



T R A N S A T O M I C

N E U T R O N I C S O V E R V I E W

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Transatomic Power Corporation

One Broadway, 14th Floor

Cambridge, MA 02142

(617) 520-4850

info@transatomicpower.com

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ABSTRACT

For nuclear energy to be a viable source of baseload power, new nuclear reactor designs must address existing concerns about waste storage by increasing fuel utilization and reducing overall waste production. This paper outlines the ways in which the Transatomic Power 1250 MWth molten salt reactor design takes advantage of its liquid fuel in order to address these challenges. By employing continuous fission product removal and core geometry modification, the TAP MSR achieves more than twice the fuel utilization of light water reactors (LWRs). When using 5% enriched uranium – the maximum enrichment readily available in the current supply chain – this increased efficiency leads to an approximate 53% waste reduction compared to LWRs. Using higher enrichments, up to the 20% Low Enriched Uranium (LEU) limit, this reduction reaches 83%.

INTRODUCTION

In many respects, nuclear energy is an ideal alternative to fossil fuels – it is a well-characterized technology that can generate a large baseload supply of carbon-free electricity at minimal operating cost. However, the nuclear power industry has historically been hampered in part by concerns about legacy technology’s production of nuclear waste. To address this concern, multiple advanced reactor technologies feature innovative designs that can achieve much higher fuel efficiencies, reducing the overall waste production. One of these technologies, the molten salt reactor (MSR), was first developed at the Oak Ridge National Laboratory in the 1950s and 1960s, and is a reactor design that utilizes liquid fuel.

Transatomic Power (TAP) has modified the Oak Ridge design, utilizing a new moderator and fuel salt to attain criticality using low-enriched uranium fuel in a much more compact core [1], and this report provides additional detail about the reactor physics and operational characteristics of such a design. While this design does not reduce existing stockpiles of spent nuclear fuel (SNF), TAP’s goal in developing this reactor is to reduce nuclear waste production by significantly increasing fuel burnup.¹ To achieve this goal, the design incorporates a new method of moderator rod operation that minimizes excess reactivity² in the core, allowing for considerably higher burnup than light water reactors and decreased waste production.

FUEL UTILIZATION IN LIQUID-FUELED REACTORS

Nuclear fuel burnup is a key parameter for measuring the efficiency of the nuclear fuel cycle. A higher-burnup fuel cycle decreases waste production, and thus potentially reduces long-term waste storage costs and environmental impact. Increasing burnup is equivalent to increasing the amount of time that a reactor maintains criticality with a given amount of fuel, which is typically accomplished by having either a large initial loading of fissile material, or a high conversion of fertile to fissile nuclei in the reactor³ [2]. The main factors limiting fuel burnup in light water reactors are structural damage and trapping of fission products in the solid fuel pellets. Over the course of operation, fission product buildup within the rods can reduce the fuel pellets’ thermal conductivity, thereby increasing the rod’s centerline temperature and decreasing the margin to the fuel’s melting point. Increased fuel temperature and radiation-induced damage can lead to swelling in the fuel pellets, which in turn can cause the fuel pellets to impinge

¹ Burnup, also known as fuel utilization, is defined as the thermal energy extracted per unit mass of heavy metal, typically given in units of GWd/MTHM, where MTHM stands for Metric Tons of Heavy Metal [4].

² Reactivity, ρ , is defined by the equation, $\rho = \frac{k_{eff}-1}{k_{eff}}$, in which k_{eff} is the reactor’s effective multiplication factor, the ratio of neutron populations from one neutron life cycle to the next.

³ This conversion is quantified by a parameter known as *conversion ratio*, which is defined as the ratio of the fissile material production rate to the fissile material consumption rate.



The typical definition of fuel burnup is given by the following equation:

$$Bu = \frac{P \cdot t}{m_I}$$

in which Bu is the burnup, P is the reactor's thermal power, t is the fuel residence time, and m_I is the initial actinide loading in the reactor.

In liquid fueled reactors, it is necessary to modify this equation to take into account the additional mass of actinides, m_a , added to the reactor over the course of operation:

$$Bu = \frac{P \cdot t}{m_I + m_a}$$

on the cladding and, in extreme cases, lead to cladding failure. Buildup of fission products within the fuel rods also reduces the system's reactivity over time, because of several isotopes' high neutron absorption cross sections.

FISSION PRODUCT REMOVAL AND REACTOR FUEL UTILIZATION

In contrast to solid fuel, liquid fuel has numerous advantages that allow for increased fuel burnup. Liquid fuel has no long-range structure to be damaged, does not experience significant volumetric swelling, and avoids fission product poisoning through continuous fission product removal. To measure quantitatively the effect of fission product removal in molten salt systems, two depletion calculations⁴ on otherwise identical models were run using the neutronics software Serpent 2.26 [3]. The results of the simulation can be seen in Figure 1, with additional parameters given in Appendix A.

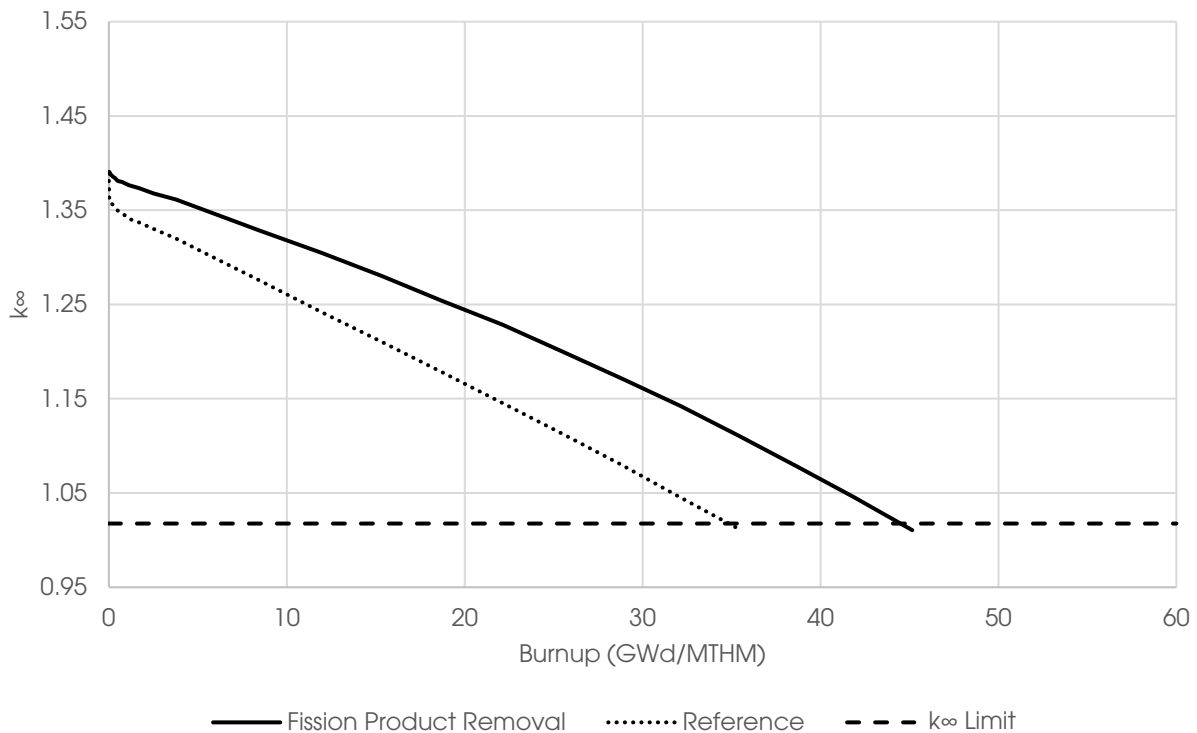


Figure 1. The infinite multiplication factor, k_{∞} , as a function of burnup, illustrating the effects of fission product removal on two otherwise identical systems. The dashed line for a k_{∞} of 1.018 indicates the point at which the reactor can no longer sustain the fission chain reaction when taking into account radial and axial leakage.

⁴ Depletion calculations are computer simulations that model burnup over the course of the reactor's life.



Although fission product removal increases the maximum burnup by more than 27% (44.5 GWd/MTHM compared to 34.8 GWd/MTHM), the final value still falls short of yielding significant improvements when compared to current LWR limits (45 GWd/MTHM). To increase this value further, it is possible to take advantage of another aspect of liquid fueled reactors: the ability to easily vary their geometry, by inserting or removing moderator rods in the reactor core.

A NEW TAKE ON REACTIVITY CONTROL

In order to account for the loss of fissile material over the course of operation, solid-fueled nuclear reactors use core configurations and fuel compositions that result in excess positive reactivity at the beginning of life. During normal operation, excess reactivity is accounted for using control rods and soluble absorbers – the additional neutrons are captured, and the reactor can be operated safely.

The concept of “reactivity swing” (the variation in reactivity in a reactor over the course of a given fuel loading) is shown graphically in Figure 2. A reactor with a large reactivity swing is inherently inefficient, as neutrons that could have otherwise been used for fission and conversion in the fuel are effectively wasted in the absorbers and control rods. An ideal reactor therefore has minimal reactivity swing, and can use its neutron population with the greatest efficiency during operation.

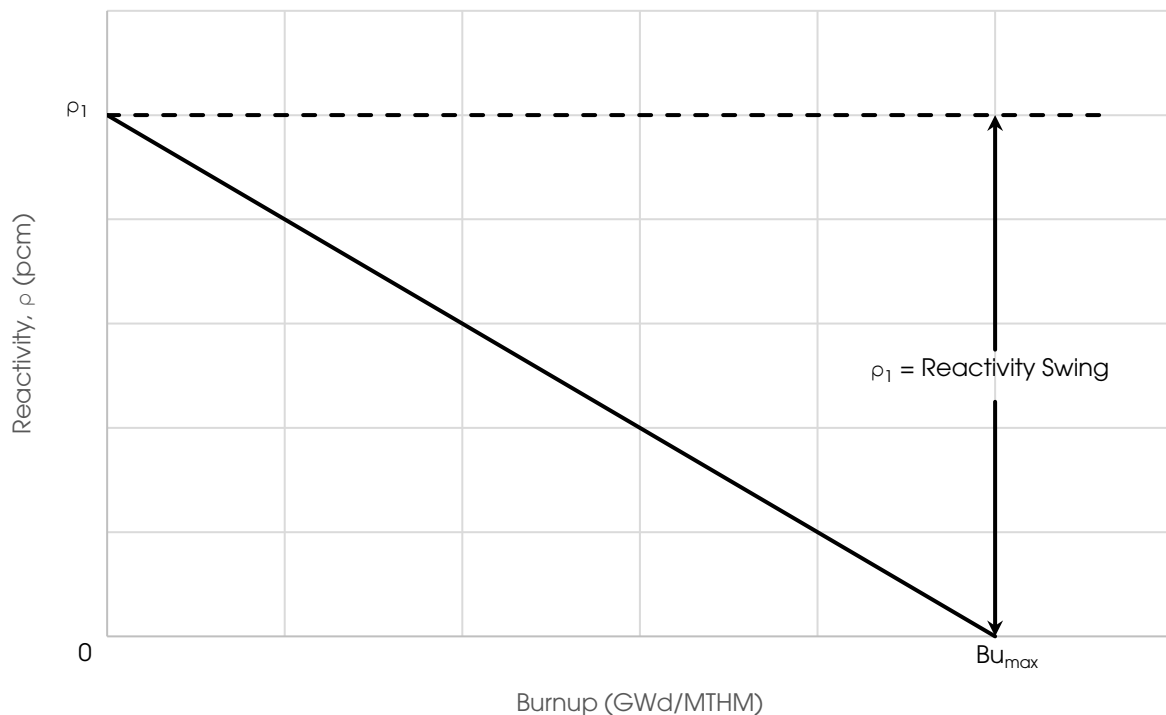


Figure 2. A graphical depiction of reactivity swing, adapted from Cochran and Tsoulfanidis, 1990 [2]. As fissile material is consumed throughout the course of operation, the extent to which positive reactivity exists in the core diminishes, up until the point at which criticality ($\rho = 0$) is no longer achievable (maximum burnup).

The neutron balance in a reactor is a function of both geometry and core material composition. In solid-fueled reactors, geometry is largely fixed, and control is achieved through modifications in the composition – insertion and removal of neutron absorbing material. Liquid-fueled reactors, however, make it possible to control reactivity by



altering the core's geometry, thereby largely eliminating reactivity swing in the system. Specifically, it is possible to readily vary the salt volume fraction (SVF; the percentage of the core volume occupied by fuel salt) in an MSR core by inserting or removing moderator rods. In order to illustrate the effect of changing SVF in the core, a scoping study consisting of multiple steady state pin cell calculations was performed in Serpent 2.26.

Varying the levels of ^{235}U enrichment (1 – 5%) and modifying SVF through the simulation of different-sized moderator rods makes it possible to estimate the evolution of the fuel with time. Figure 3 shows a plot of infinite multiplication factor as a function of SVF for these multiple enrichment levels. The graph shows that as fissile material concentration in the reactor decreases, the reactor can either maintain criticality:

- I. By beginning with a large amount of excess reactivity and a constant SVF, as in a stationary moderator rod configuration, or,
- II. By decreasing SVF, as in a moveable moderator rod configuration.

Although exact operational progression cannot be derived from this graph, since the simulations do not take into account the generation of transuranic isotopes over the course of life, the general relationships among fissile concentration, multiplication factor, and SVF can be inferred.

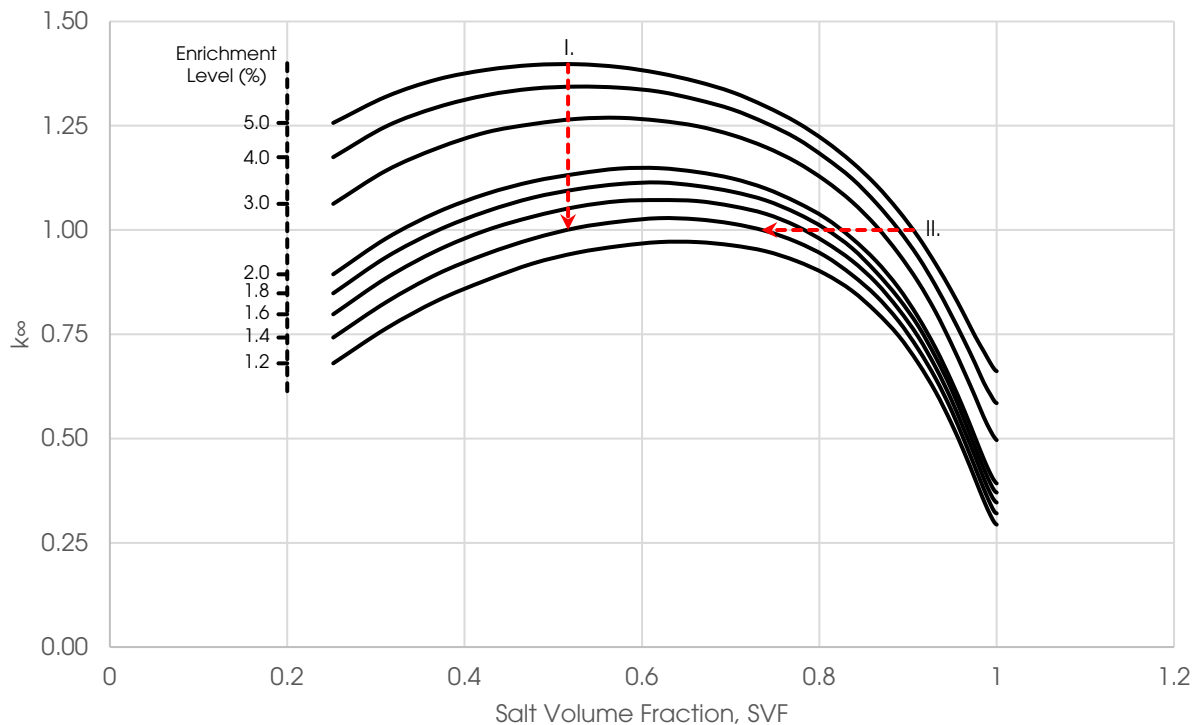


Figure 3. The infinite multiplication factor as a function of SVF and ^{235}U enrichment. The expected operational progression of both a stationary rod configuration (I., vertical) and moveable rod configuration (II., horizontal) have been presented through the use of red dashed lines. Additional simulation details are given in Appendix A.

From this trend, it can be seen that it is possible to control the system's reactivity by adjusting the reactor's SVF. Such an adjustment can thus be used to eliminate reactivity swing, thereby increasing the reactor's neutron utilization efficiency and conversion ratio. Figure 4 illustrates this effect through a plot of conversion ratio as function of enrichment and SVF. The graph demonstrates that as fissile material concentration decreases, moving from higher to



lower SVF does, in fact, produce a higher conversion ratio over the course of life (II) than does a stationary rod configuration (I).

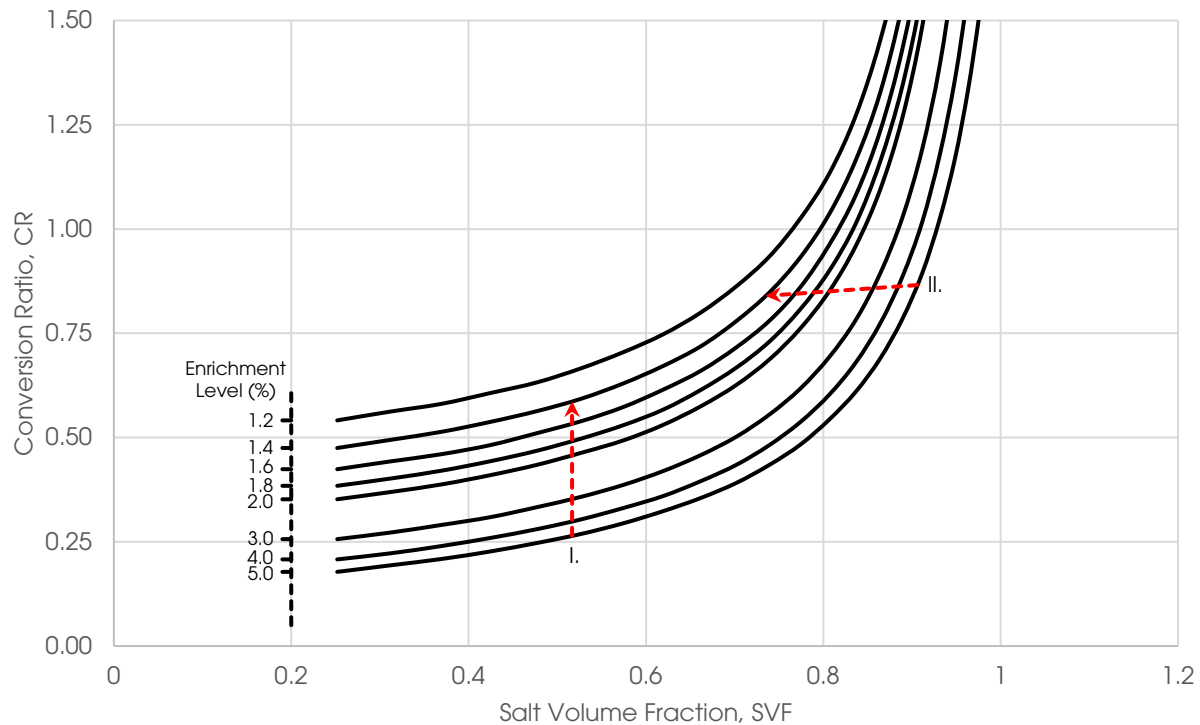


Figure 4. Conversion ratio as a function of SVF and ^{235}U enrichment. SVF was modified by changing the moderator rod radius. The expected operational progression of both a stationary rod configuration (I., vertical) and moveable rod configuration (II., horizontal) have been presented through the use of red dashed lines.

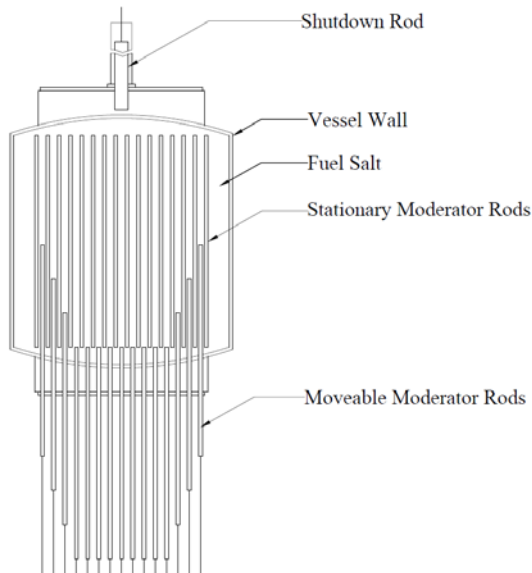


Figure 5. A conceptual depiction of a reactor core design that uses moveable moderator rods for reactivity control. The gap between the moderated region and the reactor vessel wall contains unmoderated fuel salt that acts as both a reflector and fertile nuclei conversion zone.

In practice, SVF will be varied by inserting fixed-sized moderator rods via the bottom of the reactor vessel (for safety considerations), in a manner analogous to moving the control rods in a boiling water reactor, as shown in Figure 5. The reactor therefore does not require control rods beyond a single shutdown rod for safety. As an additional shutdown mechanism, moderator rod removal may also act as a means of bringing the reactor into a subcritical state.

DEPLETION CALCULATIONS WITH MOVABLE MODERATOR RODS

The scoping calculations make it possible to infer the influence of SVF on burnup and reactivity control in an MSR. Following this initial scoping, assembly level depletion calculations were used to simulate the system evolution over time to a higher degree of accuracy. In these simulations, SVF was adjusted to minimize excess reactivity. The model

employs continual fission product removal and a fuel addition method that maintains absolute actinide concentration following each time step.⁵ The initial actinide loading is comprised of 5% LEU, with stationary rod results being presented for comparison.

Figure 6 shows the change in SVF in the TAP reactor as a function of burnup. At the beginning of life, a high SVF is used to harden the spectrum⁶ and convert the maximum amount of fertile material while the fissile content is still high. As shown in Figure 7, as fissile concentration begins to drop additional moderator rods are inserted in order to maintain criticality, leading to a decrease in SVF and subsequently conversion ratio (Figure 8).

Although the final mass of feed in these simulations is small, the fissile concentration of the feed material has a direct influence on conversion ratio, as it can be thought of as an additional fissile production term. Equation 1 illustrates this effect by presenting an effective conversion ratio (CR_{eff}), defined as the system's conversion ratio (CR) plus the fissile concentration of the feed (F_{Feed}).

$$CR_{eff} = CR + F_{Feed} \quad (1)$$

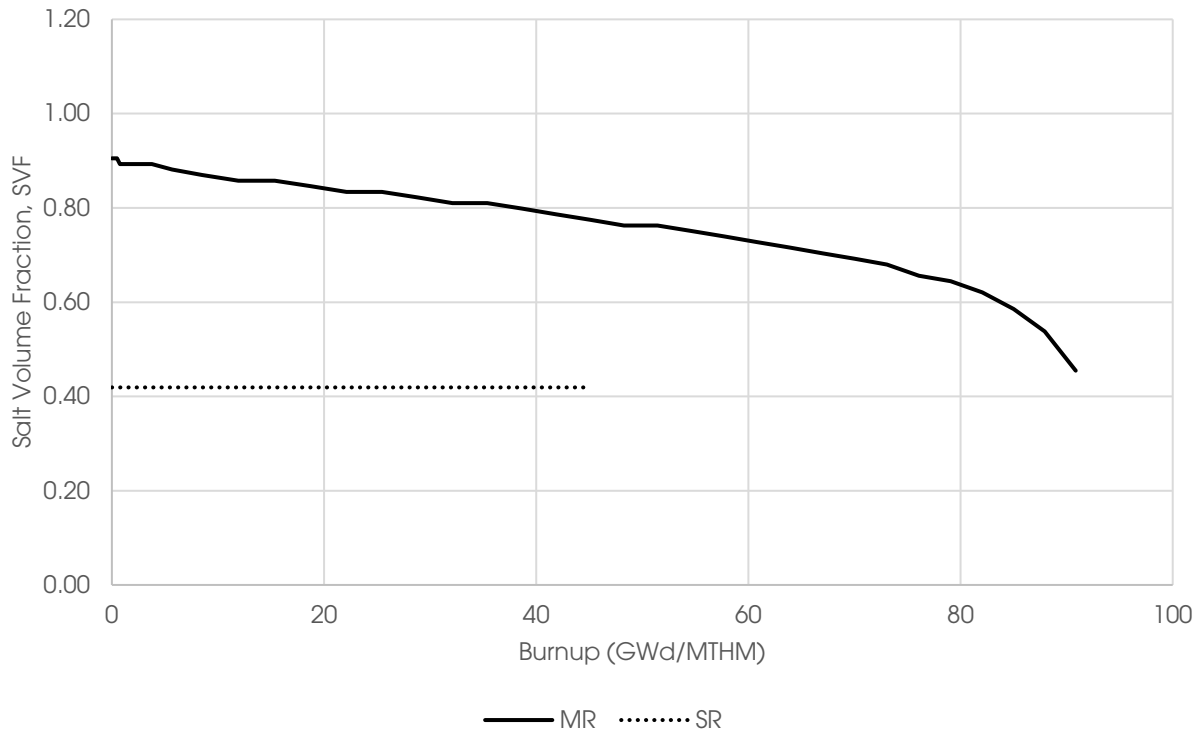


Figure 6. The change in SVF as a function of burnup in the TAP reactor for a 5% enriched initial fuel load. The legend refers to the two possible operational mechanisms: Stationary Rods (SR) and Moveable Rods (MR). Additional calculation details are given in Appendix A.

⁵ In an operating MSR, it is necessary to add additional actinide fuel to the system so that the fuel salt can remain at its eutectic composition. Additionally, further details on fission product removal are included in Appendix A.

⁶ Also note that at the beginning of life, leakage in the system is approximately 4.3 %, dropping to 1.8% by the end of life. Leakage considerations are described in more detail in Appendix B.



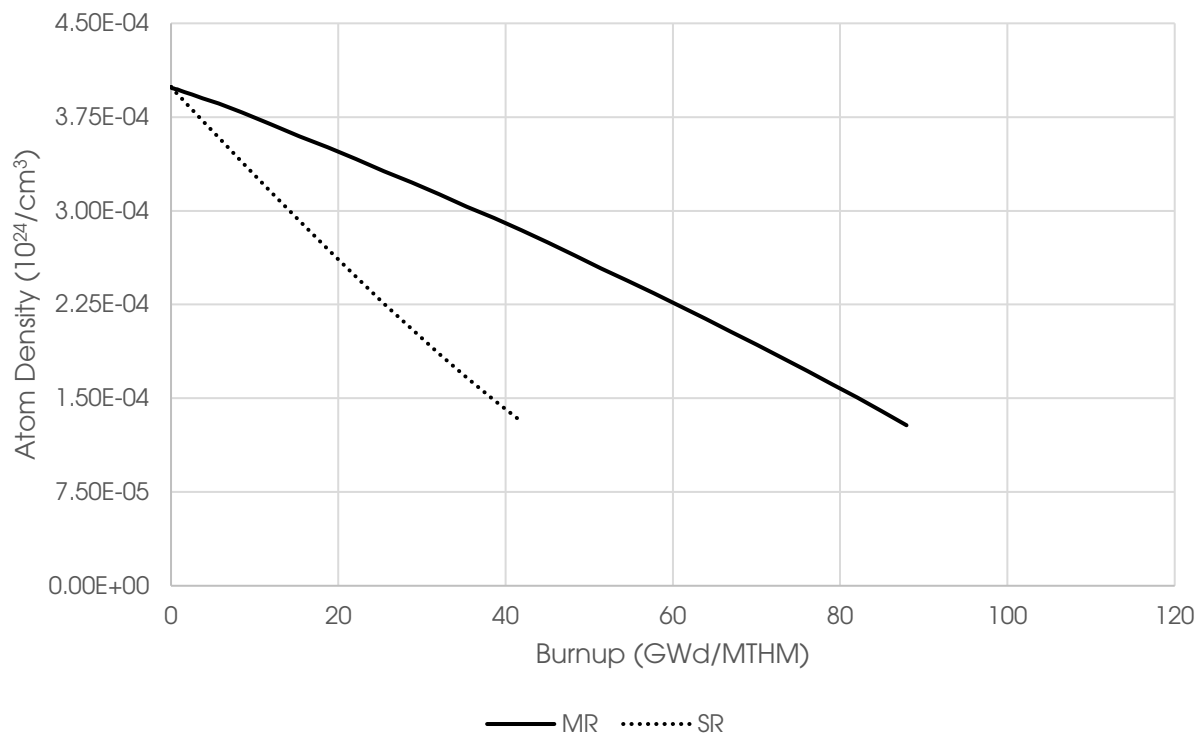


Figure 7. Fissile atom density as a function of burnup. The legend refers to the two possible operational mechanisms: Stationary Rods (SR) and Moveable Rods (MR). The data represent a summation of the ^{235}U , ^{239}Pu , and ^{241}Pu atom densities, with their respective individual trends being given in Appendix C.

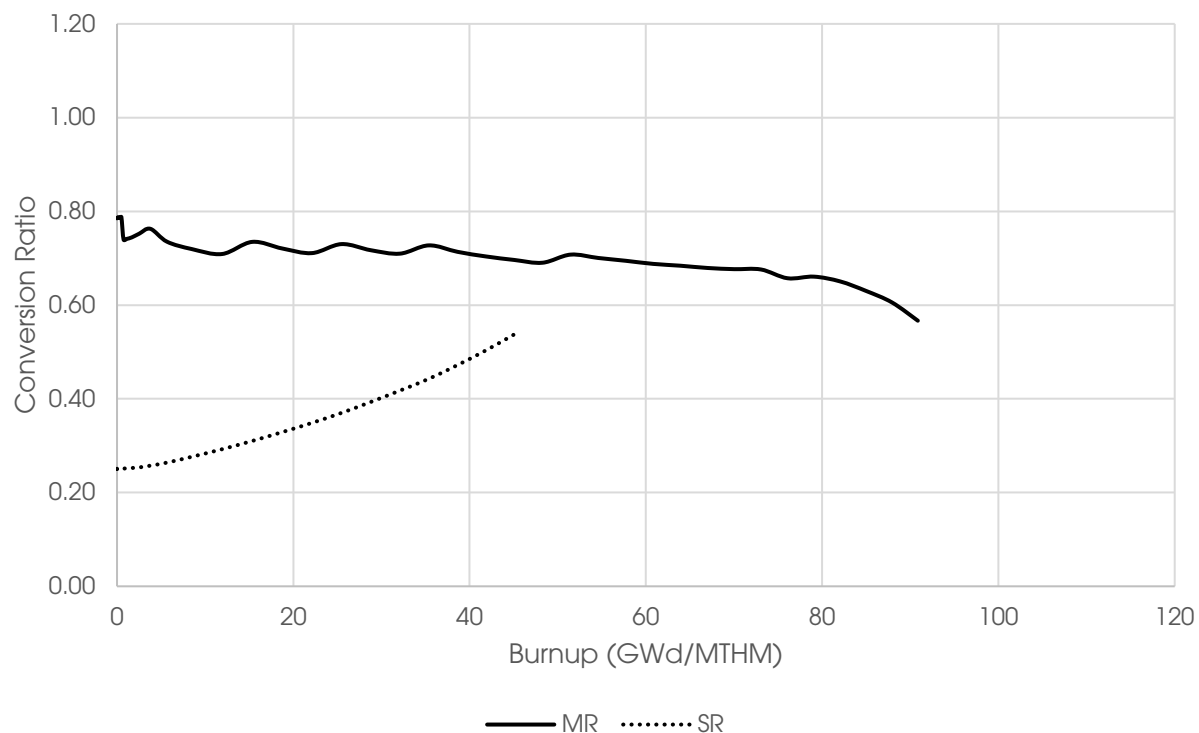


Figure 8. Conversion ratio as a function of burnup. The legend refers to the two possible operational mechanisms: Stationary Rods (SR) and Moveable Rods (MR). Additional calculation details are given in Appendix A.



WASTE REDUCTION & THE PATH FORWARD

Addressing nuclear energy's major challenge of long-lived waste disposal requires two substantial steps, as the industry must both reduce the rate at which waste is produced and find a long-term solution for the world's nuclear waste stockpiles. The TAP MSR's unique reactor design looks to address the first step, by maximizing burnup to reduce the waste production rate, and in doing so reduce the requirement for future repositories.

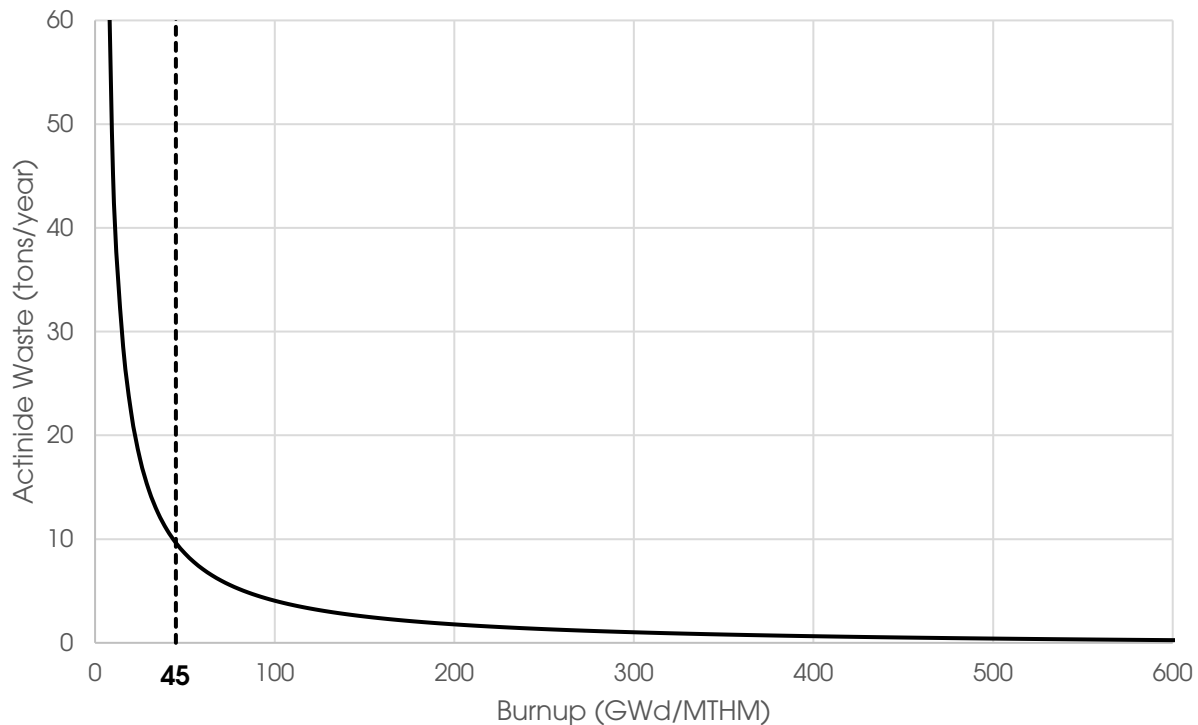


Figure 9. Annual actinide waste production as a function of burnup, normalized to a 1250 MWth power level.

As shown in Figure 9, fuel burnup and waste production exhibit an inverse power relationship, meaning that the effect of increased burnup on waste production depends highly on the reference position. With a 100% fuel utilization being equivalent to 909 GWd/MTHM and current LWR achieving burnups on the order of 45 GWd/MTHM, it is clear that there remains substantial room for improvement.

To fully explore TAP's goal of maximizing burnup in the TAP MSR, in addition to the fuel cycle discussed above, an identical depletion calculation was run for a fuel cycle in line with the enrichment levels proposed by other advanced reactor developers: a 10% enriched initial fuel load with a 20% enriched feed. Figure 10 compares the results of this calculation and those discussed previously in terms of the TAP MSR's improvement over current LWR technology.

As shown in Figure 10, higher enrichments lead to higher burnups, and therefore less waste, due to their ability to maintain criticality for longer periods of time. At the highest LEU enrichment levels, this ability results in an 83% reduction in annual waste production over LWRs. Even under the current industry fuel cycle regime of a 5% enriched initial load with a 5% feed, the TAP MSR reduces annual waste production by approximately 53%.



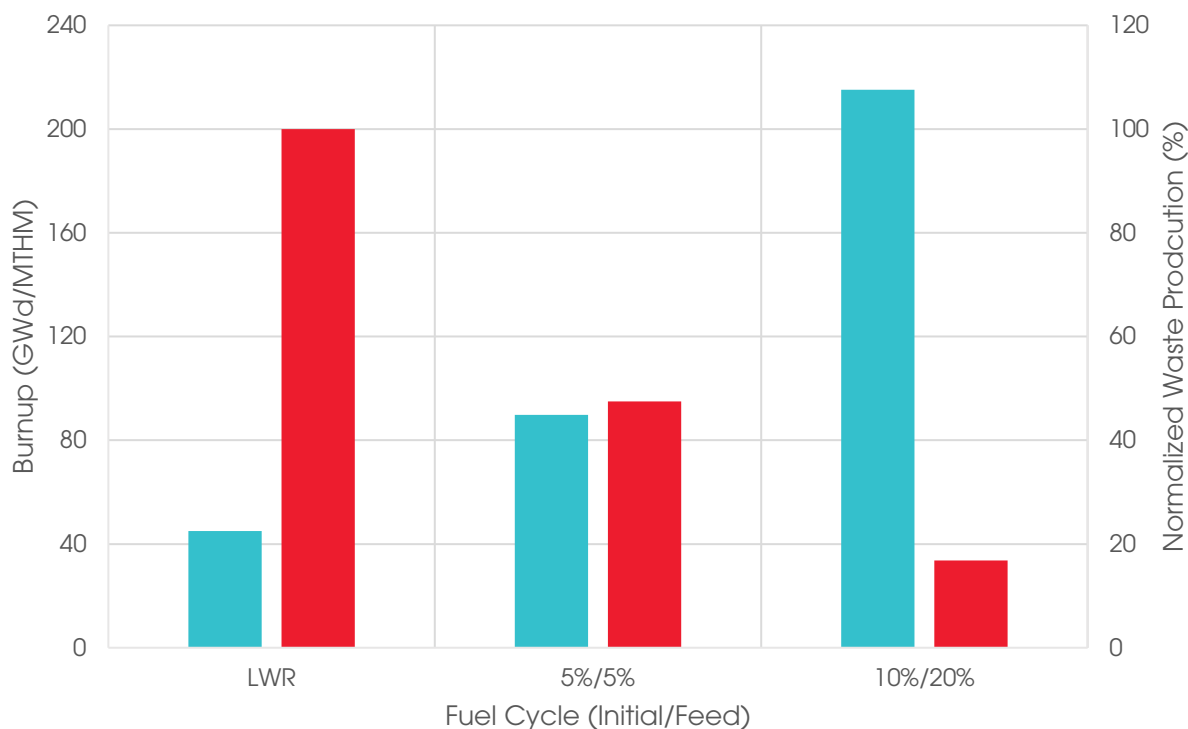


Figure 10. Burnup (left, blue) and normalized waste production (right, red) for the examined fuel cycles, normalized to a 1250 MWth power level.

Further details on the calculations used to determine the TAP MSR's fuel cycle profile can be found in Appendix D.

CONCLUSION

The TAP MSR design takes advantage of its liquid fuel to allow for more than twice the fuel utilization of light water reactors, leading to an approximately 53% reduction in waste when using 5% enriched uranium, the maximum enrichment readily available in the current supply chain. Using higher enrichments, up to the 20% LEU limit, this reduction reaches 83%. These accomplishments represent major leaps forward, inverting commonly-held beliefs about the nature of nuclear technology, and enabling nuclear energy to be a viable source of carbon-free baseload power.



REFERENCES

- [1] Transatomic Power Corporation, "Transatomic Technical White Paper," 2016.
- [2] R. G. Cochran and N. Tsoulfanidis, "In-Core Fuel Management," in *The Nuclear Fuel Cycle: Analysis and Management*, La Grange Park, Illinois, American Nuclear Society, 1999, pp. 165-205.
- [3] J. Leppänen, Development of a New Monte Carlo Reactor Physics Code, D.Sc., Helsinki University of Technology, 2007.
- [4] H. Alain, Applied Reactor Physics, Canada: Presses Internationales Polytechnique, 2009, pp. 231-235.



APPENDIX A: CALCULATION DETAILS

All data presented in this report were generated using the Monte Carlo neutronics software Serpent 2.26.

Geometry

The geometry modeled throughout the report is a 3-dimensional, 3 cm pitch, 7 by 7 rod assembly, as seen depicted in Figure A.1. The $\text{ZrH}_{1.66}$ moderator rod radius is 1.25 cm, and the rod insertion order used in the simulations is illustrated by the number located in the center of each rod, with 1 indicating the first rod inserted and 49 the last. It should be noted that 4 and 5 cm-pitch models produce similar results, and may be chosen moving forward to limit the total number of rods in the reactor.

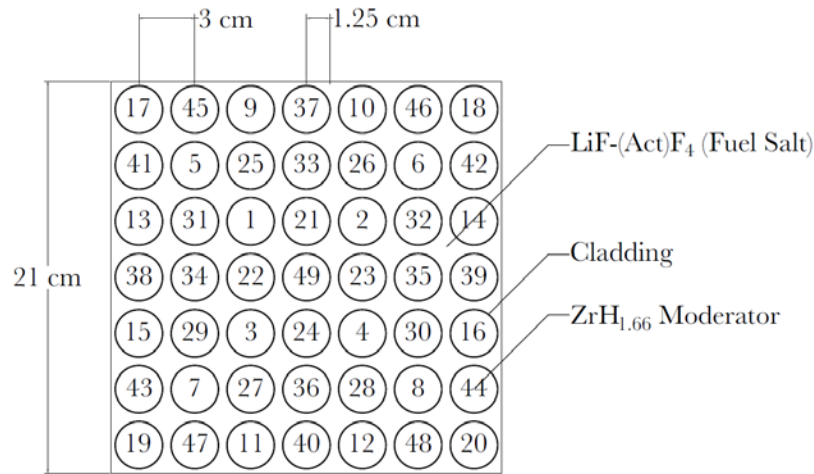


Figure A.1. Radial cross section of the assembly simulated in Serpent 2.26. The numbers located in the center of each rod indicate the insertion order used throughout the report, with 1 indicating the 1st rod inserted and 49 the last.

Materials

A summary of the material compositions used throughout the simulations is given in Table A.1. Calculations using graphite cladding were also performed and show similar results. For simplicity, only one fuel example (5% LEU) is presented in Table A.1.

Table A.1. A summary of material compositions and densities used in the simulations referenced throughout the report. The weight fractions listed are relative to the individual materials and are not representative of the system as a whole.

Fuel Salt (5.01 g/cm ³)		Silicon Carbide (3.21 g/cm ³)		Zirconium Hydride (5.66 g/cm ³)	
<i>Isotope</i>	<i>Weight Fraction</i>	<i>Isotope</i>	<i>Weight Fraction</i>	<i>Isotope</i>	<i>Weight Fraction</i>
²³⁵ U	3.1100E-02	¹² C	2.9936E-01	⁹⁰ Zr	4.9793E-01
²³⁸ U	5.9090E-01	²⁸ Si	6.4365E-01	⁹¹ Zr	1.0980E-01
⁷ Li	4.8358E-02	²⁹ Si	3.3866E-02	⁹² Zr	1.6967E-01
⁶ Li	2.4180E-06	³⁰ Si	2.3120E-02	⁹⁴ Zr	1.7569E-01
¹⁹ F	3.2964E-01			⁹⁶ Zr	2.8908E-02
				¹ H	1.8007E-02
				² H	4.1389E-06



Boundary Conditions

In order for the model to better represent the core configuration, the simulations invoked reflective boundary conditions, allowing for simulation of an infinite array of the defined assembly (Figure A.1). Leakage considerations as a result of this infinite lattice are discussed further in Appendix B.

Cross-Section Libraries

All cross sections used in the calculations were obtained through the evaluated nuclear data file data base ENDF-VII.1. Scattering kernels for zirconium hydride are not provided by default in Serpent 2.26, and therefore additional data from MCNP was converted to the appropriate format.

Neutron Population

All assembly level calculations discussed were modeled using a simulated population of 10,000 neutrons per cycle, for 30 inactive and 300 active cycles. The statistical error associated with this level of convergence is on the order of ± 0.0006 and 0.0004 for the analog and implicit k_∞ values of each time step respectively.

Fission Product Removal

The fission product removal rates used in the simulations can be seen below in Table A.2.

Table A.2. The fission product removal rates implemented in the depletion calculations discussed previously.

Element	Removal Rate (sec ⁻¹)	Element	Removal Rate (sec ⁻¹)	Element	Removal Rate (sec ⁻¹)
H	5.00E-2	Br	1.93E-7	Sn	5.79E-8
Ca	3.37E-9	Kr	5.00E-2	Sb	5.00E-2
Sc	3.37E-9	Rb	3.37E-9	Te	5.00E-2
Ti	3.37E-9	Sr	3.37E-9	I	1.93E-7
V	3.37E-9	Y	2.31E-7	Xe	5.00E-2
Cr	3.37E-9	Zr	5.79E-8	Cs	3.37E-9
Mn	3.37E-9	Nb	5.00E-2	Ba	3.37E-9
Fe	3.37E-9	Mo	5.00E-2	La	2.31E-7
Co	3.37E-9	Tc	5.00E-2	Ce	2.31E-7
Ni	3.37E-9	Ru	5.00E-2	Pr	2.31E-7
Cu	3.37E-9	Rh	5.00E-2	Nd	2.31E-7
Zn	3.37E-9	Pd	5.00E-2	Pm	2.31E-7
Ga	3.37E-9	Ag	5.00E-2	Sm	2.31E-7
Ge	3.37E-9	Cd	5.79E-8	Eu	2.31E-7
As	3.37E-9	In	5.79E-8	Gd	2.31E-7
Se	5.00E-2				



Time Step

In depletion calculations for systems out of equilibrium, it is important to ensure that the initial time steps are short enough to properly simulate the initial evolution of the different actinide vectors. As a result, exponential progression in the size of the steps, as shown in Equation A.1, is common practice, with the current simulations beginning with an initial step of 0.1 days (Δt_I), increasing by a factor of 1.5 for each subsequent step (n).

$$\Delta t = \Delta t_I \cdot 1.5^{n-1} \quad (\text{A.1})$$

Over the course of operation, fuel addition and SVF continually disrupt this aforementioned equilibrium, and it is therefore also important to limit the size of the maximum allowable time step. For all results discussed previously, 6 months was chosen as the maximum step limit ($\Delta t_{\max}$), with an even more accurate estimation into the final burnup values coming through the use of a Richardson extrapolation (A.2).

$$Bu = \frac{4Bu(\Delta t_{\max}) - Bu(2\Delta t_{\max})}{3} \quad (\text{A.2})$$

By running a second depletion calculation with a maximum time step of 12 month ($2\Delta t_{\max}$) and extrapolating in combination with the previous 6 month results, the uncertainty associated with the final burnup values is drastically reduced.



APPENDIX B: LEAKAGE CONSIDERATIONS

In order to limit excess reactivity and achieve a high conversion ratio at the beginning of life, the initial SVF must be very high, on the order of 0.9 for a 5% initial load. As a result, the neutron spectrum at the beginning of life is hardened compared to that of the end of life, as shown in Figure B.1.

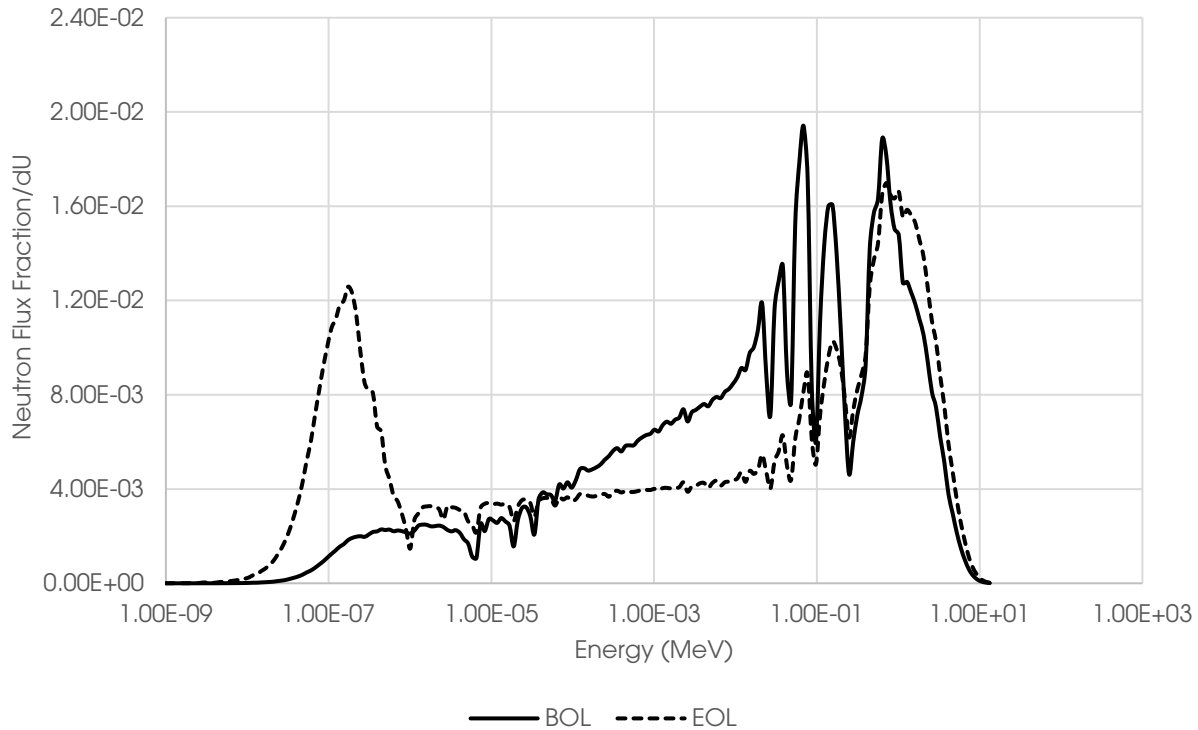


Figure B.1. Normalized neutron flux per lethargy as a function of energy at the beginning of life (BOL) and end of life (EOL). The spectra were taken from the depletion results of the 5% LEU initial load / 5% feed fuel cycle trial.

To accurately model such a spectrum, unresolved resonance probability tables were included in the calculations for several significant isotopes: ^{235}U , ^{238}U , ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Pu , and ^{242}Pu . The omission of other actinides in this consideration is not expected to introduce significant error, as their low concentrations over the course of operation minimize their overall influence.

As the leakage rate of the hardened spectrum is expected to be higher than that of the more thermal spectrum at the end of life, k_{∞} limits over the course of a depletion calculation must reflect this evolution. Figure B.2 shows the estimated k_{∞} trend as a function of the number of rods inserted in an assembly, with leakage considerations, both radially and axially, being obtained by comparing the k_{∞} of the infinite assembly to that of a representative three-dimensional core, as shown in Figure B.3.

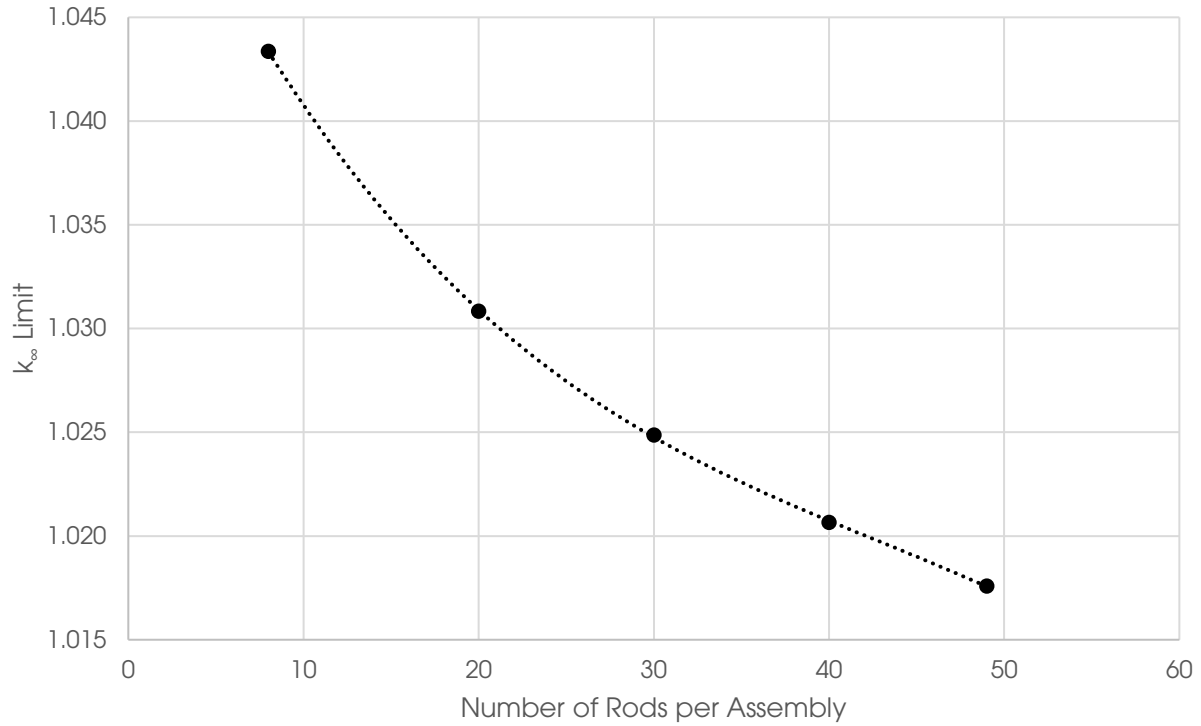


Figure B.2. Assembly level k_{∞} limit as a function of the number of rods inserted. Assemblies are comprised of a 7-by-7 array of $\text{ZrH}_{1.66}$ moderator rods.

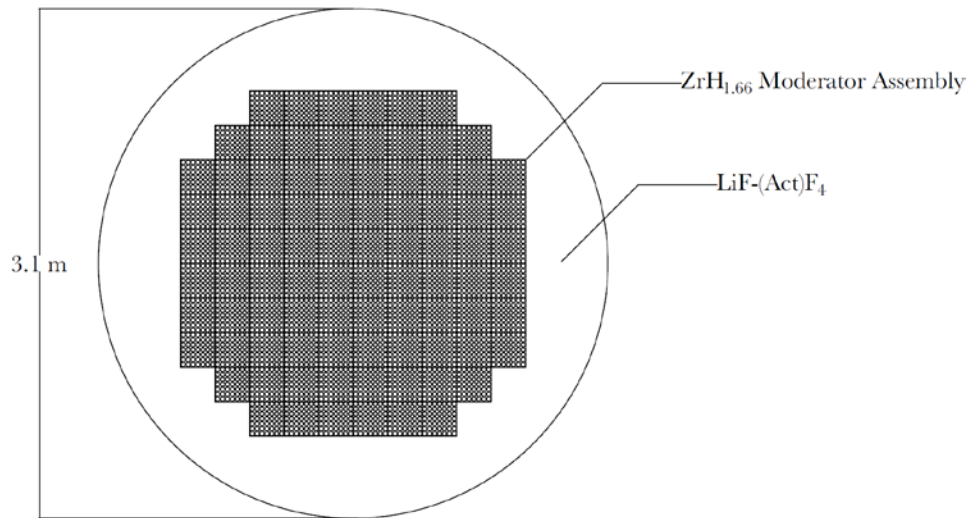


Figure B.3. A radial cross section of the EOL full core model used in the leakage calculations.

Acting as a complement to Figure B.2, Figures B.4 and B.5 show the radial flux profile of the reactor at the BOL for a 5% enriched initial actinide load, and the Shannon entropy as a function of the number of inactive cycles, respectively.

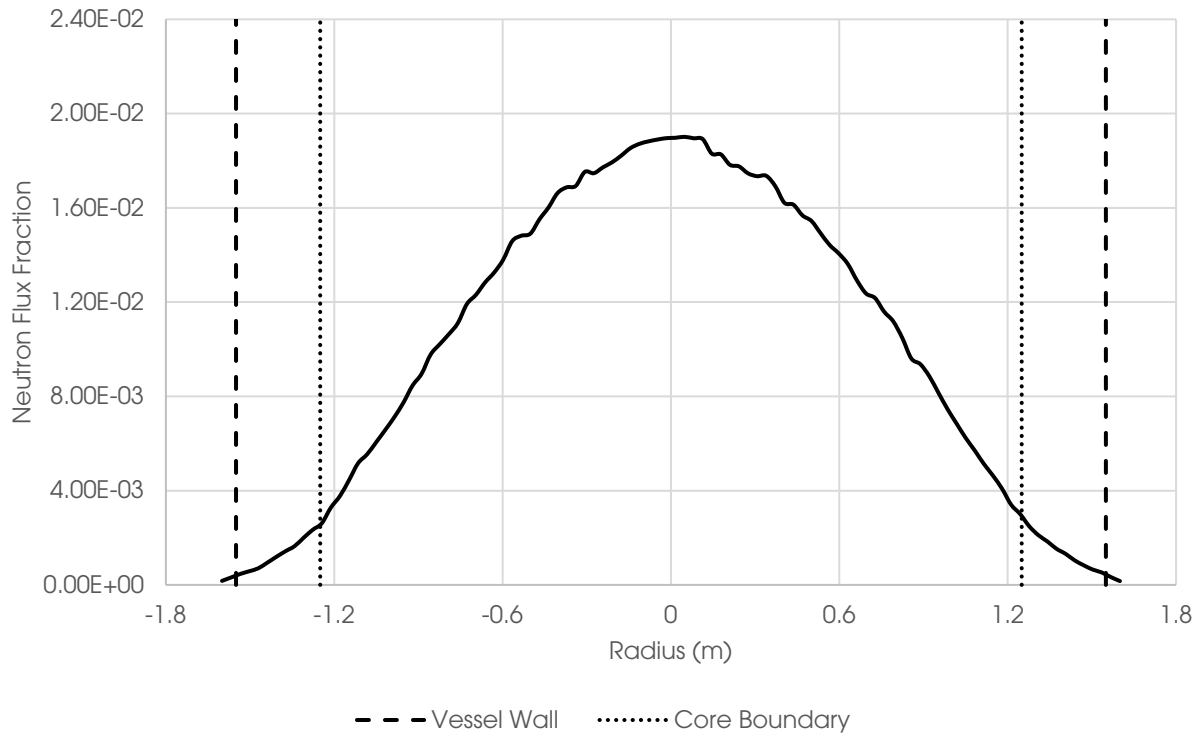


Figure B.4. Full core radial flux profile for a 5% enriched initial actinide load.

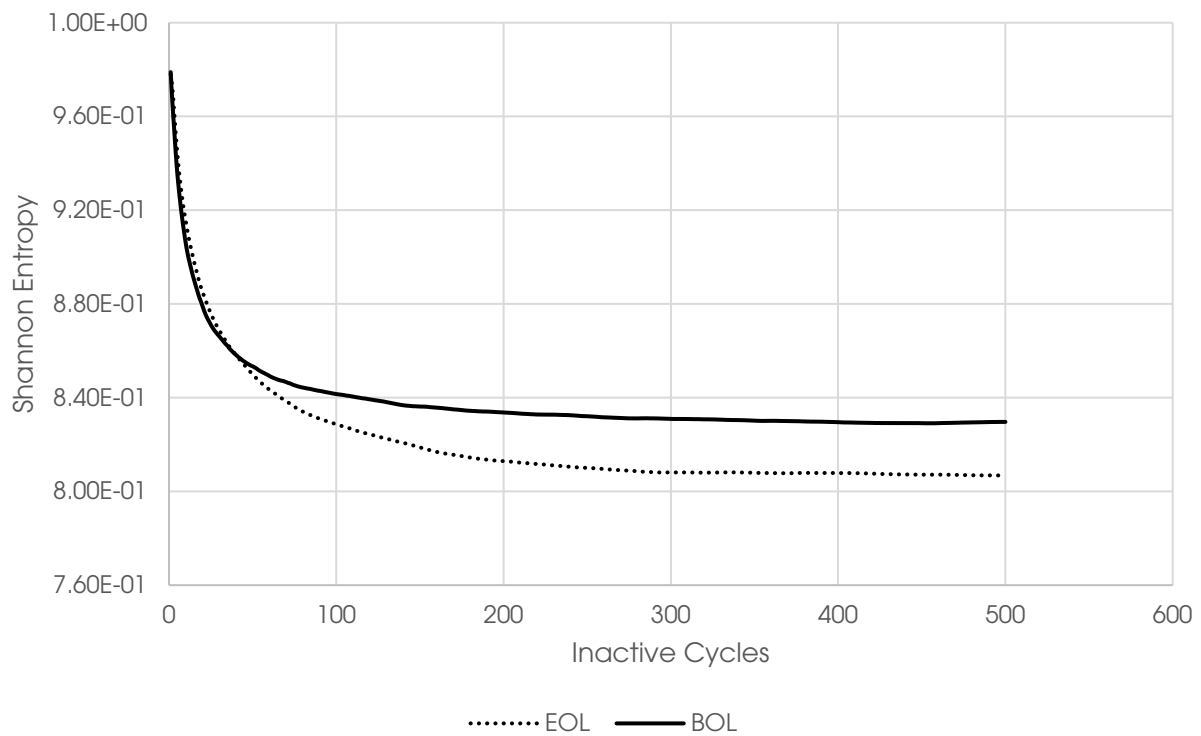


Figure B.5. Shannon entropy as a function of the number of inactive cycles for the full core leakage calculations.

APPENDIX C: ISOTOPIC EVOLUTION

In stationary rod configurations, conversion ratio is expected to increase over time as a result of the evolving plutonium vector: plutonium's fissile isotopes (e.g. ^{239}Pu and ^{241}Pu) produce both more neutrons, and greater energy per fission than that of ^{235}U , as well as ^{240}Pu possesses a significantly higher capture cross section at lower energies ($< 10^{-5}$ MeV) than that of ^{238}U . In a moveable rod method, this effect is outweighed by increased moderation, producing a thermalized spectrum at the end of life (see Figure B.1), which is unfavorable for conversion.

Figure C.1 shows the evolution of some of the primary fissile and fertile isotopes as a results of this changing neutron spectrum.

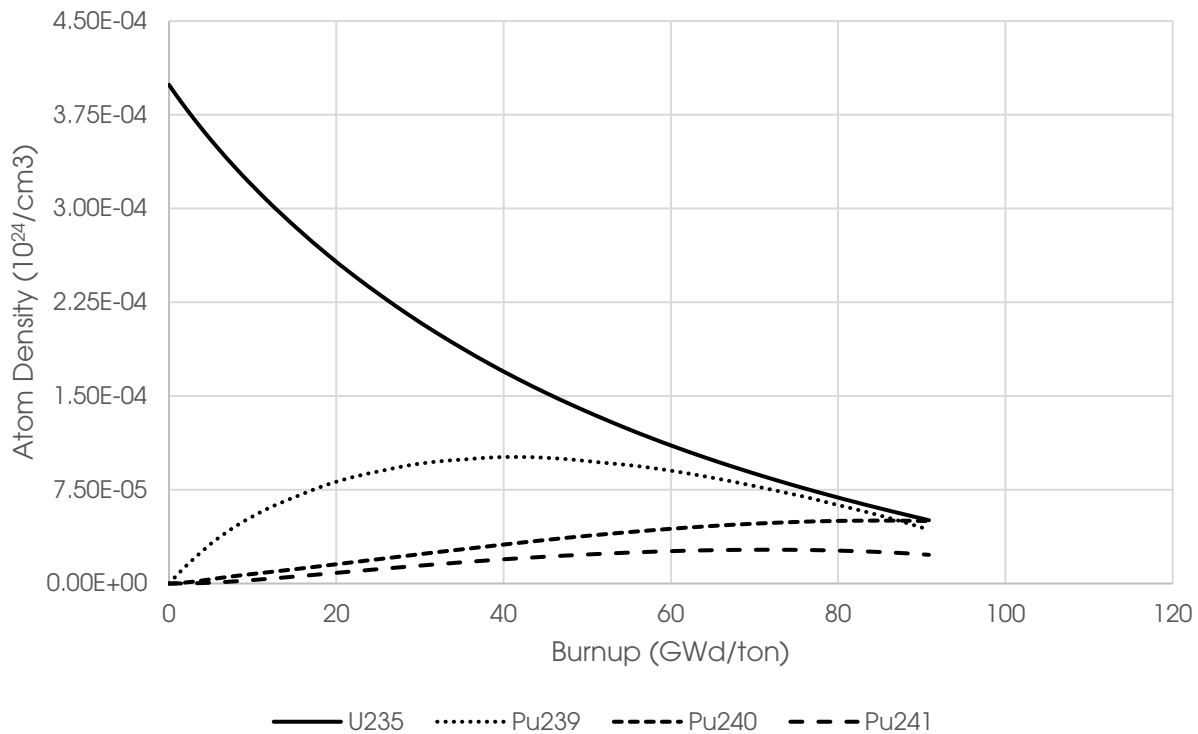


Figure C.1. The evolution of several primary fissile and fertile isotopes for the 5% initial load / 5% feed depletion simulation.

The effect of the moveable rods is most visible upon examination of the sharp decrease in ^{239}Pu . After roughly 30 GWd/MTHM, ^{239}Pu surpasses ^{235}U as the primary contributor to fission in the core, as shown in Table C.1. In addition to this larger consumption rate, ^{238}U capture can also be seen to drastically decrease over the course of life, as shown in Table C.2, reducing the production rate of ^{239}Pu .

Table C.1. Relative fission rate fraction for the three primary fissile isotopes as a function of burnup. The remaining fraction is a combination of ^{238}U fast fission and higher actinides.

Isotope	Burnup (GWd/MTHM)						
	0	15	30	45	60	75	90
^{235}U	0.93	0.56	0.42	0.31	0.24	0.20	0.18
^{239}Pu	0.00	0.33	0.43	0.50	0.52	0.52	0.50
^{241}Pu	0.00	0.03	0.06	0.11	0.15	0.19	0.25



Table C.2. Relative capture rate as a function of burnup for ^{238}U . The data are normalized to the ^{238}U capture rate at the beginning of life.

Isotope	Burnup (GWd/MTHM)						
	<i>0</i>	<i>15</i>	<i>30</i>	<i>45</i>	<i>60</i>	<i>75</i>	<i>90</i>
^{238}U	1	0.89	0.81	0.76	0.71	0.63	0.50



APPENDIX D: WASTE AS A FUNCTION OF BURNUP

The relationship between high level actinide waste production and burnup can be derived using the definitions given in Table D.1 and Equations D.1 – D.3. Table D.2 shows that four potential fuel cycles were evaluated in this analysis. Each fuel cycle contains different underlying assumptions about actinide mass and net waste – these assumptions are listed in Tables D.3 and D.4, respectively. Table D.5 gives the resulting net waste production rates for the four fuel cycles.

Table D.1. Variable Legend

Bu	Burnup	M_I	Initial actinide mass
CL	Cycle length	M_T	Total actinide cycle mass
$\bar{E}_F$	Average energy release per fission	N_A	Avogadro's number
$\bar{M}$	Average molar mass per fission	P	Thermal power
M_A	Added actinide mass	$\dot{W}_C$	Waste consumption rate
M_C	Consumed actinide mass	$\dot{W}_P$	Waste production rate
M_F	End of cycle actinide mass	$\dot{W}_{P,Net}$	Net waste production rate

$$\dot{W}_P = \frac{M_F}{CL} \quad (D.1)$$

$$Bu = \frac{P \cdot CL}{M_T} \quad (D.2)$$

$$M_C = \frac{P \cdot CL \cdot \bar{M}}{\bar{E}_F \cdot N_A} \quad (D.3)$$

Table D.2. Fuel Cycles

Case Number	Initial Load	Feed Material
1	LEU	---
2	LEU	LEU
3	LEU	SNF
4	SNF	SNF

Table D.3. Actinide mass assumptions used in deriving the waste production rate equations given in Table D.5.

Case Number	Mass Assumptions
1	$M_F = M_I - M_C$ & $M_T = M_I$
2, 3 & 4	$M_A \approx M_C$, $M_I \approx M_F$ & $M_T = M_I + M_A$

Table D.4. Net waste assumptions used in deriving the waste production rate equations given in Table D.5.

Case Number	Net Waste Assumptions
1, 2	$\dot{W}_{P,Net} = \dot{W}_P$
3	$\dot{W}_{P,Net} = \dot{W}_P - \dot{W}_C$ & $\dot{W}_C = \frac{M_A}{CL}$
4	$\dot{W}_{P,Net} = -\dot{W}_C$ & $\dot{W}_C = \frac{M_A}{CL}$



Table D.5. Waste production rate equations for the fuel cycles presented in Table D.2.

Case Number	Final Formulation
1 & 2	$\dot{W}_{P,Net} = \frac{P}{Bu} - \frac{P \cdot \bar{M}}{\bar{E}_F \cdot N_A}$
3	$\dot{W}_{P,Net} = \frac{P}{Bu} - 2 \frac{P \cdot \bar{M}}{\bar{E}_F \cdot N_A}$
4	$\dot{W}_{P,Net} = - \frac{P \cdot \bar{M}}{\bar{E}_F \cdot N_A}$

Case 4 has a negative waste production rate, which means that it consumes nuclear waste on net. In cases 1, 2, and 3, burnup appears directly in the equations for net waste production. Burnup does not appear directly in the equation for case 4, but it is still an important parameter for evaluating the fuel cycle. The waste produced by a TAP MSR has a lower fissile composition than the waste produced by a light water reactor (1.2% fissile versus 2% fissile), and therefore cannot be reused in subsequent core loadings. Maximizing burnup is therefore crucial for minimizing the quantity of new nuclear waste production, and achieving higher burnups will even more rapidly reduce worldwide stockpiles of nuclear waste.