VERSATILE TEST REACTOR: CONCEPTUAL CORE DESIGN OVERVIEW

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Abstract

The Versatile Test Reactor (VTR) is a reactor under development in the United States of America to provide a very high-flux fast neutron source that will support the development of advanced reactor technologies. This reactor will accelerate the irradiation testing of advanced nuclear fuels, materials, and potentially other components. The development efforts are structured in several phases to mature the design, and the conceptual design phase was recently completed. The VTR core design developed in this phase will be presented and discussed in this paper.

The conceptual design for the VTR is a 300 MWth pool-type sodium-cooled fast reactor. The core contains a total of 313 assemblies, 66 of which are fuel drivers, and 10 are representative test locations. Ternary metallic fuel is used in the driver fuel and allows achieving a peak fast flux of about 4.5×10^{15} n/cm^2-s in the central test location, corresponding to material damage rate in excess of 50 dpa per year. The irradiation conditions offer large irradiation volumes with very high flux levels, as well as experimental flexibility through the use of cartridge loops. Cartridge loop experiments allow using a self-contained coolant, different from the reactor coolant, to provide prototypical irradiation conditions pertaining to other types of advanced reactors. The reactor can accommodate simultaneous cartridge loop experiments in up to five locations. In addition, VTR has a rabbit facility that permits insertion and removal of irradiation samples during operation.

1. INTRODUCTION

The Versatile Test Reactor (VTR) is a reactor under development in the United States of America to provide a very high-flux fast neutron source that will support the development of advanced reactor technologies [1]. This reactor will accelerate the irradiation testing of advanced nuclear fuels, materials, and potentially other components. This includes neutron irradiation capabilities which would support alternate coolants including molten salt, lead/lead-bismuth eutectic mixture, gas, and sodium. The VTR project is supported by the U.S. Department of Energy, Office of Nuclear Energy and is a 413.3(b) project. As part of following the guidance set forth by the 413.3(b) framework, the VTR project has to go through several Critical Decision (CD) points which represent various milestone and design maturity stages. In February 2019, the CD-0 was approved, which from a design perspective corresponded to a pre-conceptual design. In September 2020, just 19 months after the approval of CD-0, the CD-1 was approved for the VTR project in September 2020. From the design standpoint, this marked the completion of the conceptual design of the reactor. An overview of the core design approved as the conceptual core design is discussed here alongside with some of the performance characteristics expected from it.

The conceptual design for the VTR is a 300 MWth pool-type sodium-cooled fast reactor. The core contains a total of 313 assemblies, 66 of which are fuel drivers, and 10 are representative test locations. Ternary metallic fuel is used in the driver fuel and allows achieving a peak fast flux of about 4.5×10^{15} n/cm^2-s in the central test location, corresponding to material damage rate in excess of 50 dpa per year. The irradiation conditions offer large irradiation volumes with very high flux levels, as well as experimental flexibility through the use of cartridge loops. Cartridge loop experiments allow using a self-contained coolant, different from the reactor coolant, to provide prototypical irradiation conditions pertaining to other types of advanced reactors. In addition, VTR has a rabbit facility that permits insertion and removal of irradiation samples during operation.

The VTR conceptual core has been designed relying on past US expertise and demonstrated technologies. It is able to achieve the above-mentioned flux level under nominal conditions while remaining within acceptable thermal-hydraulic conditions and retaining an extremely favorable passively safe behavior. The main design and performance characteristics of the conceptual VTR core are presented in this paper.

2. AREAS OF FOCUS

Maturation of the VTR core design from the pre-conceptual design [2] to the conceptual design included incorporating many inputs and ensuring proper integration with the rest of the facility design. Of particular importance was giving due consideration to the inputs received from the fuel and material experts, iterating with the safety analysis team, the engineering team and of course the experimental team. As part of the core design activities, efforts have been focused on refining the core design characteristics, increasing the fidelity of the models and calculations performed.

The core performance characteristics were determined for a two representative states of the reference core design. One corresponding to a fresh core startup, and the other one corresponding to the core operating in an equilibrium mode. While the equilibrium conditions would never really be achieved, due to VTR being a test reactor and therefore experiencing different experimental loadouts, this is a good representation of a core configuration during normal operations and is the focus of this paper. Additional activities focused on assessing the impact that experiments would have on these performance characteristics, ensuring it will remain comfortably within the safety envelope of VTR.

Shielding assessments provided information on several shielding-related metrics, but the main focus for this phase of the project was on the secondary sodium activation. Secondary sodium activation is directly impacted by the in-vessel internal components arrangement and their specific geometry, as well as by the anticipated operating conditions of the reactor. It also bears an impact on the reactor operations as it can influence radiation zones and the time necessary before maintenance operations can be started.

Along the lines of operation considerations, some of the activities focused on characterizing the need for in-vessel storage of irradiated assemblies. Coolability of all irradiated assemblies needs to be ensured, without disrupting the reactor operations or core performance characteristics. These efforts are integrated with the plant engineering efforts, as this dictates the decay heat level acceptable for moving an assembly outside of the vessel.

In parallel to these efforts, verification and validation (V&V) of the codes used for the core design work was undertaken and major progresses were achieved. While V&V efforts will be on-going for the majority of the VTR project, including after the reactor starts operating, they were crucial during the conceptual phase to demonstrate that the confidence placed in the design codes used is justified. It also helped identifying specific validation needs that can be met through design of VTR-specific laboratory experiments.

3. VTR CONCEPTUAL CORE DESCRIPTION

The conceptual VTR core, shown in Figure 1, is filled with 66 fuel assemblies, 6 primary control rods, 3 secondary control rods, 114 radial reflectors, 114 radial shield reflectors, and 10 test locations. Out of the ten test locations, 6 are instrumented test locations and therefore have a fixed location in the layout, while the other four are non-instrumented test locations. While instrumented test assemblies can only be loaded in the identified positions, other types of assemblies, such as fuel assemblies or non-instrumented test assemblies, can be loaded in those positions too. The non-instrumented test locations could be positioned anywhere in the core, and their number could be increased or reduced. Any assembly can be replaced with a non-instrumented test assembly, granted the desired irradiation condition and cycle length can be achieved. This would mean different core layouts than shown in Figure 1, and therefore slightly different performance characteristics than discussed in this paper. The reference layout used here is selected as a representative configuration, and analysis of bounding configurations departing from this reference layout are the focus of on-going work. All assemblies rest on the lower support grip plate, which has receptacles matching the inlet coolant nozzles of the assemblies and are constrained radially by the core restraint system. The overall length of each assembly from the bottom of the lower shield to the coolant outlet is slightly under four meters.

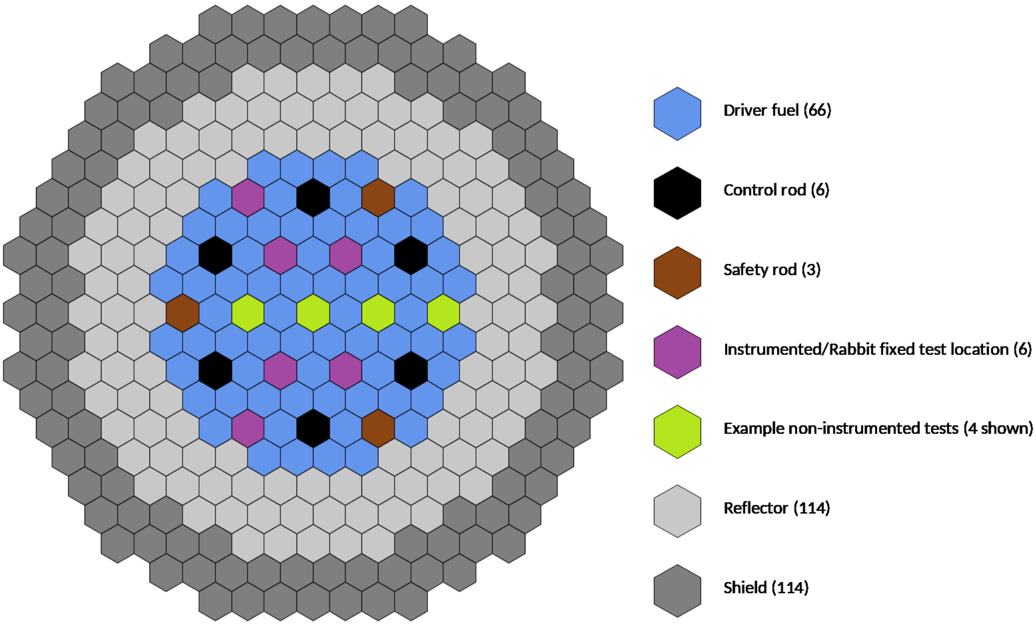

FIG 1. Conceptual core layout

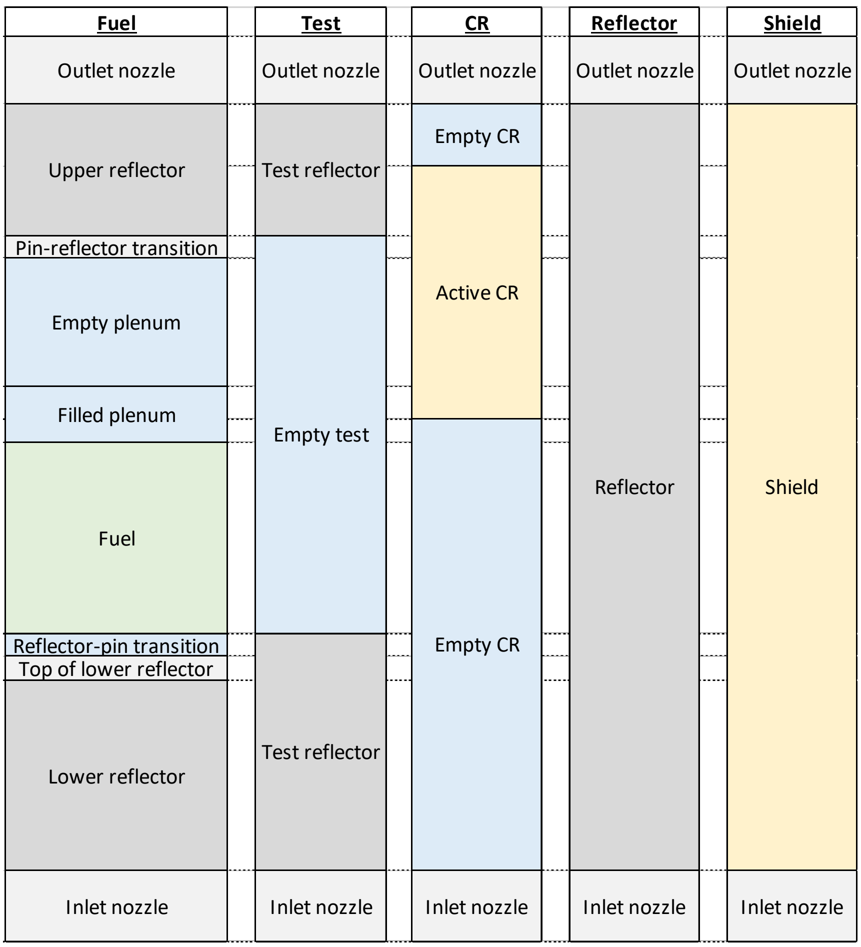

FIG 2. Assemblies axial layout

The axial layout of the six different types of assemblies is shown in Figure 2 for the "as modeled" conditions, which corresponds to the core being at full power and the radiation-induced swelling of the fuel to have reached its maximum.

3.1 Fuel assemblies

The reference core layout contains 66 fuel assemblies. Several different flow orifices will be used in order to adjust to the coolant flow rate in the various assemblies such as to smoothen the temperature distributions. In practice, this allows for more flow into the higher-power assemblies and less flow into the lower-power assemblies. The specific flow orifice design is still being updated to optimize the thermal-hydraulics performance.

The lower and upper reflectors are "block-type" reflectors. Such reflector geometry, schematically shown in Figure 3, has been previously used in EBR-II. The advantage of the block geometry over a bundle of reflector rods is a lower pressure drop resulting from a larger hydraulic diameter. An additional benefit is that a larger volume fraction of steel can be accommodated, slightly reducing the number of neutrons escaping the fueled region. The exact geometry of the block-type reflector is being refined based on thermal-hydraulics experimental results. The desired volume fractions of coolant and HT9 steel in the reflector region are 30% and 70%, respectively. Parametric and computational fluid dynamic calculations showed that higher volume fractions of steel does not provide any significant neutronic benefit but would lead to unnecessary large pressure drop. The same parametric study indicated that there is little neutronics impact from varying the specific geometry of the reflectors.

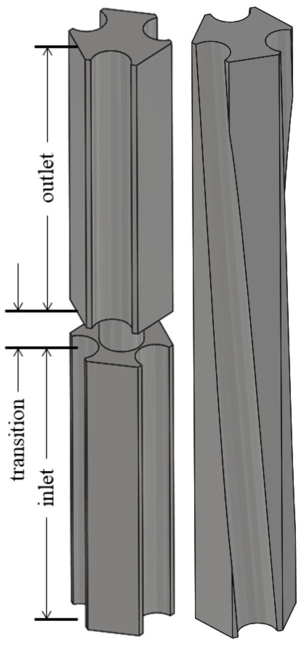

FIG 3. Schematic View of the Block-Type Axial Reflectors Options

Each fuel assembly contains 217 fuel rods, arranged on a triangular pitch in a hexagonal array within the fuel duct. Spacing is maintained between the fuel pins by a steel wire wrapped around the pins. The fuel rods are encapsulated with HT9 cladding, closed at both ends with an HT9 plug. Each rod contains slugs of ternary metallic fuel with a fuel column height of 80 cm. Sodium inside the fuel rod forms a heat-transfer bond between the fuel column and the cladding. Above the fuel, within the rod, there is an 80 cm fission gas plenum initially filled with inert gas. The reference fuel for this phase of the project is a ternary metallic alloy, U-20Pu-10Zr. Reactor-grade plutonium is used alongside enriched uranium with 5at% 235U to achieve the desired performance [3]. The smeared density (i.e., the areal density of the fuel cross section within the cladding inside diameter in the as-fabricated fuel rod) is 75%.

3.2 Control rods

The reference core design contains six primary control rods and three secondary control rods. The primary rods are moved during normal operation to adjust for changes in reactivity and to control the power level of the core. The secondary rods are fully withdrawn from the fuel region during normal operation and are fully inserted into the core when the reactor is shutdown in order to provide additional shutdown margin, for example during refueling operations. Each control rod is operated independently from other control rods.

A representative schematic of the control rods is shown in Figure 4. An assembly duct is placed in the core where control rods are located and does not move. An inner duct containing the absorber pins is located inside the fixed assembly duct. The inner duct is connected to the control rod driveline, which allows it to move axially as needed for reactivity control. A gap between the inner duct and the fixed assembly duct allows free movement of the inner absorber assembly regardless of mechanical deformation or thermal expansion of the assembly.

The inner duct contains a bundle of 37 absorber rods. The space between the rods is maintained by a steel wire wrapped around the rods. The absorber material is boron carbide (B_4C) encapsulated in HT9 cladding. Height of the active absorber region is more corresponds to the anticipated length of the fuel column after elongation plus 20 cm. Natural boron is used as the absorbing material and provides sufficient reactivity worth to achieve the required shutdown margins.

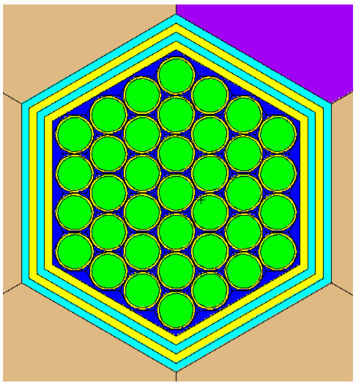

FIG 4. Control Rod Assembly, Plane View (Green: Absorber, Yellow: Steel, Blue & Cyan: Sodium)

3.3 Radial reflector assemblies

There are 114 radial reflector assemblies in the reference core layout. They are composed of a tight bundle of HT9 rods packed together (i.e. no wire wrap) such as to maximize the volume fraction enclosed in a HT9 duct. Alternatively, a block-type reflector like that described for the fuel assembly axial reflectors could be used for the radial reflectors as well. The desired steel and coolant volume fractions are approximately 80% and 20%, respectively.

3.4 Radial shield assemblies

There are 114 radial shield assemblies in the reference core layout. They are composed of a bundle of absorber rods spaced with a wire-wrap enclosed in a HT9 duct. The rods are made of a HT9 cladding containing B_4C pellets. The desired coolant, steel, and absorber volume fractions are 24%, 28% and 40%, respectively. The remainder is the bond gas (or fill gas) within the cladding. It is intended to use some of the shield locations in the outermost row as in-vessel storage (IVS) for used fuel and other assemblies discharged from the core, until they are removed from the reactor vessel or reinserted into in the core.

3.5 Instrumented test locations

The reference VTR core layout shows the six instrumented test locations represented by the purple positions in Figure 1. These positions need not be occupied by an instrumented test or a rabbit and are available for a non-instrumented test assemblies, fuel assemblies, or dummy assemblies. However, instrumented test assemblies will only be able to be loaded in one of these six identified locations. This restriction is resulting from the instrumented test assembly's hardware requiring a penetration in the cover head.

Given that test assemblies can come in a variety of designs [4], based on the type of experiment loaded, their contents are not explicitly modeled in the reference core design. Instead, the test assemblies are modeled as empty ducts containing the same lower and upper reflectors as the fuel assemblies, and only sodium in the central region.

3.6 Non-instrumented test locations

The reference VTR core layout shows four non-instrumented test locations that are represented by the green positions in Figure 1. Contrary to the instrumented test assembly locations, non-instrumented test assembly locations are not fixed, and such assemblies can be loaded anywhere in the core with the exception of the control rod locations. The four locations shown in green here are illustrative of a possible configuration. It is anticipated that 15 non-instrumented test assemblies could routinely be concurrently present in the core, with several of these being located in the reflector region.

The non-instrumented test locations shown are modeled in the same manner as the instrumented test locations (i.e., reflected sodium volumes in the core region).

4. VTR CONCEPTUAL CORE PERFORMANCE CHARACTERISTICS

The performance characteristics of the conceptual core design are discussed in this section. These include the reactor physics characteristics, fuel requirements, thermal-hydraulics characteristics, reactivity coefficients, and shutdown worth requirements.

4.1 Reactor physics performance

The performance characteristics of the reference VTR core have been determined and the values obtained for an equilibrium cycle are presented here. The conceptual fuel loading strategy is illustrated in Figure 5. The 12 central most fuel assemblies (fueled assemblies in rows 1 to 3) remain in the core for 3 cycles, the next 18 fuel assemblies (row 4) remain in the core for 4 cycles, the following 12 fuel assemblies (in row 5) remain in the core for 5 cycles, and the remaining 24 assemblies (in row 6) remain in the core for 6 cycles. This is identified in Figure 5 by the first number in each assembly. The second number provided for each assembly indicates when each assembly is replaced. For instance, "6-4" indicates that this assembly remains in the core for 6 cycles and is replaced every 4th cycle out of 6. All assemblies having the same identifier are to be replaced at the same time.

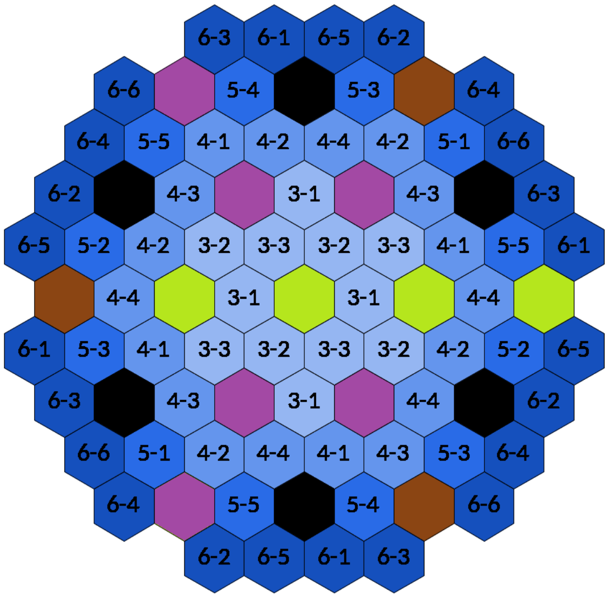

FIG 5. Conceptual Fuel Management Overview

The homogeneous equilibrium results were obtained using transport theory with the P5 flux approximation with the Argonne Reactor Computation code suite, which includes MCC3, DIF3D and REBUS. The reactor physics performance characteristics are provided in Table 1 for an 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 maximal value achieved locally in the core, and not over a large volume. The maximum absolute/relative power variations correspond to the largest absolute/relative power variations observed in any fuel assembly between the beginning of a cycle (BOC) and the end of a cycle (EOC).

Table 1. Reactor Physics Performance Characteristics for the CD-1 Reference VTR Core

Characteristic	**Unit**	**Value**
Core power	MW_{th}	300
Cycle length	EFPD	100
Number of batches	-	3 to 6
Plutonium concentration	wt.%	20.1%
Uranium enrichment	at.%	5%
Maximum excess reactivity	pcm	2186
Burnup reactivity swing	pcm	2186
Test peak fast flux at BOC	$\times 10^{15}$ n/cm^2-s	4.34
Test peak fast flux at EOC	$\times 10^{15}$ n/cm^2-s	4.23
Absolute peak fast flux at BOC	$\times 10^{15}$ n/cm^2-s	4.54
Absolute peak fast flux at EOC	$\times 10^{15}$ n/cm^2-s	4.43
Average assembly power	MWth	4.55
Maximum assembly power at BOC	MW_{th}	6.45
Maximum assembly power at EOC	MW_{th}	6.17
Fuel assemblies/year	-	44.7
Heavy metal charge/year	kg/year	1788.7
Uranium required/year	kg/year	1388.4
Plutonium required/year	kg/year	400.3
Average discharge burnup	GWd/t	50.3
Assembly-averaged peak discharge burnup	GWd/t	52.5
Peak discharge burnup	GWd/t	61.0

The peak fast flux in each assembly is shown in Figure 6 at EOC. The axial fast flux distribution is shown at EOC in Figure 7 for a fuel assembly in the second row near the core center, for a fuel assembly in the sixth row at the core periphery and for a reflector assembly in the second row of reflectors. The locations for which these curves are provided correspond to the circled assemblies in Figure 12. These axial distributions are mostly identical between BOC and EOC as well as between cycles.

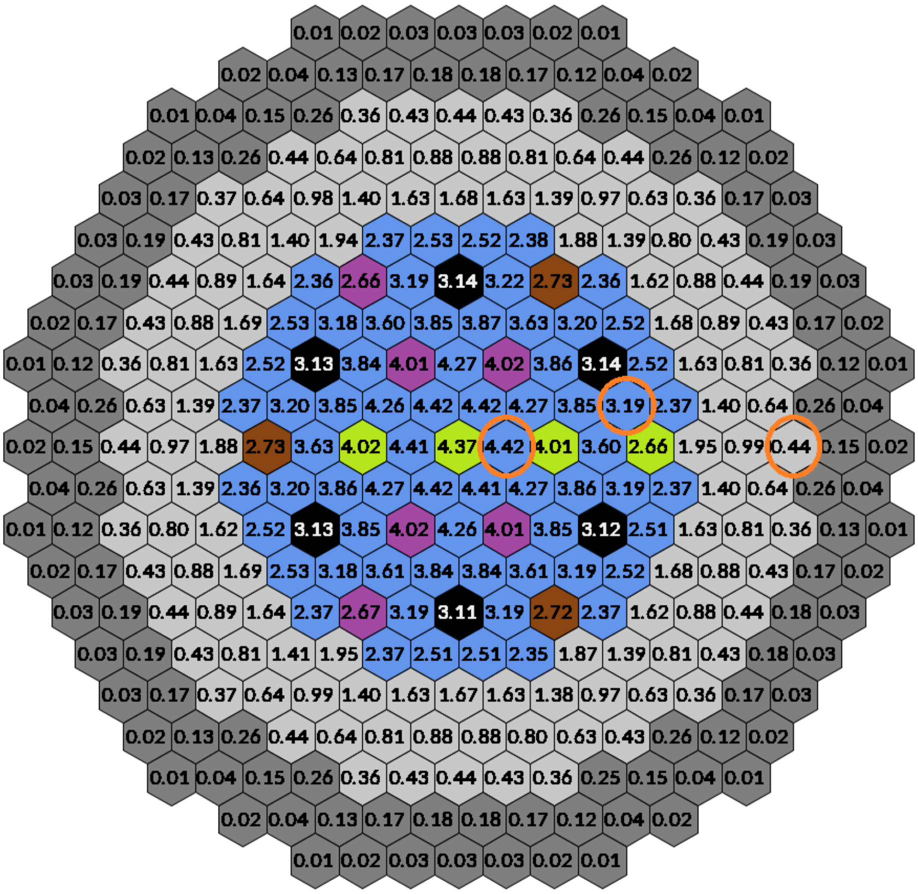

FIG 6. Peak Fast Flux (x1015 n/cm2-s) Distribution at EOC

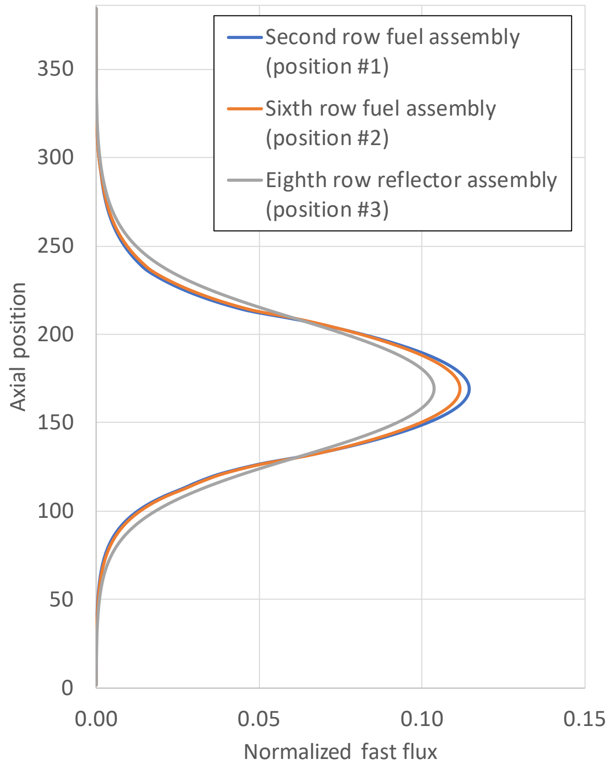

FIG 7. Axial Flux Distribution in Various Assemblies at EOC

4.2 Control rods worth

The reactivity worth of the control rods has been determined for the primary and secondary control systems separately. The total systems worths are obtained by calculating the reactivity difference between the control rods being fully withdrawn above the top of the fuel region, and the control rods being fully inserted past the bottom of 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 inserted about 32 cm into the fuel region. For the secondary system evaluation, this means one rod remains fully withdrawn from the fuel region. The reactivity worths are summarized in Table 2 at BOC.

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

CR system	**Primary**	**Secondary**
All rods in, pcm	6598	2142
All rods in, $	17.88	5.80
All rods in minus one, pcm	5464	1390
All rods in minus one, $	14.81	3.77

4.3 Reactivity coefficients

Reactivity coefficients have been determined for the core at equilibrium. The assumptions used to for the calculations are consistent with the expectations from the VTR safety analysis team. The values calculated using the homogeneous fuel management approach are summarized in Table 3 at BOC and EOC.

Table 3. Reactivity Coefficients at BOC and EOC

Parameter	BOC		EOC	
βeff	369 pcm		369 pcm	
Prompt lifetime	3.87E-07 s		4.14E-07 s	
Radial expansion	-757.6 pcm	-0.391 c/K	-765.7 pcm	-0.395 c/K
Axial expansion (fuel only)	-340.9 pcm	-0.111 c/K	-293.2 pcm	-0.096 c/K
Fuel density	-515.9 pcm	-0.742 c/K	-503.5 pcm	-0.724 c/K
Structure density	304.6 pcm	-0.061 c/K	322.4 pcm	-0.064 c/K
Sodium density	-25.3 pcm	-0.192 c/K	-25.9 pcm	-0.197 c/K
Sodium void (fuel and above)	-681.6 pcm	-1.85 $	-744.1 pcm	-2.02 $
Doppler	-209.2 pcm	-0.084 c/K	-229.0 pcm	-0.092 c/K

4.4 Thermal hydraulics performance

The thermal-hydraulic performance of the conceptual VTR core has been determined using SE2-ANL. Values have been calculated for both the BOC and EOC conditions, and with modeling the primary control rods at their critical position. Three orifice zones are used for the fuel assemblies. The orifice zones for the non-fuel assemblies have arbitrarily been fixed to seven but will be refined in the next phase of the project. With no planned fuel shuffling, the flow allocation in a given assembly is fixed for the entirety of its residence time in the core. With the power level varying of the course of a cycle and between cycles, this will lead to small deviations from the predicted temperatures.

The fuel orificing strategy used is shown in Figure 8. It corresponds to the arrangement which will yield the lowest nominal peak cladding temperatures in each group. Future efforts will further optimize the flow allocation such as to obtain the best tradeoff between minimizing the cladding temperature and the fuel temperature. The per-assembly flow rates in the three orificing zones are about 30 kg/s, 23 kg/s and 16 kg/s. The flow rate in every assembly of a given group is identical. The overall core flow rate to achieve the desired temperature rise of 150°C across the core is 1564 kg/s. A summary of the thermal-hydraulic characteristics in each group is provided in Table 4.

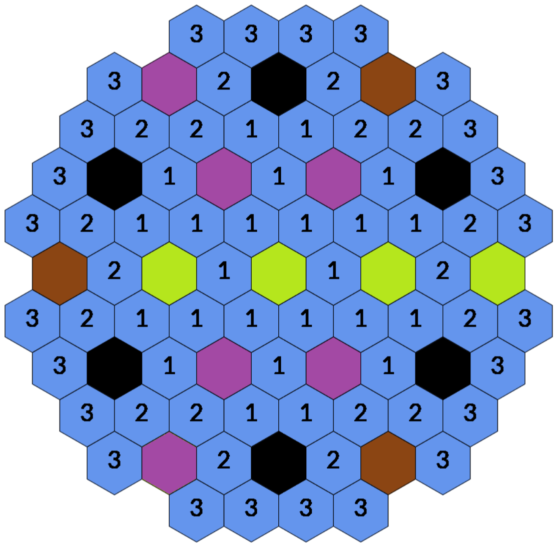

FIG 8. Orifice Groups Arrangement for the Reference VTR Core

Table 4. Summary of Thermal-Hydraulics Characteristics with Three Orifice Groups

Orifice group		Flow, kg/s	Power, MW		Coolant bulk temp., °C		Coolant peak temp., °C		Peak clad. temp., °C		Fuel CL temp., °C		Press. drop, MPa	Velocity, m/s
			BOC	EOC	BOC	EOC	BOC	EOC	BOC	EOC	BOC	EOC		
1	Ave.	29.8	5.69	5.55	500	497	513	508	532	526	788	774	0.44	9.37
	Max.		6.71	6.42	526	518	540	531	559	551	840	820		
2	Ave.	23.2	4.33	4.36	497	498	511	511	529	529	730	725	0.29	7.29
	Max.		5.12	5.07	521	520	541	537	561	558	786	776		
3	Ave.	15.9	3.16	3.27	505	510	517	524	531	541	661	676	0.15	5.00
	Max.		3.45	3.68	519	526	533	543	546	562	688	709		

5. CONCLUSION

From the completion of CD-0 in 2019 to now, tremendous progress has been made in maturing the VTR design and receiving CD-1 approval in September 2020. As part of these accomplishments, a reference conceptual core design has been established for VTR. The main design and performance characteristics of this core were discussed in this paper and are the culmination of many analyses and design choices, all aimed at achieving the VTR mission.

The conceptual core contains 313 assemblies out which 66 are fuel assemblies, which enable achieving peak fast fluxes as high as 4.5x1015 n/cm2-s and large effective irradiation volumes. The ternary metal fuel used, U-20Pu-10Zr, is intended to be discharge with burnup levels of about 50 GWd/t, which is well within the experience basis of previously obtained with this type of fuel. The large reflector region allows to extra irradiation testing space and the large shield region allows for low secondary sodium activation and potential temporary assembly storage locations.

The VTR core design continues to be refined and characterized, working toward completion of the next Critical Decision milestone and eventually starting operations.

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