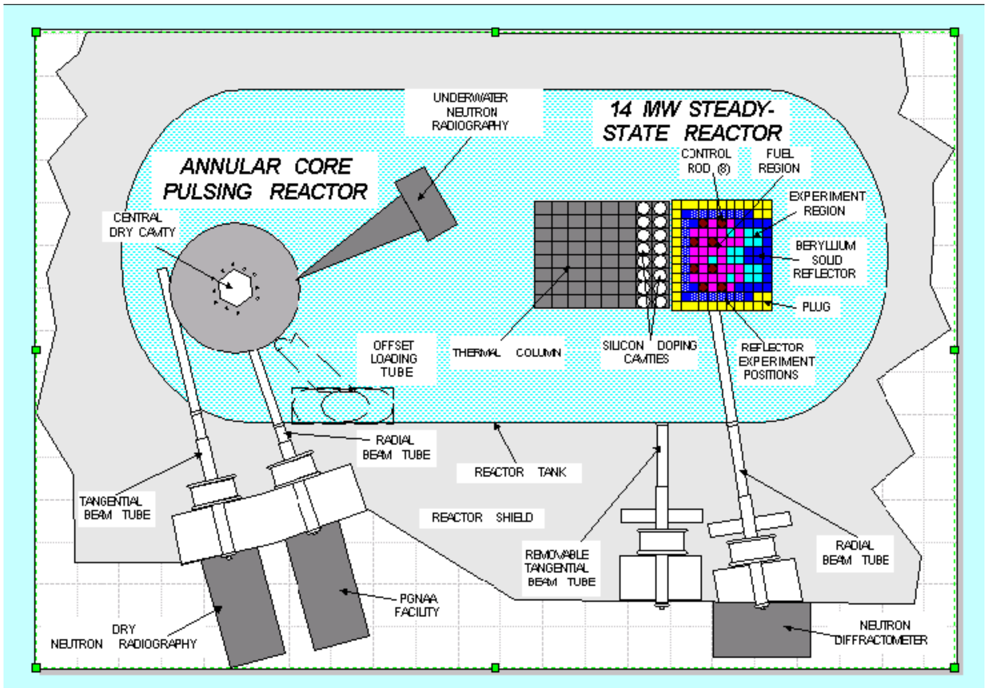


Figure C-2. Pool/plant schematic of INR’s 14-MW TRIGA reactor Facility.

Figure C-3. View of the 14-MW TRIGA core portion of the INR facility.

C-3. IVG.1M

Commissioned in 1975 and initially dedicated to testing of NTP fuel elements and assemblies for the Soviet Union, the IVG.1M reactor is located near Kurchatov, Kazakhstan. The IVG.1M reactor operates at a power of 72 MW, with a thermal neutron flux of 3.5E14 n/cm^2/s at the dry, central, physical experiment cavity. The reactor core height is 800 mm (31.49 inches) and could certainly accept specimen tests that are similar to those tested at TREAT (i.e., 10–20-inch long specimens), but at sustained prototypical temperatures for NTPs and under flowing H_2. The driver core of IVG.1M is comprised of fuel bundles or assemblies that are housed in water cooled channels. Most recently (2015–2016), IVG.1M was converted from HEU to a low-enriched uranium driver core fuel loading. During this time, the experimental handling equipment and support infrastructure were also upgraded, including gas handling apparatus such as for flowing hydrogen. Capsules and loop experiments can be installed into the dry, physical experiment cavity and can extend both above and below the core, where there is excellent access for loop support infrastructure connections to be made (see Figure C-4). IVG.1M is available for collaborative research between the U.S. DOE and Institute of Atomic Energy of the National Nuclear Center of the Republic of Kazakhstan.

Figure C-4. (Left) View of the top side of the IVG.1M reactor with an isolated loop capsule experiment and lead outs installed at the center of the reactor core. (Right) View of access under the IVG.1M core.

Advantages for using foreign facilities include that they are existing test facilities with many of the desirable facility parameters from Table 1, meaning design and fabrication of experiments can be much more economical than in domestic facilities.

However, disadvantages of using foreign facilities include that

- Many aspects of the SNP Program are export-controlled, in particular the form and content of reactor fuel
- Transport of nuclear materials and components is complicated, time-consuming, and expensive.