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DESIGN STUDIES FOR A MULTIPLE APPLICATION THERMAL REACTOR FOR IRRADIATION EXPERIMENTS (MATRIX)

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ABSTRACT

The Advanced Test Reactor (ATR) is a high power density test reactor specializing in fuel and materials irradiation. For more than 45 years, the ATR has provided irradiations of materials and fuels plus secondary missions such as the production of radioisotopes. Should unforeseen circumstances lead to the decommissioning of ATR, the U.S. Government would be left without a large-scale materials irradiation capability to meet the needs of its nuclear energy and naval reactor missions. In anticipation of this possibility, work was performed under the Laboratory Directed Research and Development (LDRD) program to investigate test reactor concepts that could serve the current missions of the ATR along with an expanded set of secondary missions. A survey was conducted in order to catalogue the anticipated needs of current and potential customers. Concepts were then evaluated to fill the role for this reactor, dubbed the Multi-Application Thermal Reactor for Irradiation eXperiments (MATRIX). This evaluation indicates that the baseline MATRIX design described herein can achieve longer cycle lengths than ATR given a particular batch scheme and using uranium enriched to less than 20%. The volume of test space in In-Pile-Tubes (IPTs) in MATRIX is larger than those in ATR with comparable magnitude of neutron flux. Furthermore, MATRIX has a greater number and volume of irradiation spaces having high fast neutron flux than ATR. From the analyses performed in this work, it appears that a new materials test reactor, based upon the MATRIX concept, can meet the anticipated needs of current and new high flux irradiation programs for the foreseeable future.

Key Words: **ATR, Flux trap, Test Reactor, LEU, MATRIX**

1. INTRODUCTION

First achieving criticality in 1967, the Advanced Test Reactor (ATR) is a high power density test reactor specializing in fuel and materials irradiation and located on the Idaho National Laboratory (INL) site. Featuring nine large flux traps that can accommodate independently-cooled high-pressure In-Pile Tubes (IPT – see Figure 1), the ATR was originally designed for irradiation testing in support of the Naval nuclear propulsion program. Over the years and the reactor's customer base has expanded to include the Department of Energy's (DOE) Office of Nuclear Energy (NE), US and international nuclear power vendors, radioisotope production, and more. Over 40 years old, ATR is still reliable and effective but increasingly expensive to maintain and operate every year. The sophisticated test hardware and experiments required of recent fuel and

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* Retired from Idaho National Laboratory

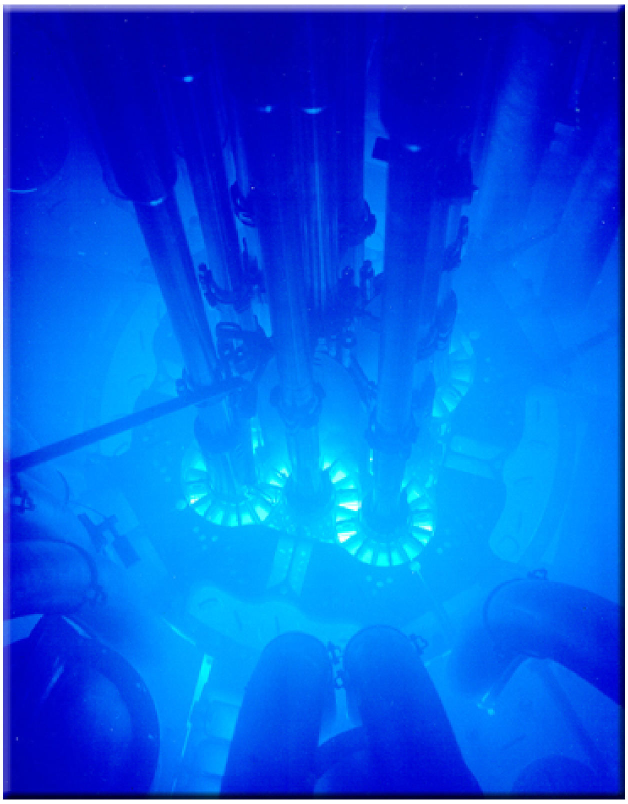

Figure 1: The top of the ATR core as seen through a viewing port. The large pipes extending upward are InPile Tubes installed in the flux traps.

materials irradiation campaigns are limited by the safety limits and operating envelope of this aging facility, which are becoming harder to meet as equipment reliability is diminished. Furthermore, the fabrication and transport of its highly enriched uranium (HEU) fuel is deemed a proliferation concern. Under the Global Threat Reduction Initiative (GTRI), ATR and the other high flux test reactors in the US fleet must eventually be converted to low enriched uranium (LEU) in order to continue operating. Studies are underway to assess the costs and technical viability of this fuel conversion. One foreseeable outcome is that the cost of upgrades, fuel conversion, and more frequent LEU fuel loadings needed to keep ATR viable will exceed those of building and operating a replacement reactor. In anticipation of this scenario, work was commenced under the Laboratory Directed Research and Development (LDRD) program to investigate test reactor concepts that could satisfy the current missions of the ATR. Naturally, the needs of other value-adding missions were considered as part of the initial design study.

In the early 1990's, the Idaho National Laboratory (then called Idaho National Engineering Laboratory, or INEL) undertook a similar study aimed at developing and evaluating concepts for a new test reactor. This reactor was envisioned to fulfill a range of customer needs in anticipation of the possible decommissioning of several aging test reactors in the U.S., including the ATR. A cataloguing and ranking of anticipated user needs was performed and a wide variety of reactor concepts were evaluated. The name given to the family of reactors conceptualized was the Broad Application Test Reactor (BATR). [1-4]

By the end of the BATR study in 1993, two reactor concepts had emerged as leading candidates. [4] Both called for plate fuel and were moderated by beryllium and/or D_2O and cooled by light water with a rated power of 250 MW_{th}. Several independently cooled IPTs were also called for in both designs. The "Multiple-Annular" (M-A) core featured arcuate fuel assemblies similar to those of the ATR, each surrounding large cylindrical flux traps supporting different loop configurations. The unperturbed spectrum in each large irradiation position could be tailored to user needs through the addition of dedicated reflectors and filters. The "Modular-Hexagonal" (M-H) core featured multiple cylindrical fuel assemblies arranged in a hexagonal matrix among an array of uniquely configured flux traps. The entire array of fuel and flux traps was surrounded by a common beryllium primary reflector, a pressure boundary, and a D_2O secondary reflector. Though each of the two concepts was judged to have advantages and disadvantages, the M-A concept was judged to be slightly superior in part because it had significantly higher fast neutron flux in test locations. The M-H core, however, also ranked highly because of its interchangeable grid, which would facilitate core reconfiguration. A third concept, which was also proposed but not analyzed in detail during the BATR project, was an upgraded ATR. This was an evolutionary ATR design calling for several key design changes including a power uprate, a change in coolant direction from downflow to upflow, and an increase in the beryllium reflector size.

In the 1990's, it was assumed that proposed research and test reactors operated by the Department of Energy (DOE) could use highly enriched uranium (HEU) fuel. Because this is no longer considered an option, any new research reactors proposed will likely be required to use low-enriched uranium (LEU) fuel, which contains less than 20% ^{235}U by mass. This presents a major departure between the current work and the BATR project.

2. OBJECTIVES

2.1. Goals of Project

Commenced in fiscal year (FY) 2012 and running through the end of FY2013, this project had the following objectives; 1) update the anticipated customer needs of a new test reactor from the BATR project, 2) re-evaluate the BATR concepts of the 1990's and investigate LEU options for them, 3) propose modifications to the BATR concepts to better meet the anticipated customer needs, and 4) analyze and compare the performance of the proposed designs.

This paper first presents the anticipated customer needs of such a replacement to ATR followed by a description of the lead candidate design emerging from the two-year project. Though only one concept is described in this paper, other designs were evaluated and were also found to show promise. Therefore, the downselection to the design of present focus should not be considered final. Other concepts are catalogued in Reference [5]. These include four main concepts having different types of fuel elements (e.g., square, cylindrical, and arcuate). The project adopted the acronym MATRIX (Multiple-Application Thermal Reactor for Irradiation eXperiments) to refer to the family of reactors considered. The lead (here called baseline) design is a particular variant of the MATRIX concept having cylindrical fuel elements.

2.2. Anticipated Customer Needs

The needs of the primary ATR customers (NE and NR), gathered through a survey and a meeting held at INL with potential stakeholders in March of 2012, are here summarized [6]:

- High availability/capacity factor (>70%)
- Must be licensed by the Nuclear Regulatory Commission (NRC)
- High neutron flux (total flux $>10^{15}\ n\cdot cm^{-2}\cdot s^{-1}$) in a large sample volumes
- Multiple independently cooled and instrumented IPTs required (variable dimensions)
- Modular type core (re-conformable, flexible) to accommodate changing missions
- Hydraulic shuttle irradiation system (rabbit)
- Easy access to loops, rabbit tubes, neutron beams, etc.
- Primarily thermal neutron flux spectrum, but some positions capable of fast and epithermal spectra
- Minimum waste stream effluents/environmental impact

- Primary coolant flow in upward direction through the core to prevent flow reversal in transition to natural convection
- Capability of producing power excursions within test assemblies (transient testing)

Assuming that the primary mission can be so served, a number of secondary missions were identified for possible, albeit not essential, implementation into the MATRIX design. These include neutron imaging (e.g. radiography), neutron scattering and activation experiments, and radioisotope production (e.g. ^{99}Mo, ^{60}Co, ^{238}Pu, ^{252}Cf, etc.).

The spectral and operational needs of these missions vary, placing significant demands on the design of the facility in addition to those listed above. Whether or not a single reactor can serve some or all of these secondary missions will depend on the modularity and configurability of the reactor (the reflector in particular) as well as the functionality of the experimental support systems. Thus far, the spectral characteristics of the reactor concepts were the focus of the work, with specific needs of potential secondary missions left for future work.

2.3. Calculation Methodology

Reactor physics analyses were performed using the Monte Carlo depletion code Serpent 2 [7]. Developed at the VTT Technical Research Center in Finland, Serpent 2 was selected based on modern programming, ease of use, built-in depletion, automatic fission source development, and greater speed than MCNP [8].

A sophisticated reactivity control scheme will be required for such a reactor to maintain criticality and power shape over short and long time intervals. Design of such a scheme was beyond the scope of this work so, for the purposes of evaluation and comparison, each simulation began with an unpoisoned (supercritical) core at beginning of life (BOL) and depleted until k_{eff} fell below 1.0. Since this is the point at which the reactor is just critical, neutron flux tallies were taken from the time step closest to this condition. The number of days of burnup required to reach this point (single batch reactivity-limited burnup) is called B_1. This value can be used to estimate and compare the achievable burnup between reactors and variations on a single reactor concept. This metric is also used to compare achievable cycle length to that of the ATR.

3. LEAD CANDIDATE MATRIX DESIGN

3.1. Baseline Concept Description and Performance

Table 1 gives parameters for the MATRIX reactor core with cylindrical fuel elements. Some of these parameters (e.g. thermal power rating, primary coolant conditions, fuel plate thickness) were taken directly from the ATR in order to narrow the design space initially and isolate the preliminary analysis to neutronics only.

Table 1. Parameters for baseline Cylindrical MATRIX core concept

Parameter	**Value**
Primary Coolant System Parameters	
Rated Thermal Power (MW_{th})	250
Primary Coolant Inlet Temperature (°C)	52
Primary Coolant Outlet Temperature (°C)	89
Primary Coolant Mass Flow Rate (kg/s)	1600
Primary Coolant Pressure (MPa)	2.3
Fuel Element Parameters and Geometry	
Active Fuel Height (cm)	120
Number of Fuel Elements	24
Fuel Plates Per Assembly	10
Inner diameter of innermost coolant channel (cm)	3.95
Outer diameter of outermost coolant channel (cm)	10.85
Width of coolant channels between fuel plates (cm)	0.2
Fuel plate thickness (cm)	0.125
Fuel meat thickness (cm)	0.05
Compositions	
Fuel Meat Material	U-10Mo
^{235}U density (g/cm^3)	3.00
Total U density (g/cm^3)	15.2
^{235}U Enrichment (wt. %)	19.7
Cladding Composition	Zircaloy-4
Central Cylinder Composition	Zircaloy-4
Inventory	
Total ^{235}U per assembly (kg)	4.2
Total ^{238}U per assembly (kg)	17.0
Total ^{235}U in core (kg)	100
Total ^{238}U in core (kg)	408

Figure 2 shows a schematic diagram of the MATRIX core and Figure 3 shows a diagram of a single fuel element. The core is comprised of cylindrical fuel assemblies of plate fuel arranged in an aluminum core rack. The core rack is assumed to contain 6.3% by volume water at primary coolant conditions. This was modeled as a homogeneous mixture of aluminum and water. As can be seen in the figure, the fuel elements are modeled as single 360º elements with no vertical ribs spanning the length of the assembly for structural support. This was a simplifying assumption used in parametric studies. Criticality or other fuel management constraints may dictate that each 360º cylinder be subdivided into multiple assemblies. For example, three assemblies, each spanning 120º angle rather than a single element. The parametric studies presented here were performed assuming the layout shown in Figure 2.

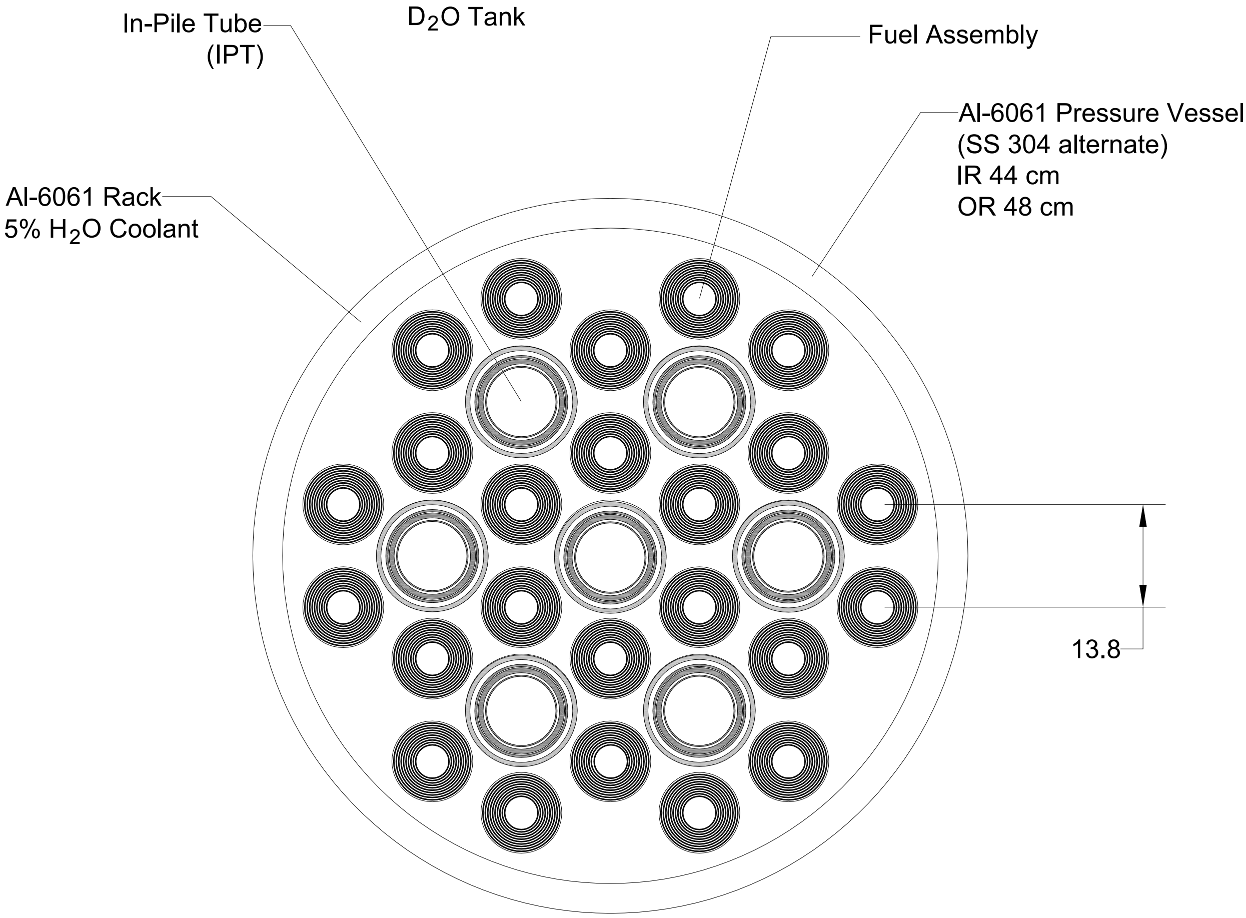

Figure 2: Layout of MATRIX core and reactor vessel (dimensions in cm).

The core rack containing the fuel elements is surrounded by a pressure vessel, nominally constructed of aluminum alloy 6064 (Al-6064) with stainless steel 304 (SS304) as a fallback alternative. The inner radius of the pressure vessel is 44 cm and it is 4 cm thick. Outside the pressure vessel is a tank of D_2O at atmospheric pressure having outer dimensions the same as other variants.

The fuel plates nominally have the same thicknesses as the other concepts and the HEU fuel of the ATR (1.25 mm thick plates with 0.5 mm thick meat). The baseline fuel form for MATRIX is U-10Mo having uranium density of 15.2 g/cm^3 and Zircaloy cladding. The innermost coolant channel has an inner diameter of 3.95 cm and the outermost coolant channel has an outer diameter of 10.85 cm. Inside the fuel assembly is a location that can either contain an irradiation test, a control rod or rod follower, or a simple filler to displace coolant water. Unless otherwise specified, this space is modeled as Al-6061. The core has an active fueled height of 120 cm, close to the 4 foot (121.92 cm) height found in ATR. The total mass of uranium in the baseline Cylindrical MATRIX core is 508 kg, 100 kg of which is ^{235}U.

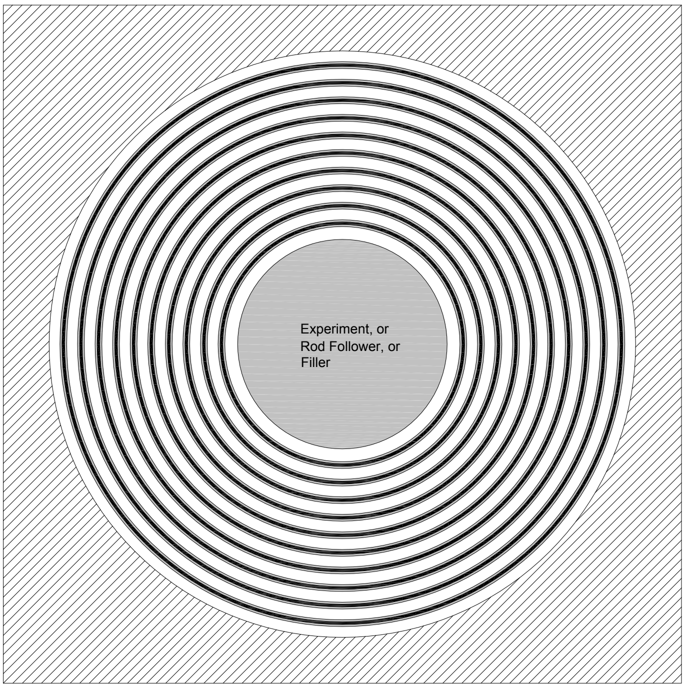

Figure 3: Schematic showing arrangement of fuel plates in cylindrical fuel type.

Dispersed among the fuel assemblies are seven IPTs, each corresponding to a flux trap. Figure 3 shows a cross section of a single flux trap and its associated IPT. These have similar features to those of the ATR with the sizes of the piping adjusted to fit the size of the flux traps in the baseline MATRIX design. The outermost layer is an aluminum baffle. In the ATR, this baffle serves as the structural feature from which fuel elements hang, held in place by gravity and hydrodynamic forces. The same type of fuel assembly support will not be possible in any of the MATRIX designs as currently envisioned. This is because 1) the coolant will be in upflow and simply hanging the elements from a baffle will not be sufficient to keep them in place and 2) the MATRIX core concept has fuel elements that do not circumscribe the flux trap structures as they do in ATR. Nevertheless, some material is included in each of the IPTs here as a placeholder for any structure that would need to be present. These may be removed or modified as design details are added. Inside the baffle are (from outermost to innermost) a water gap representing the IPT coolant inflow, a stainless steel (SS304) insulation tube, a helium gap, the SS304 pressure tube, another water annulus representing the IPT outflow, and a SS304 flow tube separating the coolant from the cylindrical test region (Figure 4). Assumed for baseline calculations to contain pure Al-6061, this test space has a diameter of 9.3 cm. This is a similar diameter to that of the large northwest flux trap in ATR. [9]

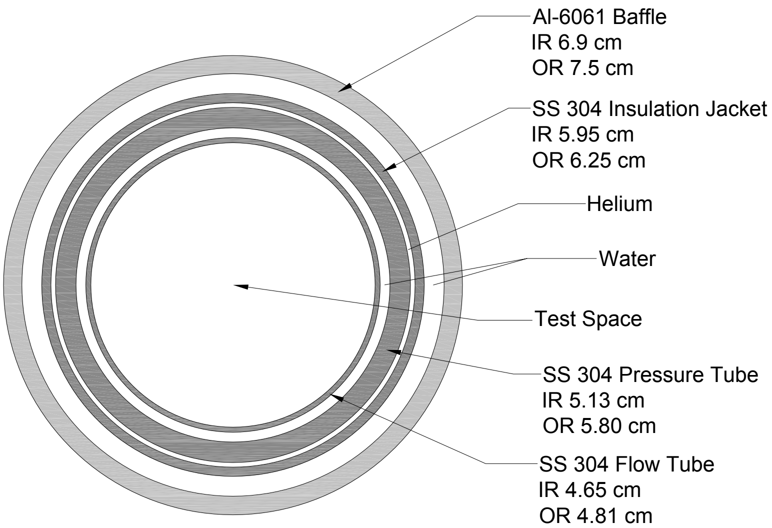

Figure 4: Diagram of in-pile tubes (IPTs).

Above and below the fueled portion of the fuel elements are assumed to be regions of plates having no fuel meat followed by end boxes. These are modeled as the homogenized regions shown in Figure 5. The end boxes are modeled by assuming them to be similar to the representation of end boxes used in ATR modeling. [9]

Figure 6 shows an above view of the core, pressure vessel, and the D_2O reflector tank. Surrounding the pressure vessel, the reflector tank contains D_2O assumed to be at atmospheric pressure and near room temperature (assumed P=1 atm and T=300 K). This feature was envisioned to provide a versatile space of very high thermal flux for irradiation experiments, beam tubes, etc. This feature is compared to other options for reflector tank contents in Section 3.2 of this report. The numbered reflector tally locations in Figure 6 give the distances from core center where flux values are tabulated. Figure 7 shows a vertical cross section of the MATRIX core design. From this, one can see how the IPTs extend from the bottom to the top of the reflector tank while the active core and fuel assemblies are centered axially in the tank. For neutronic modeling purposes, beyond the outer boundary of the D_2O tank was considered void.

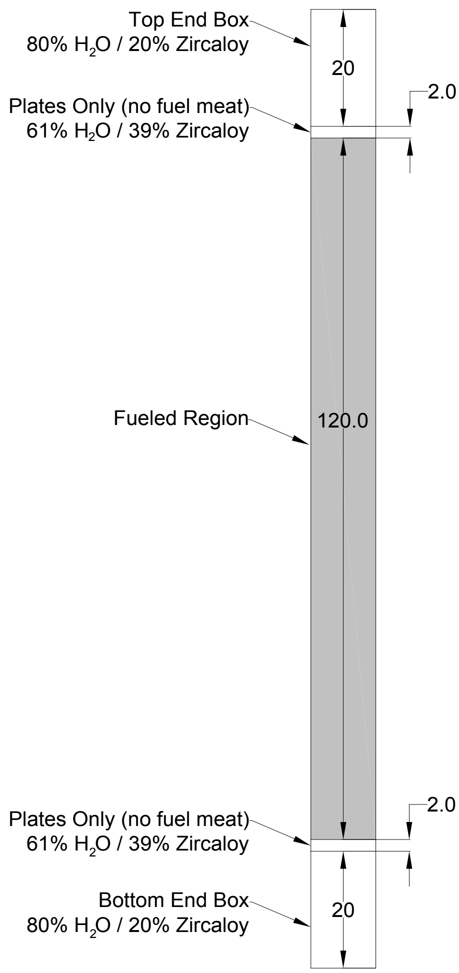

Figure 5: Diagram of axial sections of fuel assemblies (dimensions in cm).

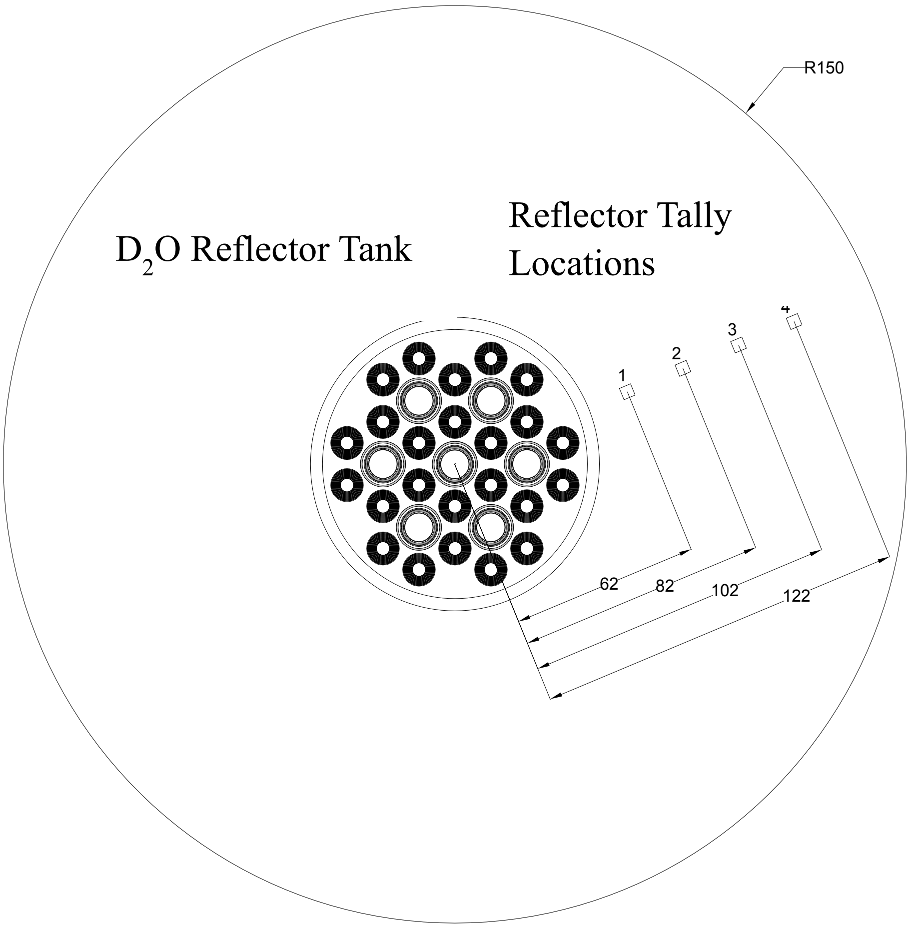

Figure 6: Above view of cylindrical concept with reflector tally locations (dimensions in cm).

Given this baseline core arrangement, the fresh-core initial k_{eff} (no xenon, no samarium) is 1.18 and B_1 is 100 days. In a 3-batch arrangement with each batch approximately equal in number of elements, this would give an approximately 50-day cycle at 250 MW_{th}. Subsequent calculations where three 0.5 cm thick vertical ribs are included in each fuel element to provide structural support indicated that B_1 would be reduced to approximately 80 days, giving a 3-batch cycle length of 40 days at 250 MW_{th}. Table 2 shows neutron flux calculated by Serpent 2 for various locations in the core broken down into energy groups. The energy groups are fast (>1 MeV), thermal (<0.625 eV), and total. The tallies were taken in 10 equal axial segments (12 cm each) extending from bottom to top of the 120 cm tall active core height. The center two segments together span 24 cm of axial height and form the "peak" flux values and the "axially averaged" values are taken as the average over the height of the active core. The center and one peripheral flux trap are tallied along with the center of an inner-ring fuel assembly. These calculations were performed assuming 100% aluminum fill in the test space within each IPT. This gives an estimate of the upper bound of fast neutron flux that is achievable and attempts to form a fair comparison to ATR (Section 4). If higher thermal neutron flux is desired, this can be moderated accordingly to produce it, although the amount of cooling water in the test location is constrained by loop void reactivity limits. Therefore, the magnitude of the fast neutron flux in the test locations is used as only an approximate measure of the achievable thermal neutron flux in these positions.

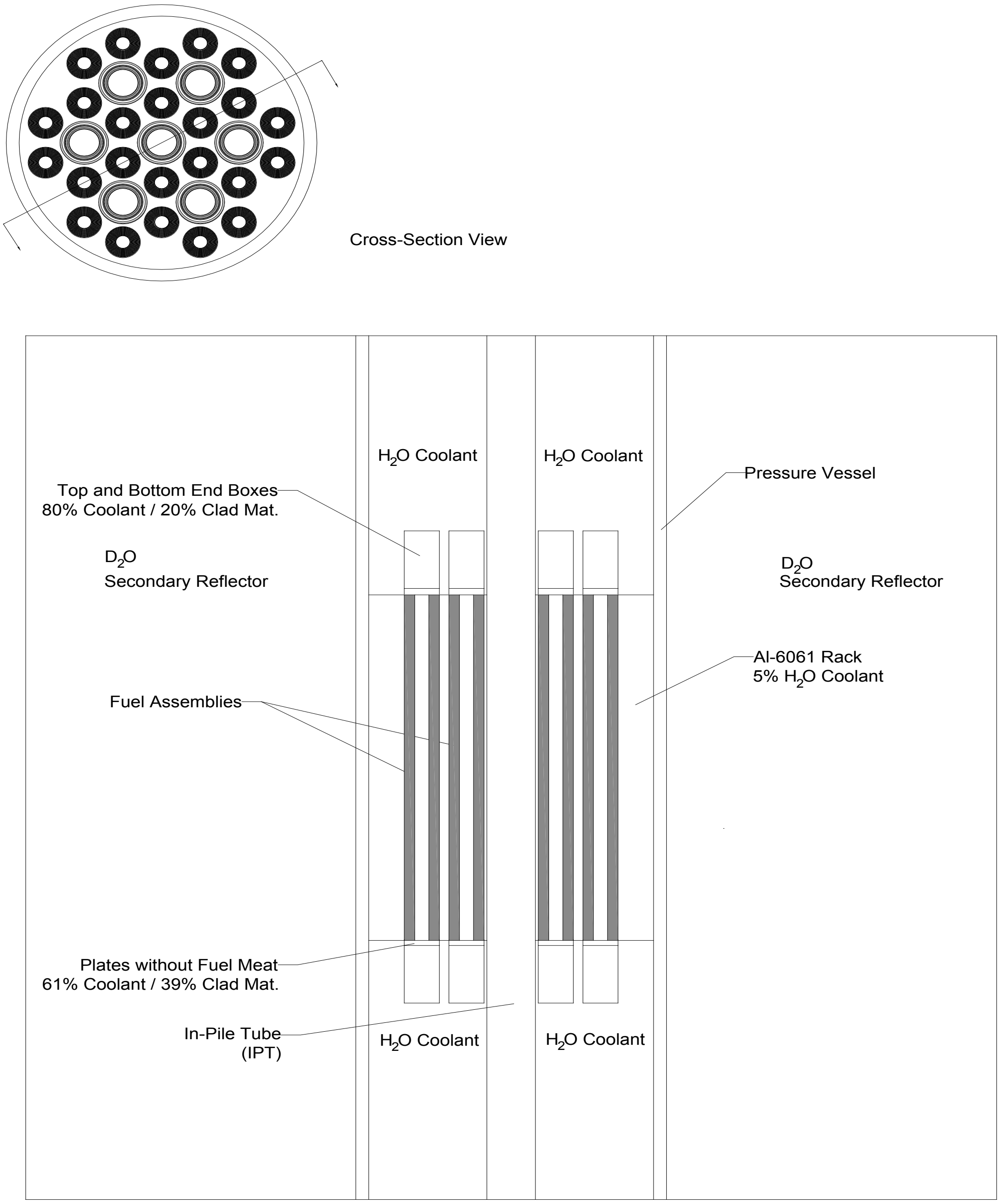

Figure 7: Vertical cross section view of cylindrical core concept.

The center flux trap has a peak fast and total flux of 3.2x10^{14} n·cm^{-2}·s^{-1} and 2.1x10^{15} n·cm^{-2}·s^{-1}, respectively and the axial flux peaking factor in the center flux trap is 1.3. The peak thermal flux in the center flux trap with aluminum fill is 2.8x10^{14} n·cm^{-2}·s^{-1}, but as mentioned before, this does not represent the actual peak thermal flux possible in the flux trap. The peripheral flux traps have a peak fast and total flux of 2.8x10^{14} n·cm^{-2}·s^{-1} and 1.8x10^{15} n·cm^{-2}·s^{-1}, respectively. The test spaces inside the inner fuel assemblies have a peak fast and total flux of 5.1x10^{14} n·cm^{-2}·s^{-1} and 2.1x10^{15} n·cm^{-2}·s^{-1}, respectively. Reflector tally position #1 (see Figure 6) has a peak thermal neutron flux of 9.8x10^{14} n·cm^{-2}·s^{-1}.

Table 2. Neutron flux in various locations of baseline Cylindrical core.

Tally Location	Neutron Flux in Reflector ($n \cdot cm^{-2}s^{-1}$) Fast Flux (>1 MeV) Thermal Flux (<0.625 eV) Total Flux		
	Peak	Axially Averaged	Peak/ Average
Center Flux Trap	3.22E+14	2.42E+14	1.33
	2.77E+14	2.12E+14	1.31
	2.07E+15	1.55E+15	1.33
Peripheral Flux Trap	2.79E+14	2.11E+14	1.32
	2.42E+14	1.87E+14	1.29
	1.80E+15	1.36E+15	1.32
Inner Fuel Assembly	5.07E+14	3.86E+14	1.31
	1.42E+14	1.12E+14	1.28
	2.14E+15	1.62E+15	1.32
Reflector Tally 1	5.52E+12	4.27E+12	1.29
	9.82E+14	7.96E+14	1.23
	1.14E+15	9.20E+14	1.24
Reflector Tally 2	—	—	—
	7.13E+14	5.96E+14	1.20
	7.22E+14	6.03E+14	1.20
Reflector Tally 3	—	—	—
	4.18E+14	3.58E+14	1.17
	4.19E+14	3.59E+14	1.17
Reflector Tally 4	—	—	—
	2.12E+14	1.85E+14	1.15
	2.12E+14	1.85E+14	1.15

Figure 8 shows a neutron flux map of the MATRIX core model at BOC. The white areas represent regions of high thermal flux, while blue areas represent low thermal flux. The yellow and red regions represent higher and lower fission power, respectively. Figure 8 illustrates how the configuration of this reactor and reflector preserve a relatively hard spectrum inside the vessel and in the IPTs while generating a very large thermal flux outside the pressure vessel in the reflector tank. The flux inside the IPTs can be thermalized up to limits dictated by water content and its associated positive loop void reactivity. This configuration also permits relatively easy access to a large volume of high thermal neutron flux in the reflector tank without having to penetrate a pressure boundary. This has significant implications with regard to options for using this neutron flux for a variety of missions.

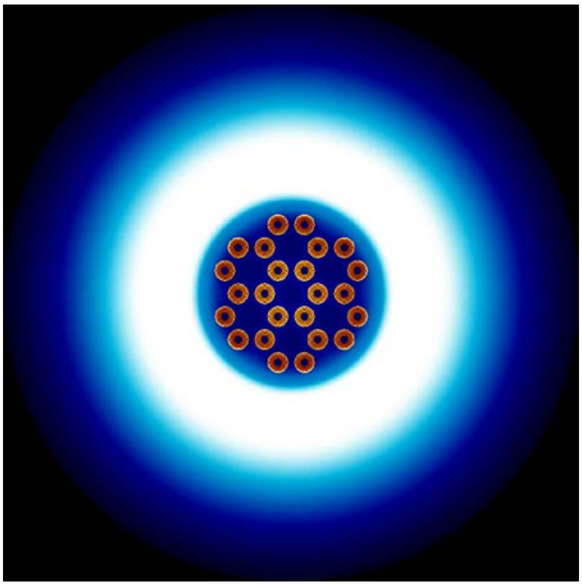

Figure 8: Neutron flux and power map of Cylindrical MATRIX core model with all fresh fuel.

Placement of the reactor vessel in close proximity to the fuel facilitates the very high neutron flux in the near-atmospheric reflector tank. A similar arrangement is planned in the Jules Horowitz Reactor (JHR) under construction at the Commissariat à l'Energie Atomique, Cadarache. (JHR reference) This very high neutron flux environment will cause damage to the vessel and replacement should be evaluated at intervals during the life of the facility.

3.2. Alternatives to D_2O Reflector Tank

The D_2O reflector tank described in the baseline case descriptions in Section 3.1 was proposed in the BATR project as a versatile and reconfigurable region of high thermal flux at atmospheric pressure. The objective of this is to allow for various types of secondary missions to be satisfied, such as beam tubes, multiple irradiation positions, etc. These would ideally be accessible during reactor operation so as not to require additional reactor shut downs on account of reflector experiment change outs. This is the motivation for placing this region outside the main pressure boundary containing the primary coolant system. With its high moderating power, extremely low capture cross section, and liquid phase at conditions of interest, D_2O is an attractive candidate for such a reflector. D_2O is expensive however and, in most applications, needs to be sealed from the environment to avoid contamination by H_2O in the air, partially compromising the accessibility afforded by its location outside the primary pressure boundary. For these reasons, other reflector options were investigated for their neutronic performance. Although the possible need for sealing D_2O from surrounding air in part drove this analysis, it should be noted that this requirement is not considered a certainty at this point as some contamination of the D_2O may be deemed acceptable.

The alternative reflector materials evaluated here are H_2O, beryllium, graphite, and a concept for removable, reconfigurable D_2O tanks in an H_2O pool. For the solid phase reflectors, some fraction of H_2O cooling is assumed. Because of its high moderating power and near-zero initial cost, H_2O is the most common reactor coolant/moderator in the world and is also considered a reflector in many reactors. Its main neutronic disadvantage is the high thermal neutron capture cross section (664 mb, mostly due to hydrogen) relative to that of D_2O (1.33 mb), beryllium (9.2 mb), and graphite (3.4 mb)[†]. This has the effect of lowering k_{eff}, but also attenuating neutron flux quite sharply in a large reflector used for irradiations such as those specified for the MATRIX concepts.

Graphite is a reflector and moderator material used in many reactors throughout history. It is evaluated here as a reflector for MATRIX, however, it is in question whether neutron-induced swelling could be accommodated. Beryllium is a reflector and moderating material used in many research reactors. This is the material used as a reflector in ATR. Despite its very attractive neutronic properties, embrittlement also limits its lifetime and the toxicity of used beryllium poses a challenge to disposal. As of the time of this writing, all beryllium reflector blocks that have been used in ATR remain in the spent fuel canal[10]. Both beryllium and graphite are analyzed as pure save for having 10% by volume H_2O coolant homogeneously mixed into them. This gives an approximation of the neutronic impact of cooling them, albeit possibly an exaggerated one since homogeneous mixing of the water can increase its impact over discrete cooling channels.

[†] 0.0253 eV values from Lamarsh & Baratta, *Introduction to Nuclear Engineering*, 3rd Edition, 2001.

Another reflector configuration to be investigated here is that of replaceable, reconfigurable D_2O tanks in an open water pool. This is proposed as a compromise of the low initial cost of H_2O and the benefits of open access to a water pool with the low capture cross section of D_2O. Figure 9 shows a three-dimensional rendering of the baseline MATRIX with replaceable, reconfigurable D_2O tanks. The tanks surround the core and pressure boundary in eight segments, each subtending a 45º angle. The tanks are constructed from Al-6061 and are here assumed to have 2 cm thick walls. These tank walls will bear minimal load since the inside and outside pressures are just above atmospheric. Test positions could be machined into the tanks such that the D_2O is sealed inside and experiments could be raised and lowered out of their irradiation positions from above the tank without shutting down the reactor and without penetrating the D_2O-filled chamber. Since these tanks would be removable, they could be sealed with space for gases (primarily tritium) to accumulate and then the tanks would be processed periodically outside the core to remove these gases.

The clearance between the D_2O tanks is assumed to be 1.0 cm and that between the pressure vessel outer surface and the D_2O tank wall is 0.5 cm. A distance of 8.0 cm is selected rather arbitrarily as the clearance between the outer D_2O tank surface and the boundary of the H_2O pool. Part of the reactor vessel is cut away to show the inside of the core and some of the tanks are removed and others cut. This shows that the D_2O tanks were conceptualized such that they extend 10 cm above and below the active core height of 120 cm. With an arrangement such as this, the removable reflector tanks described here would reduce the amount of D_2O required by roughly a factor of two, amounting to significant cost savings.

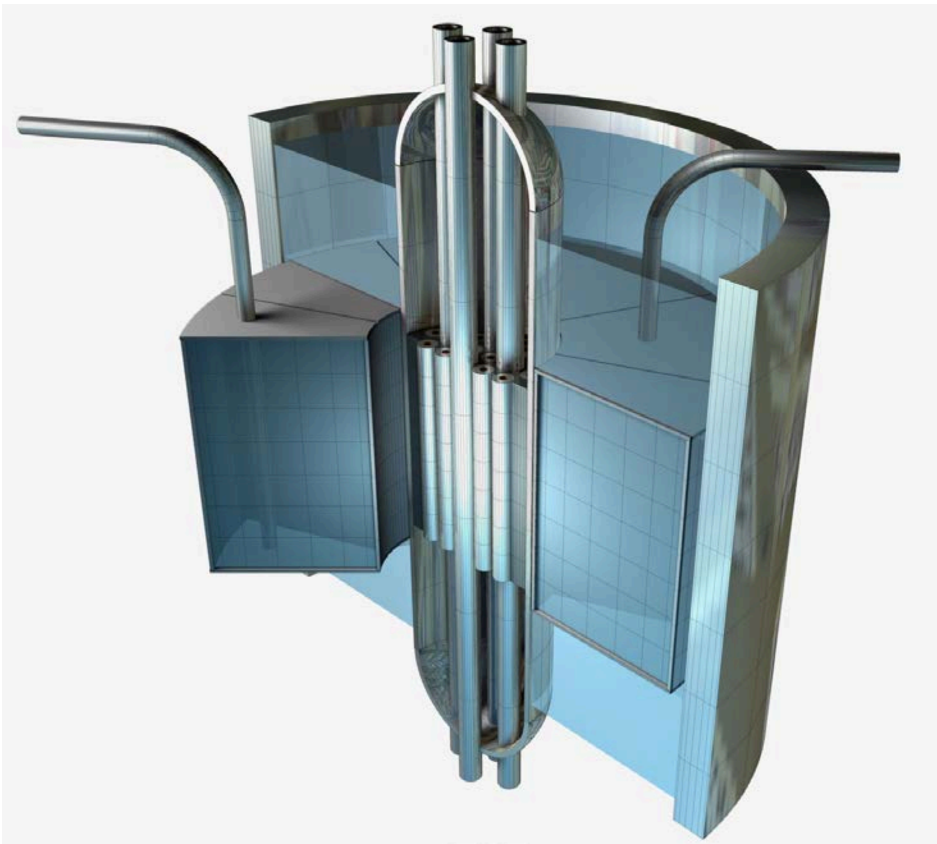

Figure 9: Three-dimensional rendering of MATRIX with D_2O tanks**.**

Serpent calculations were performed comparing seven different reflector tank contents for the otherwise-baseline MATRIX reactor. The baseline case of 100% D_2O is repeated here for comparison. The six other cases were 100% graphite, graphite with 10% H_2O, 100% beryllium, beryllium with 10% H_2O, the removable D_2O tanks described above, the same removable D_2O tanks with 10% water in the D_2O, and 100% H_2O. Figure 10 shows the peak (axially centered) thermal neutron flux versus distance from the outer surface of the pressure vessel. As expected, the reflector material that preserves the highest flux versus distance from the vessel is D_2O, with a thermal flux in excess of 10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ for essentially the entire distance to the outer reflector tank wall. This was followed by 100% graphite, and then by the D_2O tanks in a water pool. The case with 100% beryllium had the next highest flux far from the pressure vessel (although close to the vessel, it had higher thermal flux than the 100% graphite and D_2O tanks cases). The worst case by far with regard to flux in the reflector tank was the 100% H_2O case.

Table 3 shows the initial k_{eff} and B_1 values calculated for the cylindrical core with the various contents of the reflector tank region. The general trend is that the reflectors that give greatest overall thermal flux in the reflector tank also yield the highest k_{eff}. However, this trend is not absolute, as some materials are better reflectors, but may absorb more neutrons than other materials. An example is the case of 100% beryllium, which gives the highest initial k_{eff} of any of the materials examined here. However, it does not have the highest thermal flux in the reflector region (though the slope of its curve in Figure 9 indicates that it may indeed have the highest thermal flux very close to the pressure vessel. Although beryllium is the best reflector with regard to neutron reflection, it has greater absorption than some of the other candidate materials and therefore, may not be the best material for extending the thermal flux well beyond the pressure vessel. Table 3 also indicates that H_2O performs worse than any of the other candidate reflectors with regard to both initial k_{eff} and B_1.

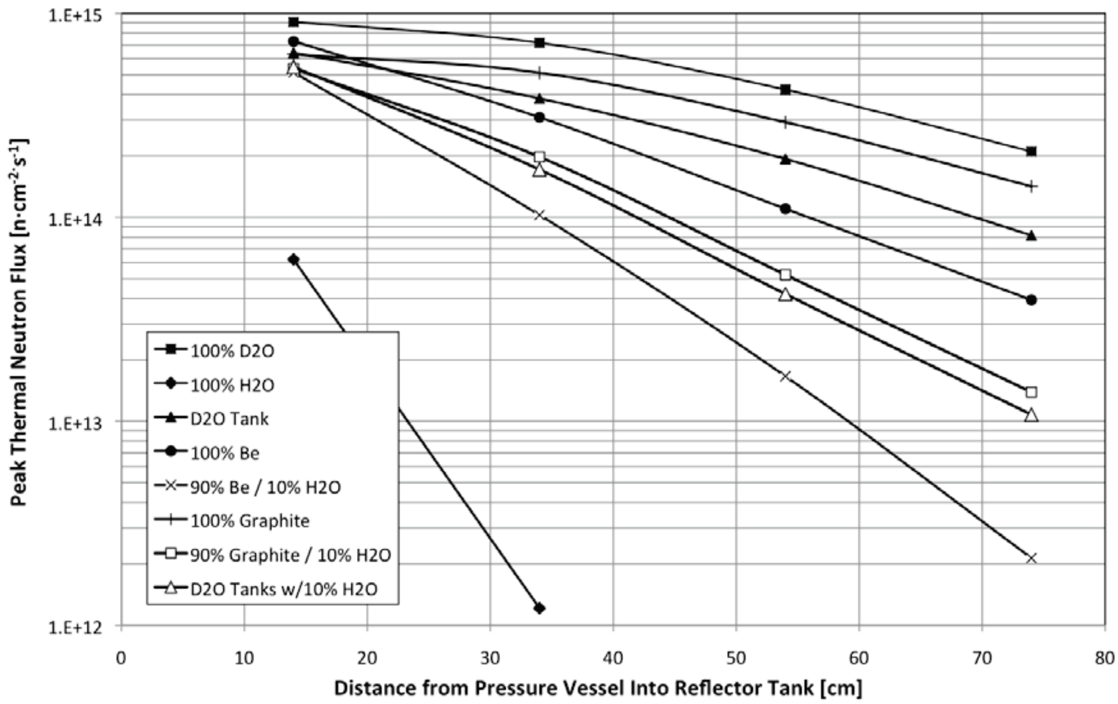

Figure 10: Peak thermal flux (at axial centerline) in reflector versus distance from pressure vessel with various reflector tank contents.

Table 3. Reactivity performance of MATRIX with various reflector contents.

Reflector Tank Contents	Initial k_{eff}	B_1 (days)
D_2O	1.184	100
H_2O	1.102	42
D_2O Tanks in H_2O	1.138	69
100% Beryllium	1.196	108
90% Beryllium / 10% H_2O	1.170	91
100% Graphite	1.191	105
90% Graphite / 10% H_2O	1.158	82
D2O Tanks with 10% H_2O in D_2O	1.126	60

3.3. Other Design Alternatives

Many other design alternatives have been evaluated in the course of this work. In order to limit the length of this paper, only three other design studies will be mentioned briefly in this section. That of alternative core rack materials, U_3Si_2 fallback fuel meat, and the stainless steel alternative pressure vessel.

3.3.1. Core Rack Alternatives (Be-Al Alloys)

The baseline MATRIX concept description specifies an aluminum core rack cooled by an assumed 6.3% by volume coolant H_2O. This configuration is meant to allow for maintaining a high fast flux, which can be thermalized if desired. A similar strategy is employed in the JHR. [11] It also avoids the use of beryllium, which is known to carry certain disposal complications. [10] Because the aluminum rack does not moderate neutrons effectively, it contributes to a rather low-reactivity core. A core with a beryllium rack and the same fuel would have a much higher reactivity. This has implications toward achievable cycle length and the feasibility of using alternative fuel forms with lower heavy metal (HM) density than U-10Mo. This will be discussed further in Section 3.3.2. As an alternative to the aluminum rack, it has been proposed that aluminum/beryllium alloys be considered for this application. Adding beryllium content affects the neutronic characteristics of the core by 1) increasing reactivity, 2) reducing fast flux in test positions, and 3) reducing fast neutron fluence on the pressure vessel.

Serpent calculations were performed on the baseline MATRIX core to quantify the effects of different core rack contents. In this analysis, beryllium content in the core rack was varied from 0 to 93.7 % with H_2O assumed to always occupy 6.3% by volume and the balance being aluminum. Though intermediate alloys were evaluated, only the bounding cases of all-aluminum and all-beryllium are presented here. Results of calculations carried out on the MATRIX core are given in Table 4. The calculations were repeated with both the baseline all-aluminum rack and the all-beryllium rack, both having 6.3% by volume cooling water homogeneously mixed throughout. The table shows k_{eff} and B_1 along with neutron flux in three different locations; center flux trap, peripheral flux trap, inside an inner-ring fuel element, and reflector tally position #1 (location from Figure 6). From Table 4, one sees that the fast flux in the center flux trap as a result of increased beryllium content in the rack decreases from 3.2×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ with all-aluminum rack to 2.7×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ with the all-beryllium rack. With the same change in rack content, thermal flux in the center flux trap increases from 2.9×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ to 3.6×10^{14} $n \cdot cm^{-2} \cdot s^{-1}$ and total flux is virtually unchanged. The same general trend is seen in the peripheral flux trap. The trend in the location at the center of an inner fuel assembly is a bit less pronounced, with fast flux seen ranging from $5.2\text{x}10^{14}$ $n{\cdot}cm^{-2}{\cdot}s^{-1}$ (all-aluminum rack) to $4.7\text{x}10^{14}$ $n{\cdot}cm^{-2}{\cdot}s^{-1}$ (all-beryllium rack). Again, total flux is virtually unchanged in this location with rack content changes. Less change in the flux inside the fuel assemblies would be expected since each of these locations is shielded from the rack by the fuel plates of its surrounding element.

Changing from all-aluminum to an all-beryllium rack was also found to decrease the thermal flux in the reflector slightly, likely a result of more absorption in the rack due to better thermalization, followed by capture in the cooling water. Meanwhile, the initial k_{eff} increased from 1.18 to 1.28, offering quite a bit of flexibility in design such as the option of reducing the fuel loading, having longer cycle lengths, etc. The B_1 values increased from 100 days in the baseline aluminum rack case to 175 days in the all-beryllium rack case. Also of note is the somewhat lower axial peaking in fast flux in the flux traps for the aluminum rack than the beryllium rack. The aluminum rack case has lower axial peaking because it has moderation concentrated at the top and bottom from cooling water with less moderation along the length of the fuel. The beryllium rack provides moderation along its length, so a more peaked shape results.

Table 4. Performance parameters for MATRIX core with varying rack contents

Core Rack Content and Tally Location		Neutron Flux ($n{\cdot}cm^{-2}{\cdot}s^{-1}$) Fast Flux (>1 MeV) Thermal Flux (<0.625 eV) Total Flux		
Core Rack ID k_{eff} and B_1	Tally Location	Peak	Axially Averaged	Peak/ Average
93.7% Aluminum 0% Beryllium 6.3% H_2O (Baseline design) Initial k_{eff}=1.18313 B_1 = 100 days	Center Flux Trap	3.21E+14	2.42E+14	1.33
		2.87E+14	2.16E+14	1.32
		2.07E+15	1.55E+15	1.33
	Peripheral Flux Trap	2.78E+14	2.09E+14	1.33
		2.50E+14	1.90E+14	1.32
		1.79E+15	1.35E+15	1.32
	Inner Fuel Assembly	5.20E+14	3.91E+14	1.33
		1.42E+14	1.12E+14	1.27
		2.14E+15	1.62E+15	1.32
	Reflector Tally Position 1	5.30E+12	4.48E+12	1.18
		1.00E+15	8.12E+14	1.24
		1.17E+15	9.35E+14	1.25
0% Aluminum 93.7% Beryllium 6.3% H_2O Initial k_{eff}=1.28381 B_1 = 165 days	Center Flux Trap	2.68E+14	1.94E+14	1.38
		3.61E+14	2.65E+14	1.36
		2.00E+15	1.44E+15	1.39
	Peripheral Flux Trap	2.41E+14	1.75E+14	1.38
		3.24E+14	2.39E+14	1.35
		1.77E+15	1.29E+15	1.37
	Inner Fuel Assembly	4.74E+14	3.46E+14	1.37
		1.57E+14	1.18E+14	1.33
		2.05E+15	1.51E+15	1.36
	Reflector Tally Position 1	4.24E+12	3.12E+12	1.36
		9.41E+14	7.56E+14	1.24
		1.06E+15	8.46E+14	1.25

Calculations also showed that replacing the all-aluminum rack with the all-beryllium rack reduces the fast fluence on the inner surface of the pressure vessel by approximately a factor of two while only slightly decreasing the total fluence. As most of the neutron damage is caused by fast neutrons, this will likely extend the service life of the pressure vessel.

3.3.2. Alternative Fuel Meat (U_3Si_2)

Monolithic U-10Mo fuel has not been fully qualified for use in a high heat flux test reactor at the time of this writing. Therefore, the MATRIX concept and its variants were evaluated for their ability to use U_3Si_2 fuel, a fuel with lower uranium density than U-10Mo, but lower anticipated technical development risk as well. U_3Si_2 fuel can be fabricated with various volume fractions of aluminum. While U_3Si_2 fuel has been qualified with a uranium density of 4.8 g/cm^3, it is believed that a uranium density of 6 g/cm^3 (53% by volume U_3Si_2) is feasible with relatively low development risk so this was used as an alternative. This change in fuel was found to carry a significant penalty in k_{eff} and B_1 and some of the lower-reactivity variants that use an aluminum rack were eliminated. Note that other fuel forms are being investigated (e.g. U-7Mo dispersion) with higher HM density than U_3Si_2. They were not evaluated here, but should be considered in future work as alternatives to the baseline U-10Mo monolithic fuel.

3.3.3. Alternative Pressure Vessel Material (SS304)

In a design such as MATRIX featuring a reactor pressure vessel (RPV) located close to the fueled portion of the core, vessel materials with low neutron capture cross-section are favorable for two main reasons. One reason is that the vessel is close enough to fuel that neutron capture affects core reactivity substantially, particularly in the cases where a low-moderation core rack is used. The other motivation for a low-absorption RPV is the desire for high flux outside the RPV. Aluminum (specifically Al-6061) was identified as a candidate material to serve as a low-absorption pressure boundary in the BATR project. [3] It is also the material to be used in the RPV for the Jules Horowitz Reactor at CEA, Cadarache. [11] An alternative material is the more conventional stainless steel 304 (SS304). Zircaloy was also evaluated as an RPV with the same thickness as the SS304 (though it could likely be made slightly thinner than SS304). SS304 was found to carry a significant reactivity penalty due to its close proximity to the core and significant parasitic absorption. However, it's higher strength and established use as a pressure vessel give it some advantages over aluminum. It was also found that Zircaloy performs best of all three neutronically, though the cost of a Zircaloy vessel of this size may be prohibitive. So while aluminum is the lead candidate RPV material due to its higher neutron transparency, SS304 should not be ruled out.

4. COMPARISONS TO SIMPLIFIED ATR

In general, a goal for MATRIX would be that it would perform as well or better than ATR by most metrics. To evaluate whether the proposed MATRIX designs can meet specified requirements, comparisons to an analogous ATR model were performed. A Serpent 2 model of ATR was acquired that had been built based on the critical tests associated with the 1994 Core Internals Changeout (94-CIC configuration). A core loaded in conjunction with the 94-CIC was comprised entirely of fresh HEU Mark VII fuel (standard ATR fuel at time of this writing). This core was never operated at full power, but used as a low-power critical configuration to validate codes and input. This core configuration was later described in detail and archived in the International Handbook of Evaluated Reactor Physics Benchmark Experiments. [9]

In the MATRIX studies presented thus far, the reactor models are quite simplified. For example; omission of burnable poisons and control rods, fully fresh cores depleted in a single pass, and uniform temperatures. While these assumptions are necessary to accommodate the large option space in early studies, they do make comparisons to real data from the operating ATR less meaningful.

In order to provide a meaningful comparison between the ATR and the simplified and immature MATRIX concepts, some simplifying changes were made to the ATR model. The most significant among the simplifications was the removal of control material. In the ATR, excess reactivity is held down by 1) neck shim rods located in the neck shim rod housing 2) rotating Outer Shim Control Cylinders (OSCCs) having a surface partially covered with hafnium absorber and 3) boron carbide burnable poison mixed into the fuel meat of the inner and outer four plates of each fuel assembly. All three of these types of control material were omitted from the ATR model in order to be consistent with the state of development of the MATRIX models.

The 94-CIC configuration specifies that the test region of the northwest IPT be filled with Al-6061. The west, southwest, and southeast IPTs were filled with coolant H_2O. Deviations made from the exact 94-CIC specifications for the purposes of this study are described below. For a full description of the 94-CIC test configuration, see Reference [9].

The Serpent model was modified in the following ways to be more analogous to the MATRIX calculations performed here:

- In the ATR's Mark VII HEU fuel, the inner and outer four fuel plates have boron carbide mixed into the fuel meat. The boron carbide was removed from the fuel meat of these plates leaving the uranium and aluminum nuclides as they were. This gave a model with the same amount of HM as would an actual all-fresh ATR core but with no burnable poison.
- All neck shim rods were modeled as withdrawn.
- The hafnium absorber material on the surface of the OSCCs was replaced with beryllium.
- The southwest flux trap was filled with Al-6061, leaving west and southeast flux traps filled with H_2O as specified for the benchmark.
- Reactor power was set to 250 MW_{th} rather than the zero power of the 94-CIC critical tests or the ~120 MW_{th} at which ATR is typically operated.

This provided a "flat" core (i.e. not purposefully tilted as is common in operation of ATR) with all fresh unpoisoned HEU fuel. Depletion was performed on this model without changing any of the neck shim positions and neutron flux tallies were calculated at the burnup step closest to k_{eff}=1.0. This methodology provides an analogous ATR case to which one may compare the MATRIX cases. This model was depleted with flux tallies in four different positions; the northwest flux trap, the southwest flux trap, the southeast flux trap, center capsule location #3, and the A-1 hole. For locations and dimensions of these locations, see Reference [9].

The initial k_{eff} for this core was calculated to be 1.222 and the B_1 value was 56 days. Table 9 shows flux values in the locations described above. As with the MATRIX studies, the tallies spanned the entire axial length of the active core and were divided into 10 axial segments. Peak/average ratios are given for each tally.

As listed in Table 9, the aluminum-filled northwest and southwest flux traps have fast flux values of approximately 2.9x10^{14} and 3.5x10^{14} n·cm^{-2}·s^{-1}, respectively, and total fluxes of 1.9x10^{15} and 2.3x10^{15} n·cm^{-2}·s^{-1}, respectively. Upon comparison of these values to those shown in Table 2 for the baseline MATRIX case, one sees that the fast flux is comparable, if not slightly higher in the ATR case. This could be due to the novel way that ATR fuel wraps around flux traps allowing fast neutrons relatively unimpeded access to test volumes. Total fluxes in flux traps are comparable between ATR and MATRIX. Overall, one would expect test locations close to the fuel to be similar in flux values between the reactors because the power density in fuel elements has been designed to be similar between the concepts. Although the IPTs have similar flux between ATR and MATRIX, the proposed size of the flux traps in MATRIX are larger, creating an advantage with regard to the experiments that can be inserted.

Table 9. Flux values from simplified ATR model depletion calculation.

Tally Location and Location Content	Neutron Flux (n·cm^{-2}·s^{-1}) Fast Flux (>1 MeV) Thermal Flux (<0.625 eV) Total Flux		
	Peak	Axially Averaged	Peak/ Average
NW Flux Trap Al-6061 filled	2.89E+14	2.08E+14	1.39
	4.47E+14	3.20E+14	1.40
	1.87E+15	1.33E+15	1.40
SW Flux Trap Al-6061 filled	3.48E+14	2.48E+14	1.40
	5.30E+14	3.68E+14	1.44
	2.28E+15	1.60E+15	1.42
SE Flux Trap H_2O filled	2.29E+14	1.60E+14	1.43
	1.22E+15	8.65E+14	1.41
	2.38E+15	1.68E+15	1.42
Center Capsule Location #3 Al-6061 filled	2.99E+14	2.10E+14	1.42
	1.26E+15	9.24E+14	1.37
	2.67E+15	1.95E+15	1.37
A-1 Hole Al-6061 flow restrictor	4.44E+14	3.20E+14	1.38
	7.24E+14	5.23E+14	1.38
	2.53E+15	1.81E+15	1.40

In the water-filled southeast flux trap, the peak thermal and total flux are 1.2x10^{15} and 2.4x10^{15} n·cm^{-2}·s^{-1}, respectively. Note that a flux trap filled completely with water would likely not be operated due to limits on the positive void reactivity of the enclosed test loop. The highest fast flux of the locations sampled here was located in the A-1 hole filled with an aluminum flow restrictor. This location had a fast neutron flux of 4.4x10^{14} n·cm^{-2}·s^{-1} due to its very close proximity to fuel. This is slightly lower in magnitude than the fast flux in the inner assembly location of MATRIX, which has a fast flux of 5.2x10^{14} n·cm^{-2}·s^{-1}. Furthermore, there is a location such as this at the center of every fuel assembly. Even if some of the fuel assembly centers are occupied by shim and/or shutdown rods, it is believed that several of these can be made available for irradiations in hard spectra. This could yield larger total volume of hard-spectrum irradiation space than is available in ATR. This objective should be continually assessed as the design is developed in future work.

5. CONCLUSIONS AND FUTURE WORK

This report provides a description and some analysis of a potential design candidate MATRIX reactor for deployment in the event the ATR were to shut down permanently. This concept is expected to be capable of higher burnup than the ATR (or longer cycle length given a particular refueling scheme and power level). The volume of test space in the IPTs is larger in MATRIX with comparable magnitude of neutron flux as ATR. In addition to the IPTs, the MATRIX concept features test spaces at the centers of fuel assemblies where very high fast flux can be achieved. This magnitude of fast flux is similar to that achieved in the ATR A-positions, however, the available volume having these conditions is greater in the MATRIX design.

Perhaps the most significant advantage of the MATRIX design over the ATR is the large reflector tank located outside the pressure vessel. Constructed from aluminum and located close to the fuel elements, the low parasitic capture pressure vessel facilitates a very high thermal neutron flux in the reflector tank. Nominally, the reflector tank would be an atmospheric pressure (or near-atmospheric) D_2O volume to serve as a high thermal flux test space. In this work, an alternative concept was introduced wherein aluminum cans containing D_2O are placed in an open H_2O pool. From preliminary neutronic studies, this concept appears to perform well.

From the analyses performed in this work, it appears that the MATRIX concept can be designed to meet the anticipated needs of current and potential ATR customers for the foreseeable future. This statement, however, must be qualified by acknowledging that this design is quite immature, and therefore the performance must be re-evaluated as the design is developed. As is always the case after a conceptual design study such as this, there is a multitude of follow-on work that needs to be performed in order to verify the feasibility and understand the performance of the MATRIX reactor. The more important topics of these include, but are not limited to:

Assembly Pitch and size – The overall proximity of fuel elements to one-another is limited by their size and by their adherence to a strict hexagonal array. Higher-reactivity configurations can be achieved by changing the element size along with clustering them closer together in non-hexagonal arrays. This is seen in the core map of the JHR. [11]

Power Tilt and Shim – The ATR is able to tilt its reactor power to a 3/1 ratio based on lobe power from southwest to northeast lobes. Though the concept of lobes may not have explicit meaning in the MATRIX reactor, the ability of achieving a 3/1 ratio of neutron flux in two separate IPTs will likely be required of MATRIX.

Shutdown – The location, geometry, and materials of the shutdown mechanisms should be evaluated in detail in future work.

Secondary Missions – Detailed analysis of performance in specific secondary missions has not been performed. For example, rate of production of various radioisotopes, and the performances of a beam tube or a neutron radiograph system should be analyzed. Figures of merit should be not only the performance of each secondary mission individually, but their effects on the core, if any, along with their interference with one another should be considered as well.

Aluminum Pressure Vessel – The neutronic benefits of an aluminum pressure vessel in the MATRIX are substantial over stainless steel. However, the cost and licensing burdens of using an aluminum pressure vessel over a more conventional material should be analyzed. Frequency of replacement compared to other materials should also be evaluated.

Fuel Element Hold-Down in Upflow – Hydraulic forces on a fuel element at the flow speeds typical in a high-power reactor such as MATRIX can be greater than the weight of the fuel element. The mechanisms for holding fuel elements in place in upflow should be considered at an early stage. Other items, such as tests and flow restrictors, also must be held in place as their ejection can introduce a positive reactivity effect. This must also be investigated at an early state of analysis.

Transient Testing – The MATRIX reactor must be capable of producing transient power excursions analogous to those of PALM cycles in the current ATR. A scheme to achieve this should be devised.

Thermal Hydraulic Analysis – In the early stages of this work, the decision was made to adopt the power density and plate thickness of the ATR for the MATRIX concepts. This was done in order to simplify the design space initially, providing some assurance that the steady state thermal hydraulic conditions would be satisfactory. A detailed analysis should be performed in order to verify that the temperatures in fuel and cladding are acceptable. The potential for passive decay heat removal should be evaluated as well.

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