

CORE DESIGN ACTIVITIES OF THE VERSATILE TEST REACTOR – CONCEPTUAL PHASE

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ABSTRACT

The Versatile Test Reactor (VTR) is a new fast spectrum test reactor being developed in the United States under the direction of the US Department of Energy, Office of Nuclear Energy. The VTR mission is to enable accelerated testing of advanced reactor fuels and materials required for advanced reactor technologies. This includes neutron irradiation capabilities which would support alternate coolants including molten salt, lead/lead-bismuth eutectic mixture, gas, and sodium. The VTR aims at addressing most of the needs of the various stakeholders, which is primarily composed of advanced reactor technologists, developers and vendors, as well as a number of others interested parties.

Design activities are underway targeting a first criticality date by 2026, with General Electric recently joining the project to contribute to the VTR plant design. Current efforts are focused on all aspects of the VTR design, with the core design being at the center of the initial steps. The VTR is currently proposed as a 300 MW_{th} sodium-cooled fast reactor able to reach peak fast flux levels in excess of 4.0×10^{15} n/cm²-s (and total flux level of about 6.0×10^{15} n/cm²-s). In this configuration, it is using ternary metallic fuel with reactor-grade plutonium and 5% low-enriched uranium.

KEYWORDS: versatile test reactor, reactor design, advanced reactors, sodium-cooled fast reactor

1. INTRODUCTION

The Versatile Test Reactor (VTR) is a program supported by the United States Department of Energy (US-DOE) aiming at designing and building a fast-spectrum test reactor to bridge capability gaps related to accelerated fuels and materials testing and qualification for nuclear applications. In its current conceptual design stage, the VTR is a 300 MW_{th} sodium-cooled fast reactor of the pool type. It will not generate electricity as to avoid competing secondary missions which could divert it from its primary mission, that is irradiation testing. The heat generated will be released to the air through several air-dump heat exchangers, conceptually similar to those used for the Fast Flux Test Facility (FFTF). The overall plant design for the VTR is based on the PRISM Mod-A reactor, designed by GE Hitachi Nuclear Energy [1]. VTR will be able to concurrently support development of several reactor technologies by providing a wide range of irradiation services. In particular, VTR will allow for use of coolants types different from the primary coolant in designated test locations, allowing it to provide value to a wide range of advanced reactors designs.

This paper provides a short summary of the VTR project status in Section 2. This is followed by a detailed description of the VTR core design and of its performance characteristics in Section 3. The steady-state reactor performance characteristics as well as the reactivity control performance are discussed.

2. VTR PROJECT STATUS

The VTR program started in 2017 under the auspices of the US-DOE Office of Nuclear Energy (DOE-NE). The objectives are to bridge a capability gap (high-flux fast spectrum neutron irradiation) for the nuclear industry in the U.S. While the U.S. had been pioneering the demonstration of fast-spectrum reactors in the early age of nuclear energy, notably with EBR-II and FFTF, no reactor currently operating in the U.S. can offer significant fast flux levels. Fast flux is very important for irradiation testing as it allows achieving material damage much faster and can reduce the irradiation time needed to study new materials by over one order of magnitude. With the growing interest in advanced reactors, irradiation needs are increasing, making having the capability to perform accelerated fuels and materials testing of the utmost importance.

The mission of the VTR program is to help accelerate the testing of advanced nuclear fuels, materials, instrumentation, and sensors. It will also allow DOE to modernize its essential nuclear energy research and development infrastructure, and conduct crucial advanced technology and materials testing necessary to re-energize the U.S. nuclear energy industry. The timeline envisioned by the VTR program is to complete the construction of the reactor and start its operation by 2026. The reason for this accelerated schedule is to enable establishing this much-needed capability in time to support most advanced reactor technologies being currently developed.

The VTR objectives are to offer the following capabilities:

- Fast flux in excess of 4×10^{15} n/cm²-s;
- Very high dpa level, in excess of 30 dpa/year;
- Test volumes in excess of 7 liters per test location;
- Large number of potential test locations;
- Effective testing heights of at least 60 cm;
- Ability to test fuel and material in prototypical environments other than sodium. This includes, but is not limited to, lead, lead-bismuth, helium and molten salts.

In 2018 the VTR program became a design project under the DOE Order 413.3B [2] that governs the management of capital acquisition projects. This directive is the process through which the DOE can enable the acquisition or construction of capital assets. This is a reflection of the intent to ensure the VTR program is being managed in accordance with the expectation set forth for a construction project. In particular, the purpose of this directive is “to provide the DOE with program and project management direction for the acquisition of capital assets with the goal of delivering projects within the original performance baseline, cost and schedule, and fully capable of meeting mission performance, safeguards and security, and environmental, safety, and health requirements unless impacted by a directed change”.

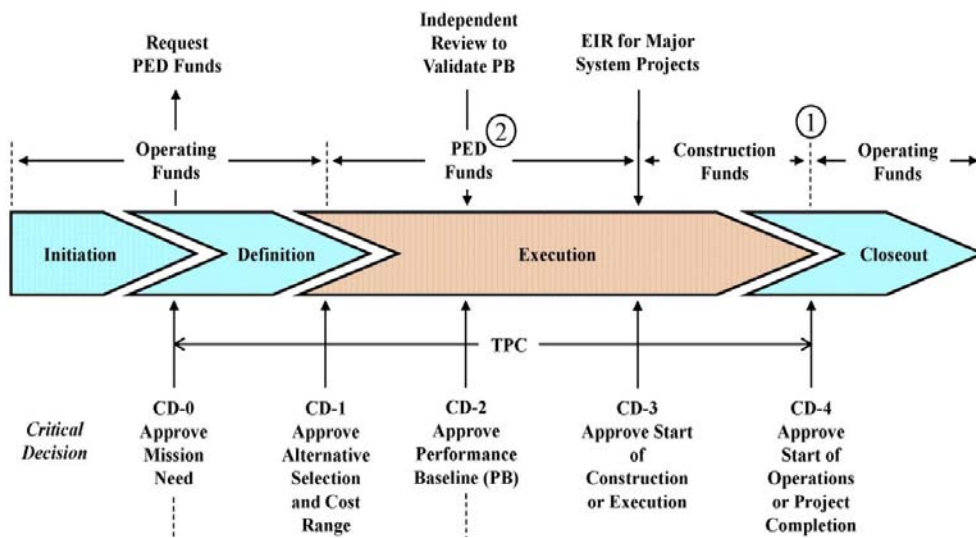


Figure 1. Critical decision chart as part of DOE directive 413.3B (Source: DOE)

In practice, by becoming a project, the VTR program must deliver on a number of critical decision (CD) points to approve the project from the design stage to final construction and operation. The five required CD points are presented in Fig. 1. CD-4 is the last one and would correspond to the start of operations for the VTR. The VTR program has successfully completed the CD-0 phase in February 2019 [3] and is now aiming at CD-1 which is planned to be completed during U.S. fiscal year 2020.

3. DESIGN ACTIVITIES

3.1. Early Determinations

The core design efforts for the VTR program have been initiated in the early stages of the program. Initial efforts focused on narrowing down the design space, in particular with respect to the core power level, and types of fuels that would allow achieving the performance objectives stated in Section 2. These early studies have been detailed in previous publications [4] and therefore only the major outcomes are mentioned here. Fig. 2 provides a summary of the results, showing the approximate minimum core power level required to achieve a particular peak fast flux level for various types of fuels.

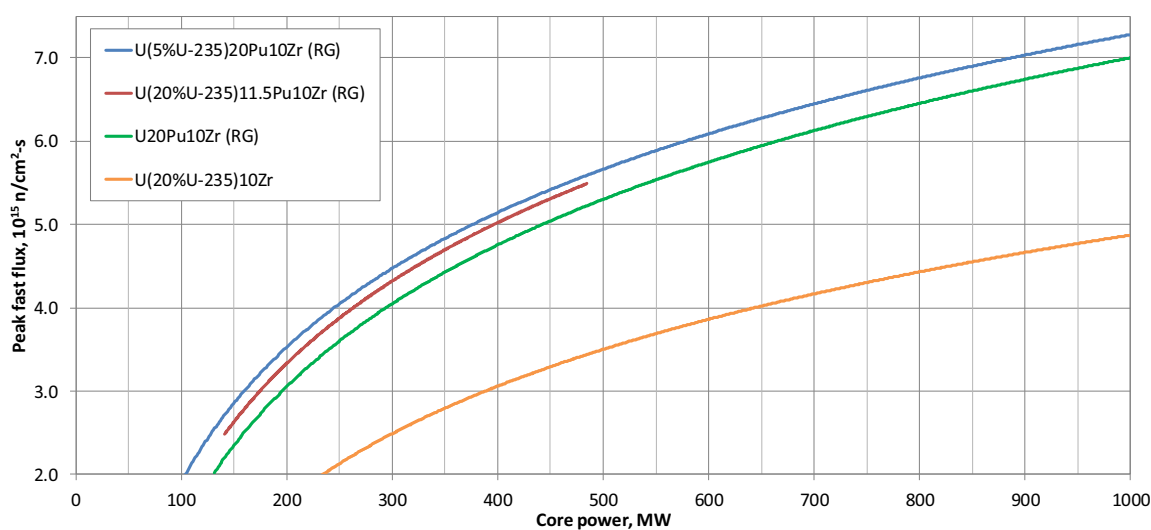


Figure 2. Required Core Power vs. Peak Fast Flux

These results indicated that with metallic fuel bearing plutonium the required core size could be limited to 300 MW_{th} or less, but that if only high-assay low-enriched uranium is available then the core size would need to be at least 650 MW_{th}. Based on these early results, the preferred fuel form for the VTR core design activities is the ternary U-20Pu-10Zr, using reactor-grade plutonium and 5% LEU. The 20wt% plutonium is determined based on the existing irradiation database for similar fuel, and the 5% uranium enrichment is based on the current enrichment capabilities in the U.S. This is the preferred fuel forms, some efforts were made on exploring other potential fuel forms [5,6].

A number of other early decisions were made with respect of the reactor design, such as to have the VTR being a pool-type reactor, with no energy conversion capabilities. Selection of the technologies supporting the VTR, not discussed in this paper, are primarily driven by leveraging demonstrated and low-risk technologies in order to minimize the overall project risk and meet the desired design and deployment timeline.

3.2. Core General Overview

Going from the pre-conceptual into the conceptual design phase, the VTR core has been refined, with more details included in the models, with a number of parameters adjusted based on interaction with the various VTR teams, such as the fuel, engineering and safety teams. The overall core layout remained the same, shown in Fig. 3, with 66 fuel assemblies, six control rods, three safety rods, 114 radial reflectors, 114 radial shield reflectors, and 10 shown test locations. The maximum power it is intended to generate remained 300 MW_{th}. There will be up to five instrumented test locations as shown in Fig. 3, which means that experiments needing instrumentation will only be able to be inserted in one of these predetermined locations. Other test locations, referred to as non-instrumented test locations, could be positioned anywhere in the core, and their number can vary. This means that the performance reported in this document pertain to the specific configurations shown in Fig. 3, and if non-instrumented test locations were to be located in different positions these performance characteristics would be slightly different. The same remark also extends to the contents of the loaded experiments. With different experiments loadings expected each operating cycle, the core performance characteristics will slightly vary and differ from those presented here. However, VTR being fast reactor, the sensitivity of performance characteristics to the detailed loading arrangement is limited.

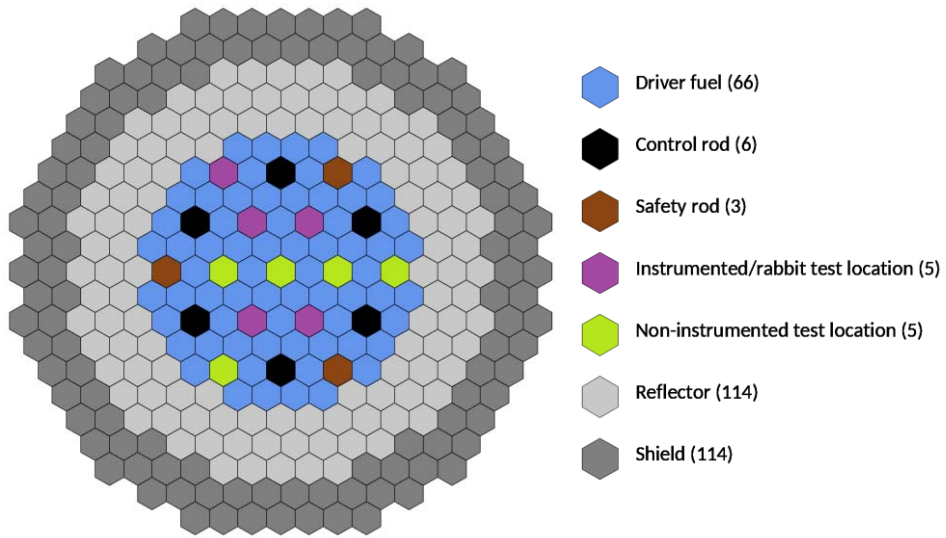


Figure 3. VTR core layout

<u>Fuel</u>	<u>Test</u>	<u>CR</u>	<u>Reflector</u>	<u>Shield</u>
Outlet nozzle	Outlet nozzle	Outlet nozzle	Outlet nozzle	Outlet nozzle
Upper reflector	Test reflector	Empty CR	Reflector	Shield
Pin-reflector transition	Empty test	Active CR		
Empty plenum				
Filled plenum				
Fuel				
Reflector-pin transition		Empty CR		
Top of lower reflector				
Lower reflector	Test reflector			
Inlet nozzle	Inlet nozzle	Inlet nozzle	Inlet nozzle	Inlet nozzle

Figure 4. Axial Layout of the VTR Assemblies

The overall length of each assembly from the extremity of the inlet nozzle to the extremity of the outlet nozzle is about 3.8 meters. The various regions currently modeled for each type of assembly is schematically represented in Fig. 4. At the current stage of the design, geometry details have only been determined for the fuel and control rod (CR) assemblies as these are the most impactful assemblies with respect to core performance.

3.3. Neutronics Performance Characteristics

The performance characteristics of the reference VTR core have been determined for an equilibrium cycle using the Argonne Reactor Computational code suite [7]. Transport theory with the P5 flux approximation was used with the ENDF/B-VII.0 nuclear database. The assumed fuel management strategy used to approach equilibrium conditions is illustrated in Fig. 5. The 12 central most fuel assemblies remain in the core for 3 cycles, the next 18 fuel assemblies remain in the core for 4 cycles, the following 12 fuel assemblies remain in the core for 5 cycles, and the remaining 24 assemblies remain in the core for 6 cycles. This is identified in Fig. 5 by the number in each assembly.

The reactor physics performance characteristics are provided in Table 1 for the equilibrium cycle. The “test peak fast flux” corresponds to the average fast flux achieved in a 20 cm tall section in the central test assembly. The “absolute peak fast flux” is the value achieved locally anywhere in the core. The maximum absolute/relative power variations correspond to the largest absolute/relative power variations observed in any fuel assembly between the beginning of an equilibrium cycle (BOEC) and the end of an equilibrium cycle (EOEC).

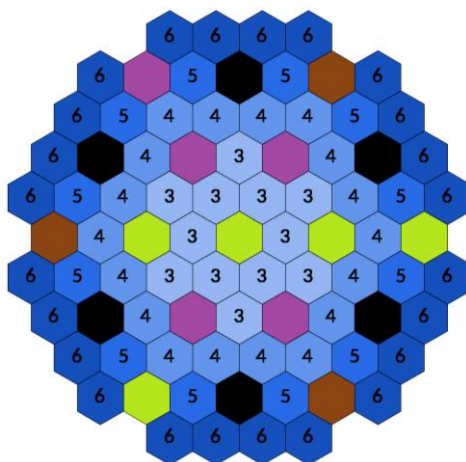


Figure 5. Fuel Management Scheme

The required plutonium weight fraction for the model used is a little larger than the target value of 20wt%. In practice, the fuel will be manufactured with 20wt% plutonium. Several of the test assemblies are expected to contain fuel, which will easily offset the reduction in reactivity when using 20wt% plutonium rather than the reported 20.13%. In fact, it is very likely that due to the fuel contained in the test assemblies it will be necessary to load fewer than 66 driver fuel assemblies.

Table 1. Reactor Physics Performance Characteristics of the VTR Core

Characteristic	Unit	Value
Core power	MW _{th}	300
Cycle length	EFPD	100
Number of batches	-	3 to 6
Plutonium concentration	wt.%	20.14%
Uranium enrichment	at.%	5%
Maximum excess reactivity	pcm	2186
Test peak fast flux at BOEC	$\times 10^{15}$ n/cm ² -s	4.34
Test peak fast flux at EOEC	$\times 10^{15}$ n/cm ² -s	4.23
Absolute peak fast flux at BOEC	$\times 10^{15}$ n/cm ² -s	4.54
Absolute peak fast flux at EOEC	$\times 10^{15}$ n/cm ² -s	4.43
Average assembly power	MW _{th}	4.55
Maximum assembly power at BOEC	MW _{th}	6.45
Maximum assembly power at EOEC	MW _{th}	6.17
Maximum absolute power variation	MW _{th}	0.28
Maximum relative power variation ^a	-	7.1%
Fuel assemblies/cycle	-	14.9
Heavy metal charge/cycle	kg/cycle	596.2
Uranium required/cycle	kg/cycle	462.8
Plutonium required/cycle	kg/cycle	133.4
Average discharge burnup	GWd/t	50.3
Assembly-averaged peak discharge burnup	GWd/t	52.5
Peak discharge burnup	GWd/t	61.0

^a Excluding variation in the primary control rods which is 36%

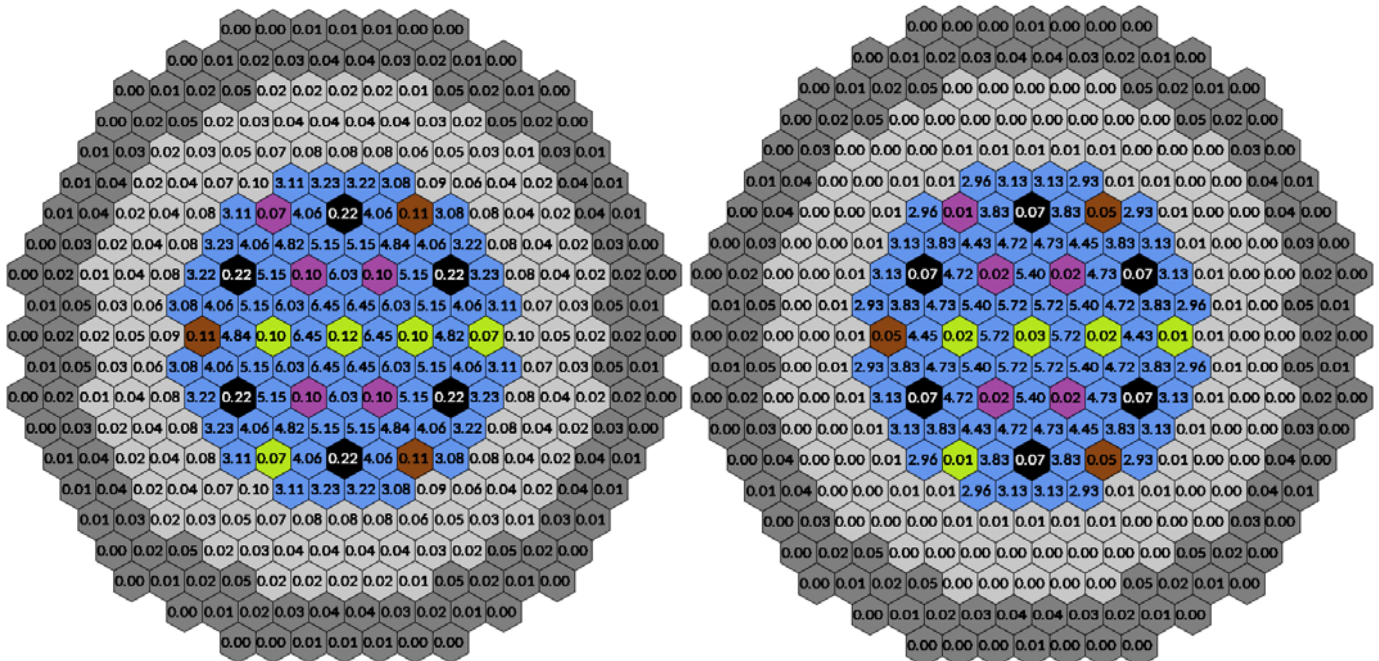


Figure 8. Total Power Distribution at BOEC (left) and EOEC (right)

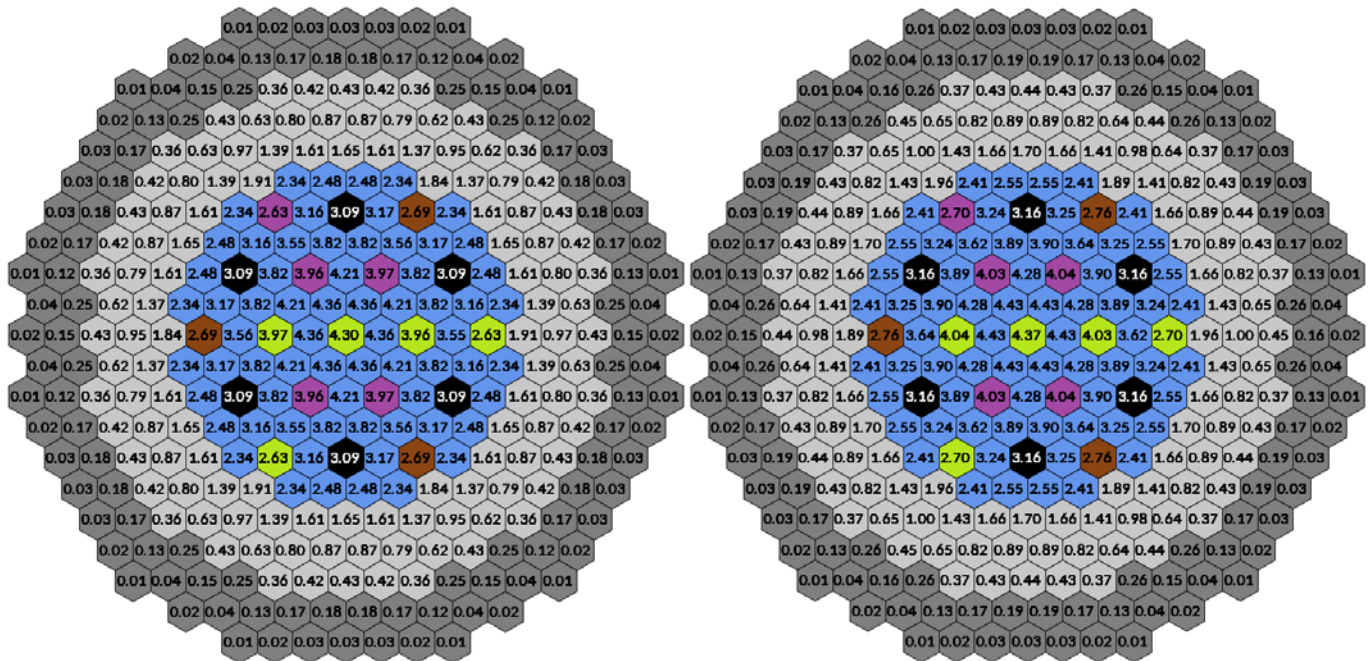


Figure 9. Peak Fast Flux Distribution at BOEC (left) and EOEC (right)

The core power, 300 MW_{th}, accounts for all sources of power within the core, in particular for fission power, neutron heating, gamma heating, and decay heat in the fuel, reflector, control, test and shield assemblies. The total power distributions at BOEC and EOEC are shown in Fig. 6. The corresponding peak fast flux achieved at BOEC and EOEC are shown in Fig. 7.

3.4. Control Rod Worth, Reactivity Feedback and Shutdown Margins

The reactivity worth of the control rods has been determined for the primary and secondary systems independently. The total systems worths are obtained by calculating the reactivity difference between the control rods being fully withdrawn and the control rods being fully inserted into the fuel region. The calculations were also repeated with the most reactive control rod stuck at the operating position. For the primary system evaluation, this means one rod remains at inserted about one third into the fuel region. For the secondary system evaluation, this means one rod remains fully withdrawn out of the fuel region. When using natural boron in the B₄C absorber, the worths of the control systems are summarized in Table 2. Through enriching boron, these worth could be significantly increased, if deemed necessary [8].

Table 2. Primary and Secondary Reactivity Control System Worths at BOEC

CR system	Primary	Secondary
All rods in, pcm	6600	2150
All rods in minus one, pcm	5500	1400

Reactivity coefficients have been determined for the core at equilibrium with both the. All reactivity coefficients are found to be negative, including the sodium void worth and sodium density coefficient even though plutonium-based fuel is used. This favorable outcome is due to the very large neutron

leakage probability of the core – nearly 45% – which compensates for spectrum hardening when sodium gets voided (or its density reduced).

The shutdown worth requirements have been determined using the reactivity coefficients to determine the reactivity difference between the different states of the reactor (*e.g.* from hot full power to cold zero power). The approach used to determine the required worth to achieve safe shutdown of the core has been described in [9]. It is calculated that the shutdown worth requirement for the primary system is about 4300 pcm, and for the secondary system it is about 760 pcm.

The required reactivity worths of the primary and secondary control systems are lower than the calculated available worth of these systems, when assuming that the most reactive control rod is stuck at the critical position. However, the assessment performed here did not yet account for the loss of reactivity in the control systems due to boron depletion and the model use does not capture the self-shielding effects.

3.5. Irradiation Capabilities

The testing performance achieved with the current set of assumptions, for the various test assembly locations, are summarized in Table 3. It should be noted that the flux reported could be affected in either way by the materials or fuels loaded in the given test location.

Table 3. Summary of Irradiation Performance Characteristics

Test assembly location	Row 1	Row 3	Row 5	Reflector
Peak fast flux, n/cm ² -s	4.17E+15	3.70E+15	2.26E+15	1.0-1.5E+15
Volume with fast flux above 1e15 n/cm ² -s, liter	17.7	17.1	13.8	-
Volume with total flux above 1e15 n/cm ² -s, liter	22.2	21.3	18.7	-
Peak fast fluence/year, n/cm ²	1.32E+23	1.17E+23	7.13E+22	3.0-4.5E+22
Estimated dpa/year in Fe	65	60	35	10-20

4. SUMMARY

The VTR project is a major initiative supported by the U.S. Department of Energy aiming to design and build a fast flux test reactor at one of the U.S. DOE sites by 2026. This project has been enabled by many years of work in support of advanced reactors development as well as by the recognition by the government of important capability gaps that would hinder the development of nuclear energy in the U.S.

Currently in the conceptual phase, the VTR is a 300 MWth pool-type sodium-cooled fast reactor fueled with ternary fuel bearing plutonium. The core fits into the PRISM Mod-A plant layout and will offer peak fast fluxes in excess of 4.3×10^{15} n/cm²-s, with many test locations concurrently available, each having several liters of available testing space. All these test locations will enable achievement of over 30 dpa/yr, with a maximum of 65 dpa/year. Furthermore, the VTR will enable irradiation testing with coolants other than sodium through the cartridge loop systems without requiring any modifications.

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The work reported in this summary is the results of R&D studies supporting a VTR concept, cost, and schedule estimate for DOE-NE to make a decision on procurement in the future. As such, it is preliminary.

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