Fuel cycle transition simulation capabilities in Cyclus

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Abstract

Recent interest in advanced reactors and the following need for techno-economic transitions has increased the demand for tools necessary to model complex Nuclear Fuel Cycles (NFCs) and advanced reactor technologies. This thesis demonstrates the capability of Cyclus, the agent-based fuel cycle simulator, to model, simulate, and analyze real-life fuel cycle transition scenarios. I introduce new methods and tools that use various databases to model and simulate real-world nuclear fuel cycle transition scenarios involving advanced reactor technologies.

The development of the capability takes three steps: (1) benchmarking Cyclus to other Nuclear Fuel Cycle Simulators (NFC Simulators); (2) developing new methods and tools necessary for modeling and simulating real-world fuel cycle transition scenarios; (3) real-world fuel cycle transition scenario modeling for France and the United States.

The methods and tools developed for such capabilities include: (1) modeling and simulating past and current nuclear fleet using historic nuclear reactor operations database; (2) modeling individual reactors and its operating history to calculate nuclear material inventory; (3) modeling Molten Salt Reactor (MSR) behavior in a large-scale fuel cycle simulation.

Benchmark results show that Cyclus results are in great agreement with results from other NFC Simulator. Additionally, this thesis demonstrates Cyclus capability to effectively model and simulate real-life NFC transition scenarios that involve advanced reactor technologies such as MSRs.

Acknowledgments

I consider all my output as a result of me standing on the shoulder of giants. There have been multiple giants that were kind enough to lend me their shoulders. Notably, the tallest of all, my advisor and mentor Kathryn Huff, who patiently guided me from my academic infancy. Also, I owe the world to my parents and loving brother who brought me up with love from my actual infancy. I also thank the staff of Oak Ridge National Laboratory Joushua L. Peterson-Droogh, Eva Davidson, Ben Betzler and Andrew Worrall for their guidance over the summer.

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Contents

List of Tables

List of Figures

Chapter 1

Introduction

The scope of this work includes development and demonstration of various methods and tools to leverage Cyclus' existing capabilities to model real-world fuel cycle transition scenarios.

1.1 Background and motivation

Increasing climate change concerns have directed attention to nuclear energy, which produces reliable base load energy with negligible CO_2 emission. With the reduction of fossil fuel based power plants and the general increase in energy demand (28% growth between 2015 and 2040 [1]), nuclear power is expected to play a crucial role in the world energy portfolio.

However, concerns of the accumulating UNF inventory, safety of the current reactor fleet, and the availability of uranium resources create a negative public perception of nuclear energy and its sustainability.

1.1.1 The Nuclear Fuel Cycle

The nuclear fuel cycle is described as a set of facilities that interact with one another to either provide or consume fuel services [2]. The goal of the cycle is to produce power economically, while managing UNF, the byproduct of power production. The discharge UNF from the reactors are eventually sent back to facilities for either recycling or disposal.

There are a large number of fuel cycle groups, categorized by the extent of recycling (no recycle, limited recycle, and continuous recycle), fuel composition (e.g. thorium-U233, uranium-plutonium), and the type of reactors (fast/thermal critical reactors, sub-critical Externally Driven Systems (EDS)). The fuel cycle evaluation and screening study by Wigeland et al. identified 40 evaluation groups [3].

Once-through fuel cycle

In a once-through cycle, nuclear fuel is used once and then sent to storage without further reprocessing [4]. This cycle is often called the open fuel cycle, and is the current cycle for most nations with nuclear energy (e.g. U.S., Korea, Finland, Sweden).

The cycle begins with mining of uranium ore, which is extracted from the ground. The mined ore is milled to form yellowcake (U_3O_8). The yellowcake is then either converted to UF_6 and enriched, or converted to UO_2 directly. This is because some reactor designs (e.g. Canada Deuterium Uraniums (CANDUs) [5]) can operate with natural uranium, while others (e.g. LWRs) need higher-than-natural levels of uranium-235. The processed UO_2 is then fabricated to pellets and loaded into fuel assemblies.

Once the fuel is depleted in the reactor, it is put in on-site pools to cool down. This process ranges from two to seven years. After cooling, the UNF is stored in dry casks as interim storage, destined to be sent to a geologic repository for permanent disposal.

Closed Fuel Cycle

In a closed fuel cycle, the UNF is recycled to be reused in a nuclear reactor. Recycling is not adopted worldwide due to its concerns of high cost and proliferation, but has two major benefits: increased fuel utilization and reduction of repository burden.

UNF discharged from a typical LWR has an approximate composition: 95% uranium, 1% plutonium, 0.1% minor actinides, and 4% fission products [6]. The uranium, plutonium, and the minor actinides have the capability to produce power through fission. Thus, every group except the fission products can be separated to create new fuel for other reactors.

Additionally, repository capacity is constrained mostly by decay heat load and radioactivity, meaning that removal of the high-activity isotopes leads to a more efficient utilization of the repository capacity. Short-lived fission products (e.g. cesium, strontium) contribute to a large heat load and radioactivity in the first 100 years of UNF disposal, and minor actinides (americium, plutonium), with their long half-lives, contribute to longer-term heat and radioactivity in the repository [7], as shown in figure 1.1.

There are two major reprocessing technologies: methods that use low-temperature chemical separation using organic solvents (e.g. PUREX [8]), and methods that use high-temperature molten salts and metals, like pyroprocessing [9]. These methods separate the UNF into different streams, which are then sent to either a high level waste (HLW) repository (fission products) or an appropriate fuel fabrication facility (plutonium).

Different closed fuel cycles use different elemental groups for recycled fuel fabrication. For example, the PUREX process is used in La Hague in France [10], THORP in the U.K [11], Mayak in Russia, and Rokkasho in Japan to separated plutonium and uranium [12]. The plutonium is mixed with either depleted uranium (tails) or reprocessed uranium to produce Mixed Oxide Fuel (MOX).

Closed fuel cycles generally involve fast-spectrum reactors to control TRU inventory. A fast-spectrum reactor can be designed to either burn (reduce TRU), breed

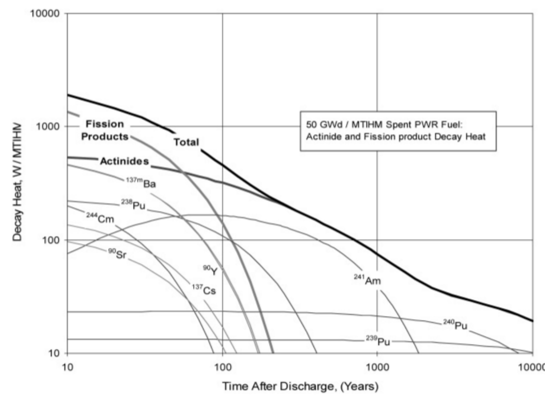

Figure 1.1: Decay heat contributions in UNF from a PWR irradiated to 50 GWd/MTHM [7].

(produce more TRU), or break-even (maintain TRU amount). Selection of the fast-spectrum reactor design depends on the goal of the deploying institution.

The fuel cycle deployed in this work is a cycle with continuous recycle of U/Pu or U/TRU with U fuel in fast critical reactors (Evaluation Group 23 or 24 [3]).

1.2 Objectives

This thesis demonstrates real-world NFC transition scenario modeling capabilities of Cyclus. The goal is to develop tools that leverage Cyclus' modularity to add capabilities required for modeling real-world fuel cycle transition scenarios, and demonstrate its capabilities by using the developed tools to perform NFC transition scenarios for France and the United States.

1.3 Methods

This thesis accomplishes the objective in three steps. First, the general fuel cycle simulation tool, Cyclus, is benchmarked to demonstrate its agreement with other fuel cycle simulation tools. Second, I identify the tools and methods necessary for modeling and simulating real-world transition scenarios. Finally, I construct and run fuel cycle transition scenarios for France and the United States.

A previous study by Feng et al. [13] validates existing NFC Simulators in a fuel cycle transition scenario, where a LWR fleet transitions into an SFR fleet with continuous reprocessing. This study compares four well-known NFC Simulators DYMOND [14], VISION [15], ORION [16], and MARKAL [17]. The results from each code were compared to a set of 'model solutions' that were generated from an excel worksheet for different metrics (e.g. fuel loading in reactor, UNF inventory). I reproduce the transition scenario in Cyclus, and compare the Cyclus results with those from the 'model solutions'.

In order to model real-world transition scenarios into an advanced fuel cycle, I developed two major tools. First, I developed a python module that automates extraction from the curated International Atomic Energy Agency (IAEA) Power Reactor Information System (PRIS) database [18]. The database lists each nuclear reactor's country, name, type, net capacity (Megawatt Electric (MWe)), status, operator, construction date, first criticality date, first grid date, commercial date, and shutdown date (if applicable). The module extracts the information from this file to generate a Cyclus compatible input file, which lists the individual reactor units as agents. Second, I developed a tool that models MSRs using a database generated from a high-fidelity MSR depletion calculation. The database is an output of Saltproc [??? Cite saltproc], a python module that drives SERPENT 2 [19] to model on-line reprocessing in an MSR. The HDF5 database contains the historic compositions of each stream in and out of the reactor, composition history inside the reactor, and keff values. The developed tool then reads the HDF5 file to mimic MSR behavior by requesting and offering feed and waste material to the Cyclus framework.

Finally I construct the fuel cycle transition scenario for France and the United States. I make different assumptions for the two scenarios to account for each nation's different goals, initial conditions (i.e. currently existing fleet, UNF inventory), and their potential reactor technology.

The structure of this thesis is as follows. In chapter 2, I review other fuel cycle simulation tools and their gaps, and explain why Cyclus has the unique capability for modeling real-world fuel cycle transition scenarios. Chapter 3 contains explanation and development of the tools that add the capabilities needed to accomplish the objective. Chapter 4 shows the results from the benchmark study, where Cyclus results are compared to results from other fuel cycle simulation tools. Chapter 5 and 6 shows the results from the France and United States fuel cycle transition scenario.

Chapter 2

Nuclear Fuel Cycle Simulators

NFC Simulators are system-level analysis tools that allow tracking of material flow in a NFC. Its functionalities include, but are not limited to, isotopic decay, depletion calculations, and separation of material streams. The goal of a NFC Simulator is to calculate 'metrics' - quantitative measures of performance that can be compared among fuel cycle options [20].

Table 2.1 lists the NFC Simulators that are considered in this section. The NFC Simulators listed generally are focused on one functionality (e.g. multi-regional analysis, detailed isotopic tracking, demand-driven deployment, cost analysis, sensitivity study) but lacks in the flexibility to perform other functionalities [21]. In other words, no NFC Simulator has all the functionalities to perform the superset of analysis types.

Table 2.1: List of NFC Simulators considered in this paper. Reproduced from [21]

Name	Developer	Reference(s)
CAFCA	MIT	[22]
COSI6	CEA (Frane)	[23]
DANESS	ANL	[24]
DESAE2.1	Rosatom (Russia)	[25]
EVOLCODE2	CIEMAT (Spain)	[26]
FAMILY21	JAEA (Japan)	[27]
GENIUSv1	INL	[28]
GENIUSv2	Univ of Wisconsin	[29]
NFCSS	IAEA	[30]
NFCSim	LANL	[31]
VISION	ANL/INL	[15]

2.1 Capabilities required for modeling transition scenarios

There are specific capabilities required for modeling transition scenarios. A study by Brown et al. [32] identified nine common functionalities for NFC Simulators for modeling transition scenarios - material compositions, deployment of fuel cycle facilities, front-end facility models, separations and material recycle facilities, reactor facilities, back-end features, starting the new fuel cycle, materials queuing and prioritization under capacity limitations, and energy demand algorithms. Brown et al. categorizes each functionality into three tiers - basic, integral, and exemplary. The functionalities, features, and their hierarchies are organized in table 2.2.

2.2 Additional capabilities identified for real-world fuel cycle transition scenario

From Brown et al. [32], I identified two additional functionalities for modeling real-world NFC transition scenario - modeling discrete, real-world reactor fleets and operational history; modeling liquid-fueled reactors with continuous reprocessing.

2.2.1 Modeling discrete, real-world reactor facilities

For modeling real-world nuclear fuel cycle transition scenarios, the initial condition (e.g. existing fissile inventory, existing reactor fleet) is important to strategize the transition scenario, such as reactor deployment scheme, fuel type, and reactor design. This requires the NFC Simulator to correctly model the current fleet and its remaining lifetime. The purpose of a fuel cycle is to produce power, thus the objective function of a fuel cycle simulation is generally power demand. A transition scenario involves advanced reactor (fast-spectrum reactors) filling in the gap of decommissioned LWRs. Once the energy supply of the current fleet is calculated, the analyst can determine the deployment scheme of future reactors to meet a certain power demand in the future. Thus, it is important for the NFC Simulator to be able to simulate discrete, real-world reactor fleets to have an accurate deployment scheme to meet power demand. Additionally having discrete facilities will translate into having a high-resolution calculation of the material flow as well.

Discrete reactor facility modeling

Discrete modeling of reactors allow a higher resolution of the power supply and material flow. In the real world, especially in the United States, existing reactors do not have the exact same power output, or core size. This means that lumping the reactor fleet together causes loss in accuracy.

NFC Simulators like COSI 6 [23], EVOLCODE [26], FAMILY21 [27], do have discrete facility modeling capabilities, while DESAE2.2[25], and VISION [15] do not [33]. Even for NFC Simulators with discrete modeling capabilities, some do not model distinct activities. For example, COSI models reactors operating in sync [34].

Similarly, most NFC Simulators do not treat disruption events (lack of fuel supply, or decommissioning of a reactor) discretely. For example, ORION shuts down the entire simulation if there is a lack of fuel supply, and cannot decommission reactors mid-cycle. DESAE 'borrows' lacking fuel from storage (leaving a negative value) instead of shutting down the reactor [35].

Past work on modeling real-world fleets

Modeling real-world fleets requires data about the current existing fleet, such as power capacity, core size, and expected shutdown date (remaining lifetime).

One published study is done modeling the current U.S. nuclear fleet, using ORION [36]. However, this work is far from modeling real-world U.S. nuclear fleet since it assumed a LWR deployed power capacity of 90 GWe in 2015, which decreases by 5 GWe every year starting from 2030, meaning that no consideration is given to the actual shutdown dates of existing reactors. This simplification stems from ORION modeling reactor as a fleet governed by a power demand, not discrete facilities.

There exists a study done on the French nuclear fuel cycle transition scenario by Carre et al [37], but it is unclear what tool they use. [I emailed him to find out.]

However, modeling real-world fleets is possible in Cyclus, due to two major reasons. First, Cyclus models discrete facilities with their own events and material flow. Second, Cyclus has a text-based input file structure, meaning that the input files (and thus the scenario) can be generated from a database, which is what is done in this work.

2.2.2 Modeling liquid-fueled reactors with continuous reprocessing

MSR reactor designs have recently gained attention due to their potential to be safer, more efficient, and sustainable [38]. Multiple companies in the U.S. are now pursuing commercialization of MSR design reactors, such as Transatomic [39] , Terrapower, Terrestrial [40], and Thorcon [41]. Other parties such as China (TMSR-LF [42]) and the European Union (MSFR [43], MOSART [44]) are developing MSR designs.

However, modeling a MSR is challenging due to its online reprocessing and continuously flowing fuel. The material flow in and out of the reactor is continuous and dynamic, as well as the composition inside the core. The neutronics and depletion calculations have to be performed continuously while the composition of the fuel changes by depletion and reprocessing. Reactor physics and depletion calculations on the MSR have been done, notably by Oak Ridge National Laboratory researchers who developed ChemTriton [45], a python script that drives SCALE, to perform semi-continuous reprocessing of the fuel [46, 47]. However, there is no existing NFC Simulator that has the capability to model MSRs due to the large computational burden associated with frequent depletion calculations.

This challenge of large computational time in a NFC Simulator can be overcome by 'outsourcing' the computationally heavy work to the higher-fidelity reactor physics and depletion codes. The output from the high-fidelity codes can be saved as a database, which then Cyclus reads to model the behavior corresponding to the MSR design. For example, a high-fidelity code would run a certain MSR design for its lifetime, and the history of its feed and waste recorded in a database. A Cyclus facility module would read this database and mimic the feed and removal behavior listed in the database, effectively modeling MSR interactions with the ‘market’. This allows MSR modeling in a larger-scale system analysis without heavy computational burden, while securing fidelity of the depletion calculation.

Table 2.2: Nine common functionalities identified for NFC Simulator to perform fuel cycle transition scenarios. Reproduced from Brown et al. [32]

Functionality	**Feature**	**Hierarchy**
Composition Features	Modeling of implicit consideration of fuel materials including primary fissile and fertile actinide isotopes	Basic
	Fuel's initial heavy metal mass modeled as lumped masses of the remaining actinides and fission products to conserve mass	Basic
	Isotopic decay of materials in storage	Exemplary
	Modeling of intermediate isotopes (e.g. Pa-233)	Exemplary
	Tracking of fission products beyond a simple lumped sum	Exemplary
	Modeling of compounding materials in fuels and waste forms	Exemplary
Fuel Cycle Facility Deployment	Facility deployment and retirement	Basic
	Construction time delays	Basic
	Strategic deployment to meet demand	Integral
Front-end Facilities	Source (mining and milling)	Basic
	Details of mines and mills including annual and total quantities available	Exemplary
	Conversion and enrichment facilities	Basic
	Timing and capacity of recycle facilities	Basic
	Fuel fabrication	Basic
	Time delays and losses in separations and fabrication	Basic
Separations and Material Recycling	Separations facilities may be required for UNF	Basic
	Cooling time	Basic
	Losses in separations	Basic
	Material selection from the UNF supply	Basic
Reactor Facility	Fueling: number of batches, cycle length and fuel per batch	Basic
	Multiple fuel types in reactor facility (driver, blanket)	Basic
	Pre-generated charge and discharge isotopic compositions	Basic
	Real time calculations based on reactor physics models	Exemplary
	Reactor facility lifetime, construction time, and decommissioning time	Basic
	Initial charge for first core and discharge for final core	Basic
Back-end	Cooling of used fuel	Basic
	Conservation of mass - consistency with charged mass and generated power	Basic
Fuel Cycle Startup	External source of fissile material	Basic
	Startup on recycled fuel from other facilities	Integral
	Primary and back-up fuel types	Exemplary
Material Prioritization	Material accumulation	Basic
	Material prioritization	Integral
	Radioactive decay	Exemplary
Energy Demand Algorithm	Technology allocation accounting for availability	Integral
	Ordering and deployment of multiple reactor technologies	Integral

Chapter 3

Tools Used for this work

3.1 Cyclus

CYCLUS is an agent-based fuel cycle simulation framework [20], meaning that each reactor, reprocessing plant, and fuel fabrication plant is modeled as an agent. A CYCLUS simulation contains prototypes, which are fuel cycle facility models (archetypes) with pre-defined parameters, that are deployed in the simulation as `facility` agents. Encapsulating the `facility` agents are the `Institution` and `Region`. A `Region` agent holds a set of `Institutions`. An `Institution` agent can deploy or decommission `facility` agents. The `Institution` agent is part of a `Region` agent, which can contain multiple `Institution` agents. Several versions of `Institution` and `Region` exist, varying in complexity and functions [48]. `DeployInst` is used as the institution archetype for this work, where the institution deploys agents at user-defined timesteps.

At each timestep, agents make requests for materials or bid to supply them and exchange with one another. A market-like mechanism called the dynamic resource exchange [2] governs the exchanges. For output analysis, each material resource has a quantity, composition, name, and a unique identifier.

In this work, each nation is represented as a distinct `Region` agent, that contains `Institution` agents, each deploying `Facility` agents. The `Institution` agents then deploy agents according to a user-defined deployment scheme.

The Cyclus NFC Simulator framework and its modeling ecosystem, the suite of agents and other physics plug-in libraries compatible with it, incorporate modern insights from simulation science and software architecture [20]. Cyclus has a multitude of benefits from other available NFC Simulators codes - open-sourceness, modularity, and extensibility. Its agent-based modeling approach is ideal for modeling coupled, physics dependent supply chain problems common in NFCs. The framework allows for dynamic loading of external libraries, which allows the users to plug-and-play different types of physics models for NFC simulation.

3.1.1 Open Source

License agreements and institutional approval is need for most NFC Simulators like COSI, DANESS, DESAE, EVOLCODE, FAMILY21, NFCSim, ORION and VISION [49], making it hard for both the usage and development in an academic settings. On the other hand, Cyclus relies completely on open source, free libraries, allowing all users to both use and develop the Cylcus framework and existing libraries. The open-sourceness of Cyclus encourages collaboration - any user can propose improvements or provide input for Cyclus.

3.1.2 Modularity and Extensibility

In most modern NFC Simulators, the facilities and their behavior (and its fidelity) is confined in the software. Also, most modern NFC Simulators are restricted to model a set of fuel cycles (once-through, continuous reprocessing) with restrictions to connections between facilities. On the other hand, Cyclus allows users to plug-and-play different libraries that contain different fidelity modules within the Cyclus framework (shown in figure 3.1). Also, the Cyclus simulation relies on a market-based model for material trades between facilities, so the user can design any novel fuel cycle. Also, this gives Cyclus the option to not be limited to nuclear fuel cycles, but any system analysis involving multiple connected facilities with physics-based calculations.

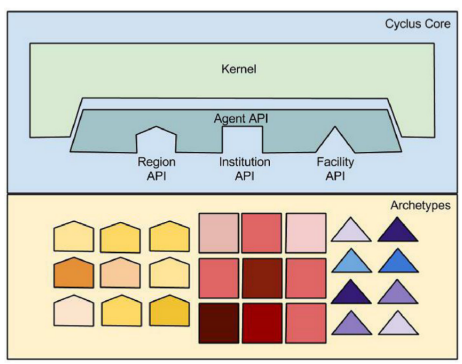

Figure 3.1: The Cyclus core provides APIs that allows the archetypes to be loaded into the simulation in a module fashion [20].

The connection between the framework and the agents is the dynamic resource exchange (DRE), where the agents interact with each other through material offers and requests. The kernel solves the material offers and requests, and executes the transaction between two agents.

3.1.3 Cyclus' fitness for real-world NFC transition scenario

The Cyclus framework and its extension libraries fulfill all the functionalities specified by Brown et al. [32]. Additionally, its text-based input structure and discrete facility modeling capabilities allow modeling of real-world, individual reactors. Finally, the addition of a MSR model can be achieved without altering Cyclus due to its modularity. The details of the implementation of these capabilities are shown in the next chapter.

3.2 SaltProc

SaltProc is the online reprocessing simulation driver for SERPENT2 [50], for simulating liquid-fueld MSR operation [??? new zenodo]. Saltproc uses a semi-continuous approach to simulate continuous MSR material feed and removal [51]. It is coded in Python (compatible both in 2.7 and 3.6), and outputs an HDF5 [52] database with feed, removal and in-core isotopic history.

SaltProc's structure and capabilities are similar to that of the ChemTriton tool for SCALE, developed in ORNL [53]. The computationally heavy work - Monte Carlo neutron transport and burnup calculations - are done in SERPENT, while SaltProc parses through the output material compositions, processes the fuel (removal and feed), and creates a new SERPENT input. The user can specify removal and feed rates and removal efficiencies for each material stream. At each timestep, the material compositions after the depletion calculation and after fuel processing are recorded in the database, as well as the feed and removal stream.

The logic flow of SaltProc is illustrated in figure 3.2. Initially, Saltproc reads a user-defined SERPENT 2 input file that contains input cards with parameters such as geometry, non-fuel components' composition, neutron population, criticality cycles, depletion time, total power, and boundary conditions. SERPENT 2 then performs neutron transport and depletion calculations, and returns the number density of the depleted fuel. Saltproc then reads the depleted composition, writes the composition in the database, processes the depleted material according to a user-defined scheme, and then outputs a new fuel composition input card for SERPENT 2. This again is then read by SERPENT 2 and the cycle continues until the user-defined timestep is reached.

One of the benefits of having a semi-continuous external driver for SERPENT 2 is that the user can set up SaltProc so that the density of a certain isotope in the fuel remains constant. In other words, the feed rate can vary over time to meet a certain 'quality' of the fuel. Also, using a Monte Carlo code such as SEPRENT allows users to vary in geometric fidelity, from a single cell model to a full core model.

3.2.1 Usage in this work

SaltProc's output, the HDF5 database can be imported through a Cyclus module, where the module can mimic the MSR feed and removal behavior throughout its lifetime. The composition in the core can be ignored, since the data of interest for a system-level NFC simulation is the material flow in and out of the reactor. This

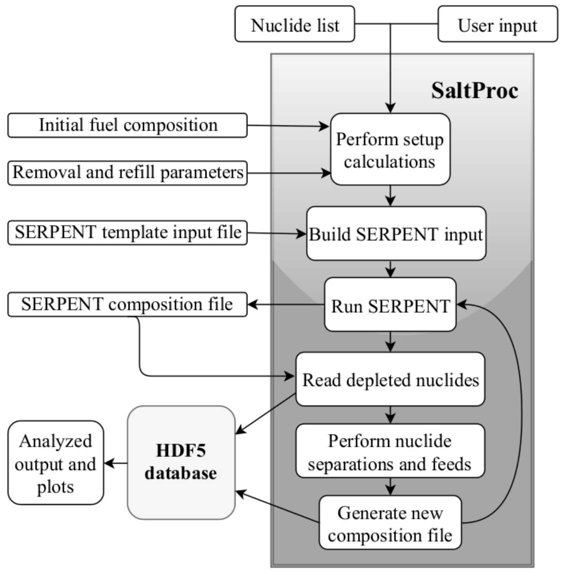

Figure 3.2: Flow chart for the Saltproc tool [51].

method can then effectively model MSRs in a large system-scale NFC simulation without a large computational burden for the NFC simulation.

Chapter 4

Tools developed for this work

I developed two extensions to leverage the capabilities of Cyclus to model real-world fuel cycle transition scenarios. The first extension is a python input-generating module that allows automated scenario generation of the real-world nuclear fleet at any point in time. The second extension is a Cyclus archetype that mimics MSR feed and removal behavior using a HDF5 database generated from SaltProc.

4.1 write_input.py

The objective for the `write_input.py` module is to automate the population of Cyclus input files to model the state of reactor fleets at a given point in time.

The module reads from the PRIS database and extracts data on each reactor's country, reactor unit, type, net capacity (MWe), status, operator, construction date, first criticality date, first grid date, commercial date, and shutdown date (if applicable). The user inputs simulation configurations such as start year, start month, and simulation duration. The module uses the collected data to fill out a template into a Cyclus input file. The logic flow of the module is shown in diagram 4.1

4.1.1 Assumptions

The module calculates the deployment scheme of the reactors and their lifetimes by assuming that all reactors shut down after 60 years of operation. If the expected shutdown date is later than the user-input simulation start date, the reactor is not written in the input. If the reactor was operational before the simulation start date, the reactor is deployed at the beginning of simulation with its remaining lifetime. If the reactor is reactor's start date is later than the simulation start date, and the shutdown time is undefined, the reactor is deployed at the defined start date with 60 years of lifetime.

The module calculates the core sizes of various reactor types by using a linear core size model. It assumes that reactor cores scale linearly from a model reactor design. The model reactor designs are listed in table **??**.

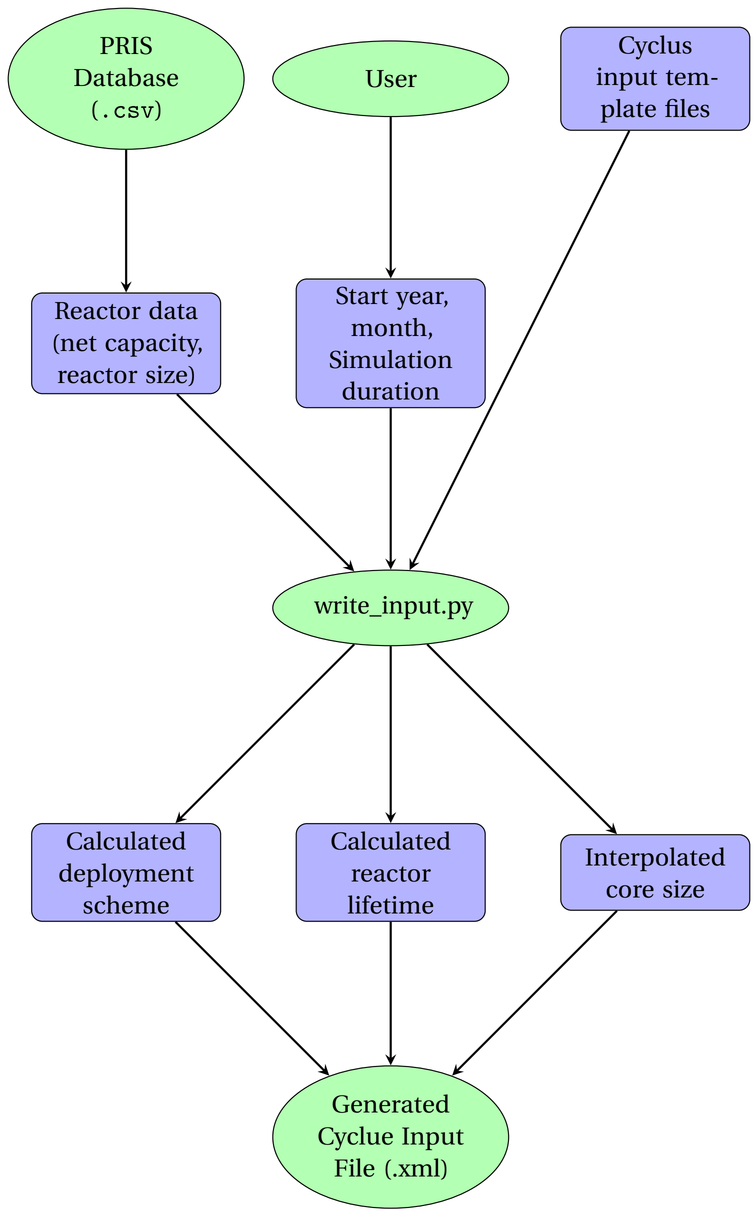

Figure 4.1: Logic flow of `write_input.py`. Green circles and blue boxes represent files and data, respectively.

Table 4.1: Reactor model designs used for the linear core size model.

Category	**Model Reactor**	**Power [MWe]**	**Assembly Mass [kg]**	**Assemblies in Core**	**Reference**
Generic PWR	AP-1000	1117	446	157	[54]
Generic BWR	4-MK I	1098	180	764	[**?**]
Generic PHWR	CANDU6	700	24.17	4,560	[**?**]

4.2 HDF5-reactor

The HDF5 reactor is a Cyclus facility archetype designed to model MSR behavior. It roughly couples Saltproc [51] and Cyclus, by using the output from SaltProc to mimic MSR feed and removal behavior in Cyclus. Most of the computationally heavy work is done in SERPENT (driven by Saltproc) in generation of the database, which avoids the large computational burden when running Cyclus. This method is similar to the simplified implementation of recipe reactors, where the depletion calculation is performed outside of the fuel cycle simulation. However, this reactor uses a database of recipes rather than a single recipe in order to model the continuously varying nature of liquid-fueled reactors like MSRs.

4.2.1 Code Description

The user inputs only of the commodity names for each stream, and the database path, since the HDF5 database already contains the notion of reactor design, reprocessing scheme, and other reactor parameters (shown in figure **??**). The commodity names are needed for reactor agent to communicate with other Cyclus agents in exchanging material.

At every timestep, The `HDF5 Reactor` calculates the cumulative material

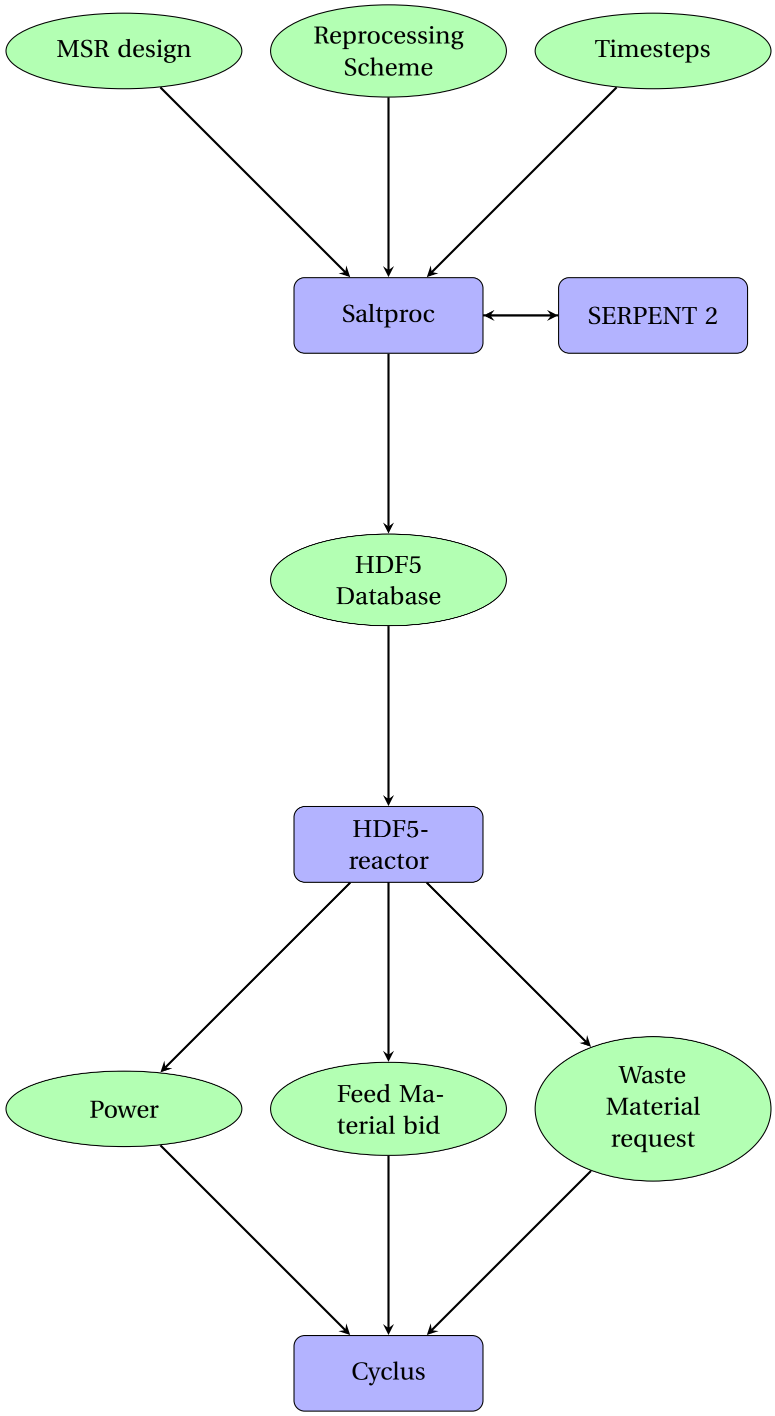

Figure 4.2: Logic flow of `HDF5 Reactor`. Green circles and blue boxes represent files and data, respectively.

Chapter 5

Modeling Real-World Nuclear Fuel Cycles

- Current policy (to transition)
- Current trend and interest (reactor design)
- Initial conditions (UNF inventory, Current fleet)
- Strategy for each (France - take UNF from other nations)
- Scenario parameters for each

Chapter 6

Cyclus benchmark Study

This chapter demonstrates Cyclus' agreement with other NFC Simulators by benchmarking the results of Cyclus to a previous verification study by Feng et al [13]. This verification study compared four well-known NFC Simulators DYMOND [14], VISION [15], ORION [16], and MARKAL [17]. The results from each code were compared to a set of 'model solutions' that were generated from an excel worksheet for different metrics (e.g. fuel loading in reactor, UNF inventory) in a transition scenario. I take the input parameters from this study and reproduce the transition scenario in Cyclus, and compare the results. Results show that Cylcus results are in good agreement with the results from Feng et al., with minor differences caused by reactor module behavior.

6.1 Methodology

Feng et al. comprehensively defines simulation parameters sufficient to reproduce the transition scenario in CYCLUS. In this study, we used the CYCAMORE [20] archetype library to model all fuel cycle facilities. CYCAMORE libraries contain simple fuel cycle facility models.

CYCLUS results are output in either `.sqlite` or `.h5` format. In this study, we used the `.sqlite` format and analyzed the results using a python script. The post-processed output data, was overlapped with the results with the model solution from the verification study [13]. The input file and analysis procedures are all in [zenodo].

The analysis and benchmark were performed iteratively, where we improve the original result by communicating with the authors of the benchmark. We analyzed the reasons for the differences from the original result, and made small edits in the source code. Major differences in the facility behavior algorithms were not edited but simply explained in detail as to how they contributed to the difference in the results.

6.2 Fundamental Modeling Differences in CYCLUS

CYCLUS has fundamental code differences from the fuel cycle analysis codes used in the benchmark [13].

CYCLUS has a default time step of a month. The verification study solutions are evaluated with 1-year time steps, so cumulative and annual averages were used. For example, decommissioning of facilities occurs at the end of a timestep, and building of facilities occurs at the beginning of a timestep.

The CYCAMORE recipe reactor depletes half of its core when decommissioned, whereas the codes in the benchmark [13] deplete all their reactors' fuel when decommissioned. This causes a major discrepancy for the TRU inventory. For this study, we changed the CYCAMORE source code to deplete all its assemblies to the depleted recipe. Also, the CYCAMORE recipe reactor treats each batch (and assembly) as a discrete material, while some codes have continuous fuel discharge. This produces differences in the results because the batches in the benchmark [13] are in time-averaged values. In this study, the LWR batch size and cycle time is increased, while decreasing the batch number to keep the core size constant. We round up the SFR batch number, while the batch size and cycle time are kept constant. This increases the core size by 1.08%, which is negligible, but will be discussed in the results section. The differences are listed in table 6.1.

Table 6.1: Difference in Batch number and core size

Category	**Model Solution [13]**	**CYCLUS**
LWR Batches	4.5	3
LWR Batch size [tHM]	19.91	29.86
LWR Core size [tHM]	89.59	89.59
LWR Cycle time	1 year	1.5 years
SFR Batches	3.96	4
SFR Batch size [tHM]	3.95	3.95
SFR Core size [tHM]	15.63	15.8

Note that all these differences could have been resolved by changing the archetype source code. However, the only change made was the reactor depletion behavior at decommission due to its large impact on plutonium inventory. The goal of this study is to show current CYCLUS agreement with other codes and identify differences, not to alter CYCLUS to match the other codes.

6.3 Results

We represent each CYCLUS result as a solid line, and the benchmark solution as a dotted line for visualization. The results are simply a reproduction of the plots displayed in the benchmark. We obtained the benchmark solutions through personal contact with benchmark author Bo Feng at Argonne National Laboratory.

Figure 6.1 shows the deployed reactor capacity, and figure 7.4 shows the LWR retirement and SFR deployment. The two plots show exact agreement with the benchmark solutions.

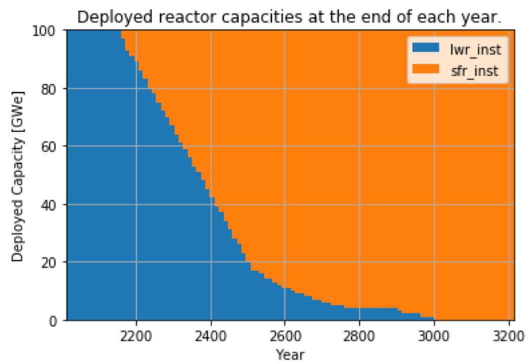

Figure 6.1: Deployed reactor capacities at the end of each year.

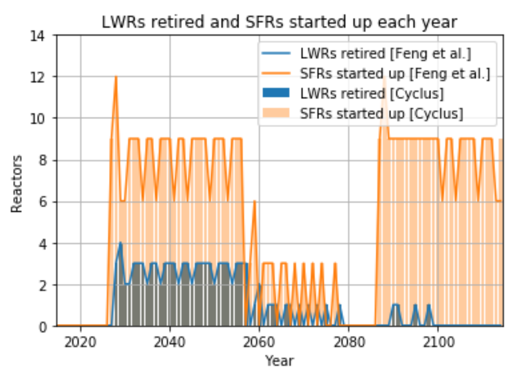

Figure 6.2: LWRs retired and SFRs started up each year.

Figure 6.3 shows the annual fuel loading rate. The initial fuel loading for 100 LWR reactors was edited to match the plot in the verification study results. The oscillations caused by the 18 month refueling period were aggregated into 12 month groups. As a result the total fuel loaded are equal for both plots.

Although indistinguishable in figure 6.3, there is a small difference between SFR fuel loading proportional to the core mass difference, as mentioned in the previous section. Figure 6.4 shows the differences normalized by the core mass differences, overlapped with the SFR deployment. This shows that the differences only occur during deployment due to the difference in core mass.

Figure 6.5 shows the inventory of discharged UNF in the mandatory cooling stage

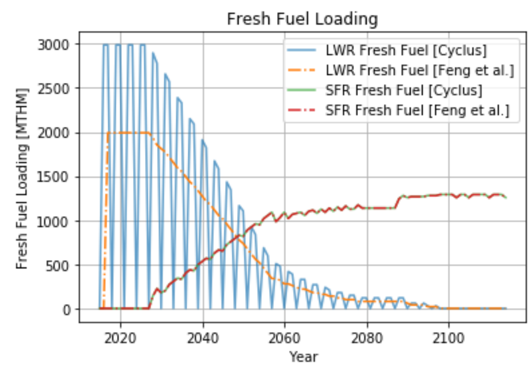

Figure 6.3: Annual fresh fuel loading rates (first cores and reload fuel).

(four years for LWR, one year for SFR). It also oscillates between the benchmark's solution and converges, caused by the influx and the outflux of UNF into and out of the storage facility. The SFR inventory and fuel loading solutions exactly matches the benchmark solutions, minus the small (1.07%) difference due to core size.

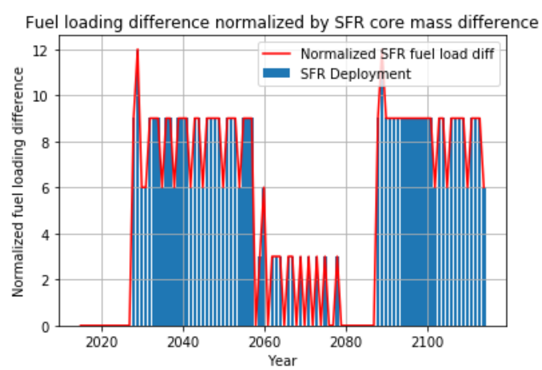

Figure 6.4: Difference between annual fresh SFR fuel loading rates (Cyclus - Benchmark) normalized by the core mass difference of an SFR due to fractional batch size.

Figure 6.6 shows the amount of cooled UNF waiting for reprocessing. The value is calculated by subtracting the cumulative difference between the cooled inventory and the UNF reprocessing throughput. The oscillation is between the cooled inventory in the storage facility before (high) and after (low) it sends its inventory for reprocessing.

Figure 6.7 shows the reprocessing throughput, which oscillates around the benchmark solution. No oscillation exists in the beginning because the LWR UNF reprocessing plant throughput peaks at 2,000 tons per year.

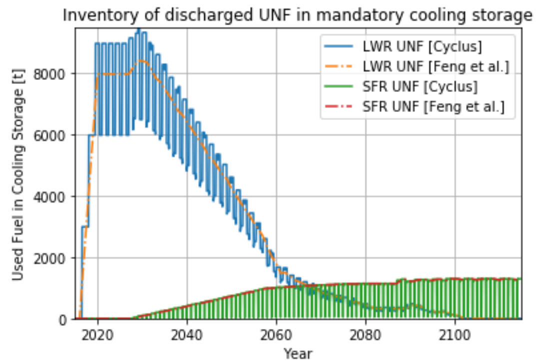

Figure 6.5: Inventory of discharged UNF in mandatory cooling storage.

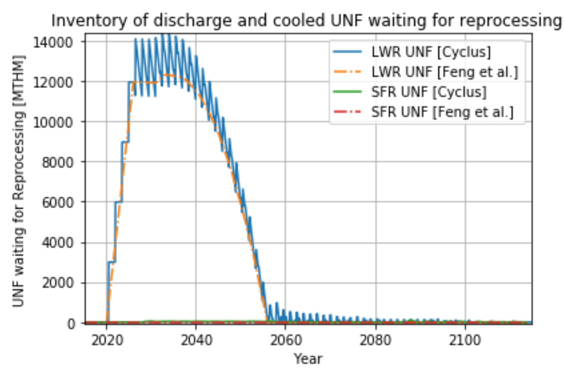

Figure 6.6: Inventory of discharged and cooled UNF waiting for reprocessing.

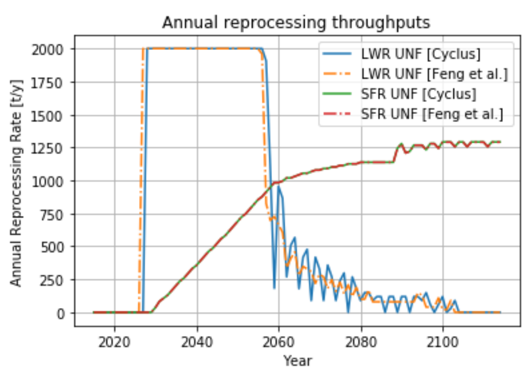

Figure 6.7: Annual reprocessing throughputs.

Figure 6.8 shows the inventory of unused TRU recovered from UNF. The CYCLUS results follow the benchmark solutions closely. However, the larger SFR core size causes CYCLUS results to be smaller than the benchmark results, since more TRU is used to start up the newly deployed SFRs. The difference decreases as the SFRs decommission, discharging more UNF (and hence TRU) than the benchmark.

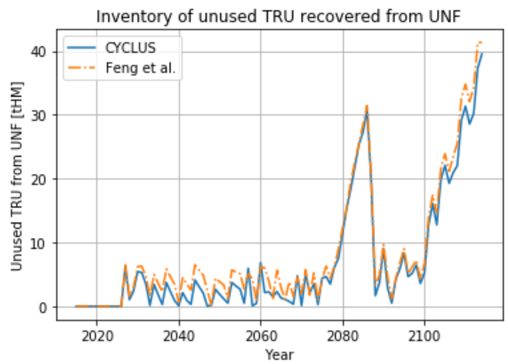

Figure 6.8: Inventory of unused TRU recovered from UNF.

6.4 Discussion

We benchmarked CYCLUS with results from an established verification study and saw good agreement in a transition scenario.

Throughout this work, two major differences were identified that led to the deviation of CYCLUS results from that of the benchmark solution. First, the CYCAMORE reactor depletes only half of its core when decommissioned. Second, CYCLUS, unlike other codes examined in the benchmark (except ORION), fully resolves discrete batches for fuel discharge. We resolve the first discrepancy by changing one line in the source code.

This study proves CYCLUS as a capable tool for modeling fuel cycle transition scenarios, and shows promise for expansion and future development.

Chapter 7

France

The stated long term plan for nuclear deployment in France targets a technology transition to SFRs[55]. However, the current inventory of French UNF is insufficient to fuel that transition without building new LWRs.

If instead, France accepted UNF from other EU nations and used it to produce MOX for new SFRs, the MOX created will fuel a French transition to an SFR fleet and allow France to avoid building additional LWRs.

To simulate this cooperative scenario, I simulated the entire EU region and all its nuclear reactor operating history and UNF accumulation up to the nearest foreseeable future. Then, France would take as much UNF it needs to transition into a fully SFR fleet without building additional LWRs.

This chapter includes the results for a French NFC transition scenario from a LWR fleet to a fully SFR fleet by taking other EU nations' UNF.

7.0.1 EU Deployment Schedule

The IAEA PRIS database [56] contains worldwide reactor operation history. The computational workflow in this work, shown in Figure 7.1, automates data extraction from the PRIS database. We import this database directly as a `csv` file to populate the simulation with deployment information, listing the country, reactor unit, type, net capacity (MWe), status, operator, construction date, first criticality date, first grid date, commercial date, shutdown date (if applicable), and unit capacity factor for 2013. Then only the EU countries are extracted from the `csv` file. We developed a python script to generate a CYCLUS compatible input file accordingly, which lists the individual reactor units as agents.

Projections of future reactor deployment in this simulation are based on assessment of analyses from references, for instance PRIS, for reactors planned for construction [56], the World Nuclear Association [57], and literature concerning the future of nuclear power in a global [58] and European context [59]. Existing projections extend to 2050.

Table 7.1 lists the reactors that are currently planned or under construction in the EU. In the simulation, all planned constructions are completed without delay or

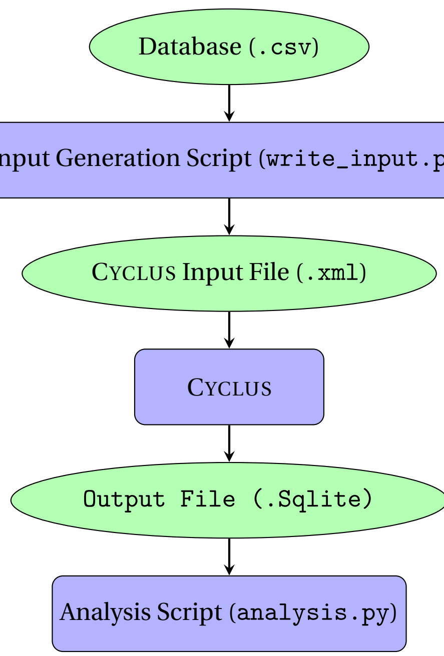

Figure 7.1: Green circles and blue boxes represent files and software processes, respectively, in the computational workflow.

failure and reach a lifetime of 60 years.

Table 7.1: Power reactors under construction and planned. Replicated from [57].

Exp. Operational	Country	Reactor	Type	Gross MWe
2018	Slovakia	Mochovce 3	PWR	440
2018	Slovakia	Mochovce 4	PWR	440
2018	France	Flamanville 3	PWR	1600
2018	Finland	Olkilouto 3	PWR	1720
2019	Romania	Cernavoda 3	PHWR	720
2020	Romania	Cernavoda 4	PHWR	720
2024	Finland	Hanhikivi	VVER1200	1200
2024	Hungary	Paks 5	VVER1200	1200
2025	Hungary	Paks 6	VVER1200	1200
2025	Bulgaria	Kozloduy 7	[1]AP1000	950
2026	UK	Hinkley Point C1	EPR	1670
2027	UK	Hinkley Point C2	EPR	1670
2029	Poland	Choczewo	N/A	3000
2035	Poland	N/A	N/A	3000
2035	Czech Rep	Dukovany 5	N/A	1200
2035	Czech Rep	Temelin 3	AP1000	1200
2040	Czech Rep	Temelin 4	AP1000	1200

For each EU nation, we categorize the growth trajectory is categorized from "Aggressive Growth" to "Aggressive Shutdown". "Aggressive growth" is characterized by a rigorous expansion of nuclear power, while "Aggressive Shutdown" is characterized as a transition to rapidly de-nuclearize the nation's electric grid. We categorize each nation's growth trajectory into five degrees depending on G, the growth trajectory metric:

$$G = \left\{ \begin{array}{ll} \text{Aggressive Growth,} & \text{for } G \geq 2 \\ \text{Modest Growth,} & \text{for } 1.2 \leq G < 2 \\ \text{Maintanence,} & \text{for } 0.8 \leq G < 1.2 \\ \text{Modest Reduction,} & \text{for } 0.5 \leq G < 0.8 \\ \text{Aggressive Reduction,} & \text{for } G \leq 0.5 \end{array} \right\} = \frac{C_{2040}}{C_{2017}}$$

$$G = \text{Growth Trajectory } [-]$$

$$C_i = \text{Nuclear Capacity in Year i } [MWe].$$

The growth trajectory and specific plan of each nation in the EU is listed in Table 7.2.

[1]The fate of many planned reactors is uncertain. The proposed reactor types are also unclear. The ones marked 'N/A' for type are assumed to the PWRs in the simulation.

Table 7.2: Projected nuclear power strategies of EU nations [57]

Nation	Growth Trajectory	Specific Plan
UK	Aggressive Growth	13 units (17,900 MWe) by 2030.
Poland	Aggressive Growth	Additional 6,000 MWe by 2035.
Hungary	Aggressive Growth	Additional 2,400 MWe by 2025.
Finland	Modest Growth	Additional 2,920 MWe by 2024.
Slovakia	Modest Growth	Additional 942 MWe by 2025.
Bulgaria	Modest Growth	Additional 1,000 MWe by 2035.
Romania	Modest Growth	Additional 1,440 MWe by 2020.
Czech Rep.	Modest Growth	Additional 2,400 MWe by 2035.
France	Modest Reduction	No expansion or early shutdown.
Slovenia	Modest Reduction	No expansion or early shutdown.
Netherlands	Modest Reduction	No expansion or early shutdown.
Lithuania	Modest Reduction	No expansion or early shutdown.
Spain	Modest Reduction	No expansion or early shutdown.
Italy	Modest Reduction	No expansion or early shutdown.
Belgium	Aggressive Reduction	All shut down 2025.
Sweden	Aggressive Reduction	All shut down 2050.
Germany	Aggressive Reduction	All shut down by 2022.

Using this categorization to drive facility deployment, the simulation captures regional differences in reactor power capacity and UNF production as a function of time. Accordingly, fig. 7.2 shows the resulting simulated installed capacity in EU nations. Sudden capacity reductions seen in the 2040s result from end-of-license reactor retirements and nuclear phaseout plans in nations such as Germany and Belgium.

7.0.2 French SFR Deployment Schedule

Figure 7.3 shows the French transition to SFRs modeled in this simulation. Historically aggressive growth of nuclear in the 1980s leads to a substantial shutdown of nuclear in the 2040s, which, in the simulation, are replaced by new SFRs. The net capacity is kept constant at 66 GWe.

Figure 7.4 shows the deployment required to support the transition in fig. 7.3. France must build four reactors per year, on average, to make up for the end-of-license decommissioning of power plants built in the 1980s and 1990s. The second period of aggressive building occurs when the first generation of SFRs decommission after 80 years. Starting in 2040, France deploys 600-MWe SFRs to make up for decommissioned French LWR capacity. This results in an installed SFR capacity of 66,000 MWe by 2078 when the final LWR is decommissioned.

Finally, Figure 7.5 shows the total deployment scheme we simulated. The French transition to SFRs couples with the historical and projected operation of EU reactors. The steep transition from 2040 to 2060 reflects the scheduled decommissioning of reactors built in the 1975-2000 era of aggressive nuclear growth in France.

These figures reflect that, for the given assumptions, bursts of construction are necessary to maintain capacity. In reality, a construction rate of five reactors every

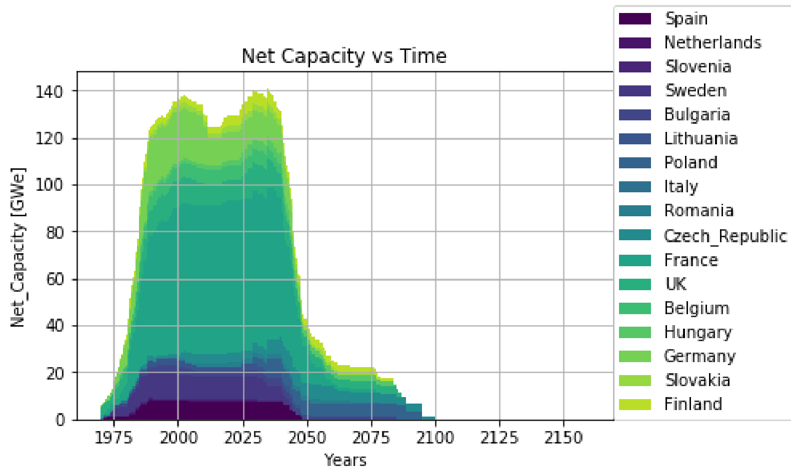

Figure 7.2: Installed nuclear capacity in the EU is distinguished by `Regions` in CYCLUS.

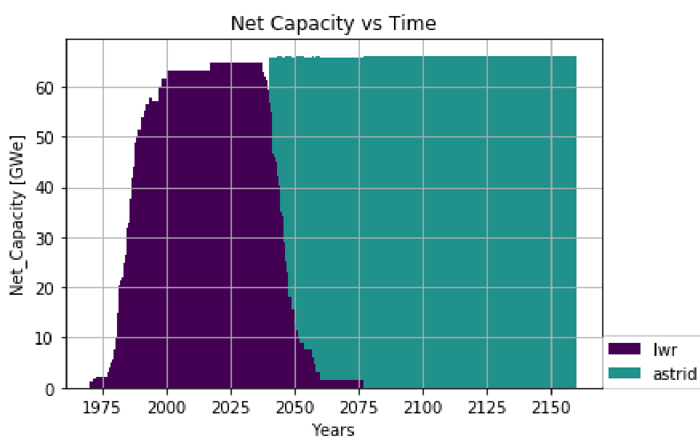

Figure 7.3: The potential French transition from LWRs to SFRs when assisted by UNF from other EU nations.

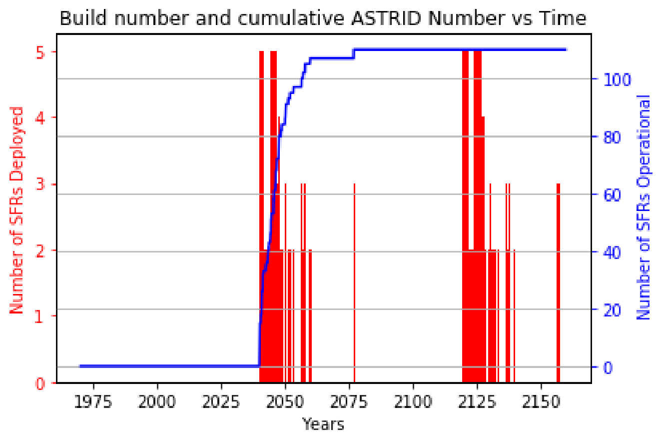

Figure 7.4: The deployment of SFRs in France is characterized by a period of aggressive building.

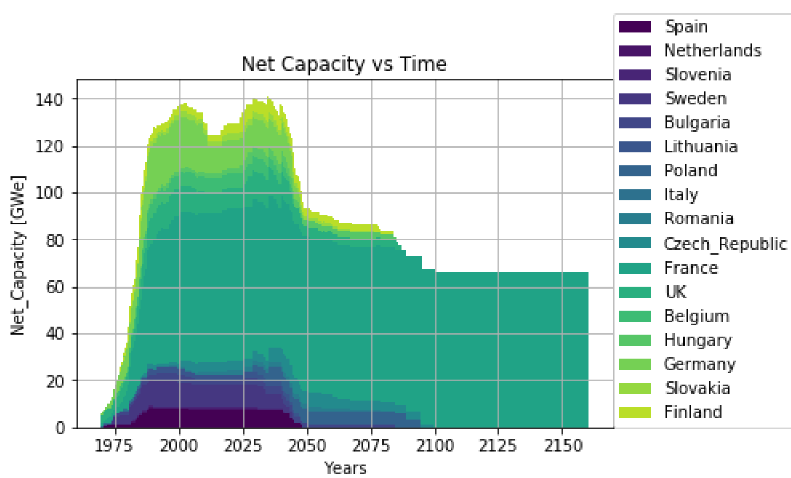

Figure 7.5: The total deployment scheme we simulated relies on UNF collaboration among nations.

year is ambitious, but might have the advantage of larger scale production of components and more modular assembly and construction if major components can mostly be built off site.

This analysis establishes a multi-national material flow and demonstrates that, if such an aggressive deployment scheme took place, the SFRs would have enough fuel.

7.0.3 Scenario Specification

The scenario specifications defining the simulations presented in this work are listed in table 7.3. The reprocessing and MOX fabrication capacity in France prior to 2020 is modeled after the French La Hague and MELOX sites [10, 60].

Table 7.3: Simulation Specifications

Specification	**Value**	**Units**
Simulation Starts	1970	year
Simulation Ends	2160	year
Production of ASTRID fuel begins	2020	year
SFRs become available	2040	year
Reprocessed uranium usage	Not used anywhere	-
Minimum UNF cooling time	36	months
Separation efficiency of U and Pu	99.8	%
Reprocessing streams	Pu and U	-
Reprocessing capacity before 2020	91.6 [10]	$\frac{\text{metric tons of UNF}}{month}$
Reprocessing capacity after 2020	183.2	$\frac{\text{metric tons of UNF}}{month}$
LWR MOX fabrication throughput	16.25 [60]	$\frac{\text{metric tons of MOX}}{month}$
ASTRID MOX fabrication throughput	No limit (∞)	$\frac{\text{metric tons of MOX}}{month}$
LWR MOX recycling	Not reprocessed	-
ASTRID MOX recycling	∞-pass	-

7.1 Reactor Specifications

Three major reactors are used in the simulation, PWR, BWR, and ASTRID-type SFR reactors.

For LWRs, we used a linear core size model to capture varying reactor capacity. For example, a 1,200 MWe PWR has $193 * \frac{1,200}{1,000} = 232$ Uranium Oxide Fuel (UOX) assemblies, each weighing 523.4 kg. After each 18 month cycle, one-third of the core (77 assemblies) discharges. Refueling is assumed to take two months to complete, during which the reactor is shut down. The specifications are defined in table 7.4 which details the reactor specifications in this simulation. LWR specifications are modified linearly for varying power capacity.

Table 7.4: Baseline LWR and ASTRID simulation specifications.

Specification	**PWR [61]**	**BWR [62]**	**SFR** [63]
Lifetime [y] [2]	60	60	80
Cycle Time [mos.]	18	18	12
Refueling Outage [mos.]	2	2	2
Rated Power [MWe]	1000	1000	600
Assembly mass [kg]	523.4	180	–
Batch mass [kg]	–	–	5,568
Discharge Burnup [GWd/tHM]	51	51	105
Assemblies per core [3]	193	764	–
Batches per core	3	3	4
Initial Fissile Loading [t]	3.1 ^{235}U	4.2 ^{235}U	4.9 Pu
Fuel	UOX or MOX	UOX	MOX

7.1.1 Material Definitions

Depletion calculations of the nuclear fuel are recipe-based, such that a fresh and used fuel recipe is defined for each reactor type. For the compositions of the used fuel, a reference depletion calculation from ORIGEN is used (see table 1). ORIGEN calculates buildup, decay, and processing of radioactive materials [64]. This recipe recipe has also been used for repository performance modeling [65].

Table 7.5: Fresh fuel compositions in the simulation [65, 63].

	Composition [%]		
Recipe	U-235	U-238	Pu
Fresh UOX Fuel	3.1	96.9	-
Fresh LWR MOX Fuel	0.2	90.7	9.1
Fresh ASTRID Fuel	0.2	77.7	22

[2] The simulated reactor lifetime reaches the licensed lifetime unless the reactor is shut down prematurely.

[3] Number of assemblies and corresponding LWR core masses are reported for a 1000-MWe core. Reactors with different core powers are modeled with a linear mass assumption.

7.1.2 Results - Transition Scenario

This section displays the simulation results if France utilized UNF from other EU nations to fuel the transition into a fully ASTRIDs fleet.

Nuclear Fuel Material Inventory

Table 7.6 lists EU material inventory in 2050. The materials continue to accumulate after 2050, but the UNF France receives before 2050 is most impactful for the feasibility of the transition. Note that table 7.6 distinguishes the UOX in the simulation either stored or reprocessed to create MOX.

Table 7.6: EU nuclear material inventory in 2050.

Category	**Value [MTHM]**	Specifics
UOX Loaded	161,894	UOX used in EU (minus France) reactors 1970-2050
MOX Loaded	6,945	MOX used in French reactors 1970-2050
Available used UOX (EU)	95,193	Used EU (minus France) UOX in storage for future ASTRID MOX production
Available used UOX (France)	10,029	Used French UOX stored for future ASTRID MOX production.
Reprocessed UOX (France)	53,590	Used French UOX already reprocessed for the production of LWR MOX
Tails	980,294	(Tails generated) − (Tails used for production of LWR MOX)
Natural U Used	1,142,189	

Figures 7.6 and 7.8 show the accumulation of tails and used fuel over time in the EU. Tails accumulate as a by-product of uranium enrichment. For every ton of UOX fuel, about nine times of tails is produced. Spent fuel is discharged from reactors every refueling period. The entire core is discharged when the reactor decommissions. A total of about $1,000,000$ MTHM of tails and $100,000$ MTHM of UNF have accumulated by 2050. Figure 7.7 shows the amount of fuel used in the EU. The tails mass accumulation rate is fairly steady, with peaks occurring when new reactors are deployed. In fig. 7.8, the peaks are caused by reactor decommissioning which triggers all the batches in the final reactor core to be sent to the repository.

French SFR Deployment

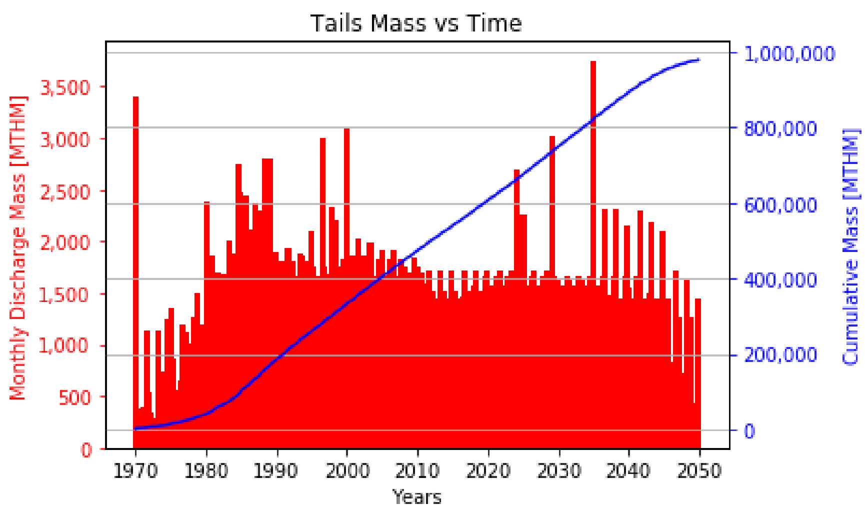

Figure 7.6: Simulated accumulation of tails in the EU is shown as a function of time.

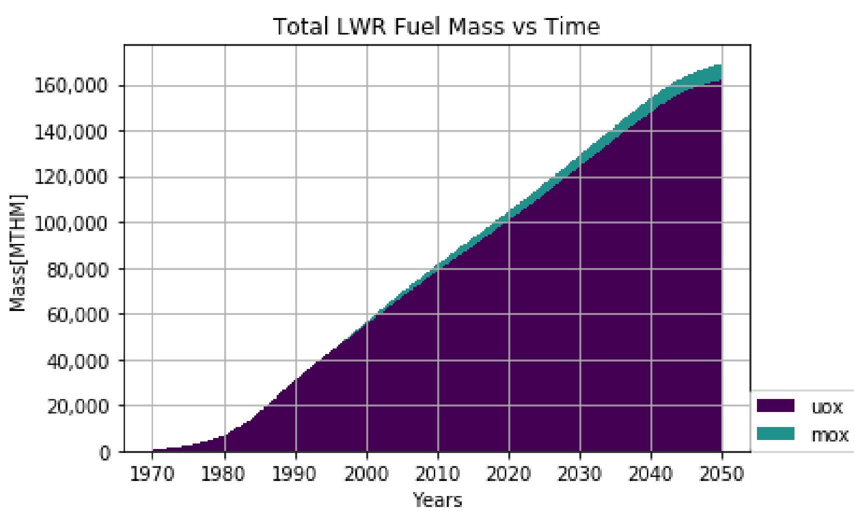

Figure 7.7: Simulated total EU fuel useage is shown as a function of time.

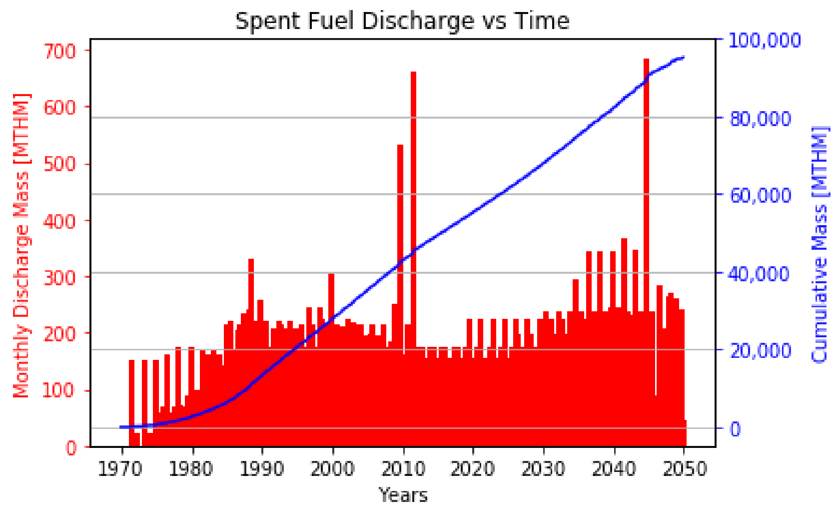

Figure 7.8: Simulated EU UNF accumulation and discharge is shown as a function of time.

Reprocessing the UNF collected from all EU nations can provide the initial cores for approximately 180 SFRs. Table 7.7 lists the isotope, mass fraction, and quantity of plutonium that can be obtained from the 2050 UNF inventory. With the SFR breeding ratio above one, France can transition into a fully SFR fleet without extra construction of LWRs.

Table 7.7: Plutonium in the UNF inventory.

Isotope	**Mass Fraction in Used Fuel [%]**	**Quantity [t]**
Pu238	0.0111	10.52
Pu239	0.518	545.05
Pu240	0.232	244.11
Pu241	0.126	132.58
Pu242	0.0487	51.24
Total	**0.9358**	**983.52**

From Varaine et al. [63], a French ASTRID-type 600MWe SFR consumes 1.225 metric tons of plutonium a year, with an initial plutonium loading of 4.9 metric tons. Thus, the number of SFRs that can be loaded with the reprocessed plutonium from UNF can be estimated to be 200, assuming adequate reprocessing and fabrication capacity as well as abundant depleted uranium supply.

Used MOX from an ASTRID reactor is 23.95% plutonium in this simulation (see table 1), whereas fresh MOX is 22% plutonium. The plutonium breeding ratio in this simulation is thus assumed to be ≈ 1.08.

Figure 7.9 shows MOX loaded in the SFRs per month. The plot has peaks during a period of aggressive deployment of SFRs followed by an equilibrium at 100 metric ton of heavy metal (MTHM). The peaks reoccur with the deployment of the second generation of SFRs. The spikes are due to initial fuel demand correspoding to these new deployments. The initial cores loaded into new SFRs rely on the MOX created from legacy UNF. Once the deployed SFRs create enough extra plutonium, the legacy UNF is no longer used. Notably, this switch from a less preferred fuel origin to a more preferred fuel origin is handled automatically within CYCLUS via user-defined preferences within its dynamic resource exchange algorithm [66].

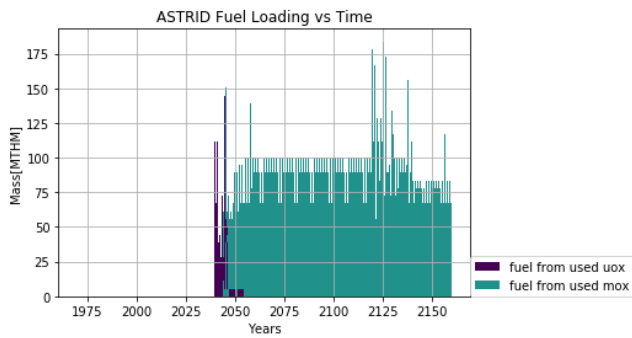

Figure 7.9: Fuel loaded into SFRs was simulated in discrete batches.

Figure 7.10 shows the separated plutonium discharge per month from the reprocessing plant. The plutonium outflux does not precisely follow the fuel demand because CYCLUS agents have material buffers that store commodity fuel for later usage. The reprocessed plutonium from legacy UNF is stored for the initial loading of SFRs. Plutonium separated from legacy UNF meets plutonium demans sufficiently to reduce the reprocessing demand for the first aggressive deployment of SFRs. The plutonium from reprocessing legacy fuel is a flat rectangle because the reprocessing throughput was set to 183.2 $\frac{MTHM}{month}$ to avoid reprocessing all the legacy in one timestep.

Table 7.8 lists metrics obtained from the second simulation.

These results demonstrate that despite the large amount of initial plutonium that has to be reprocessed prior to ASTRID deployment, the 20 years (2020-2040) of ASTRID fuel preparation allows a reasonable level of average UOX reprocessing capacity demand. UOX reprocessing continues until 2057, when the ASTRID spent fuel can supply the plutonium for its own fuel.

Conclusion

France can transition into a fully SFR fleet with installed capacity of 66,000 MWe without building additional LWRs if France receives UNF from other EU nations. Supporting the SFR fleet requires an average reprocessing capacity of 73.27 MTHM per month, and an average fabrication capacity of 45.29 MTHM per month.

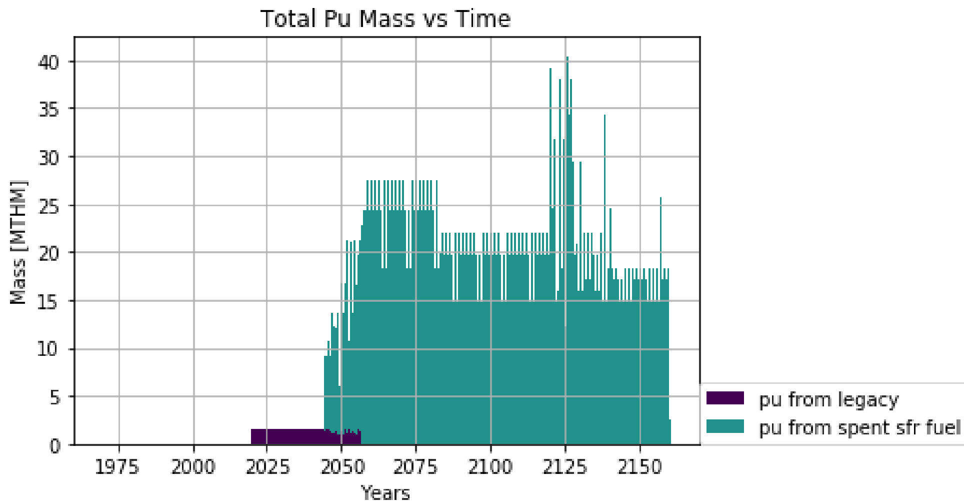

Figure 7.10: The separated plutonium discharge from the reprocessing plant in $\frac{\text{MTHM}}{\text{month}}$.

Since most EU nations do not have an operating UNF repository or a management plan, they have a strong incentive to send their UNF to France. In particular, the nations planning aggressive nuclear reduction will be able phase out nuclear without constructing a permanent repository. France has an incentive to take this fuel, since recycling used fuel from other nations will allow France to meet their MOX demand without new construction of LWRs.

Table 7.9 lists EU nations and their UNF inventory in 2050. We analyzed a strategy in which the nations reducing their nuclear fleet send their UNF to France. The sum of UNF from Italy, Slovenia, Belgium, Spain and Germany provides enough UNF for the simulated transition ($\approx 54,000$ MTHM). These nations are shown in bold in table 7.9. Sweden is not considered because of its concrete waste management plan.

On the other hand, in these simulations, some complex political and economic factors were not incorporated and various assumptions were present in this scenario. For example, Germany's current policy is to not reprocess its LWR fuel [67], and this policy would create a shortage in the supply of LWR UNF for ASTRID MOX production. Continuation of that German policy would not, however, be incompatible with a change in EU policy that frees EU countries from creating their high level waste repositories, since France could still agree to take in Germany's UNF for direct disposal. The analysis method described herein could readily be adapted to account for such possibilities. The collaborative option explored here may hold value for the EU nuclear community, and may enable France to advance more rapidly into a closed fuel cycle.

Table 7.8: In the French transition to SFRs, the total legacy UNF reprocessed is the amount of UNF France needs for a transition into a fully SFR fleet.

Category	**Unit**	**Value**
Total ASTRID MOX used	MTHM	63,447
Average UOX Reprocessing	MTHM/month	**123.27**
Average Total Reprocessing	MTHM/month	**63.23**
Average Fuel Fabrication	MTHM/month	**74.31**
Total SFRs Deployed		220
Total Plutonium Reprocessed	MTHM	14,831
Total ASTRID fuel from UOX Waste	MTHM	2,895
Total ASTRID fuel from MOX Waste	MTHM	60,552
Total Tails used	MTHM	49,488
Total legacy UNF reprocessed	MTHM	**53,595**
Total Reprocessed Uranium Stockpile	MTHM	159,383
Total Raffinate	MTHM	24,789

Table 7.9: EU nations and their respective UNF inventory.

Nation	**Growth Trajectory**	**UNF in 2050 [MTHM]**
Poland	Aggressive Growth	1,807
Hungary	Aggressive Growth	3,119
UK	Aggressive Growth	13,268
Slovakia	Modest Growth	2,746
Bulgaria	Modest Growth	3,237
Czech Rep.	Modest Growth	4,413
Finland	Modest Growth	5,713
Netherlands	Modest Reduction	539
Italy	**Modest Reduction**	**583**
Slovenia	**Modest Reduction**	**765**
Lithuania	**Modest Reduction**	**2,644**
Belgium	**Aggressive Reduction**	**6,644**
Spain	**Modest Reduction**	**9,771**
France	**Modest Reduction**	**9,979**
Sweden	Aggressive Reduction	16,035
Germany	**Aggressive Reduction**	**23,868**

7.1.3 Results - Parameter Sweep

7.1.4 Results - Sensitivity Study

Chapter 8

United States

The United States have been the forerunner of nuclear energy, with a current installed capacity of about 100 GWe. With its size and long history of nuclear energy, the United States have accumulated about $70,000$ MTHM of UNF.

The problem with modeling the U.S. transition scenario is that the U.S. does not have a defined advanced reactor, whereas France has a central plan to transition into ASTRIDs [68, 63]. Although the most prominent and 'canonical' reactor design when considering transition into fast-spectrum, breeding reactors]is the SFR, the fact that the U.S. nuclear reactor fleet is decided by economic interests (industries), this allows me to explore different options, such as the MSR design.

MSRs reactor designs have recently gained attention in the U.S. due to it being a potentially safer, more efficient, and sustainable form of nuclear power [38]. Multiple companies in the U.S. are now pursuing commercialization of MSR design reactors, such as Tranasatomic [39] , Terrapower, Terrestrial [40], and Thorcon [41]. Other parties such as China (TMSR-LF [42]) and the European Union (MSFR [43], MOSART [44]) are developing MSR designs.

In this chapter, I explore the U.S. transition scenario from a LWR fleet into a MSR fleet.

8.1 Initial Conditions and Scenario Parameters

Unlike the French scenario, where the UNF inventory at the present time is unknown, there is a detailed database that describes the U.S. UNF inventory up to 2013 May. The Used Nuclear Fuel Storage Transportation and Disposal Analysis Resource and Data System (UNF-ST&DARDS) database is a comprehensive, controlled source of UNF information, including dry cask attributes, assembly data, and economic attributes [69]. This database allows the transition scenario simulation to start from 2013, instead of 1970, like the French simulation. The UNF inventory mass and composition in 2013 will be imported from UNF-ST&DARDS and will be 'initiated' in the simulation as a `Source` facility.

8.2 U.S. Deployment Schedule

8.2.1 Energy Demand Prediction

8.3 Scenario Specification

8.4 Reactor Specifications

8.5 Material Definitions

8.6 Results

8.7 Conclusion

Chapter 9

Appendix

.1 Fresh and Used Fuel Composition

Isotope	Used ASTRID Fuel	Used UOX Fuel	Used MOX Fuel
He4	8.2631E-05	9.4745E-07	2.5108E-05
Ra226	2.306EE-13	9.7885E-14	6.8586E-14
Ra228	6.029EE-21	2.7508E-20	1.0769E-19
Pb206	5.2269E-18	5.5747E-18	3.6378E-18
Pb207	1.0722E-15	1.6859E-15	1.0589E-15
Pb208	4.4347E-10	3.6888E-12	2.0018E-12
Pb210	1.3841E-16	3.0238E-19	1.1829E-19
Th228	7.7910E-10	8.4756E-12	4.9017E-12
Th229	3.5259E-11	2.7278E-12	1.4379E-12
Th230	1.1419E-08	2.6258E-09	2.3998E-09
Th232	6.3415E-11	4.1748E-10	8.7655E-10
Bi209	2.5042E-13	6.6077E-16	2.6878E-16
Ac227	2.8317E-14	3.0968E-14	2.4608E-14
Pa231	8.8076E-10	9.2465E-10	7.0696E-10
U232	1.4693E-07	0.0000	5.9336E-10
U233	4.0461E-08	2.2139E-09	1.0359E-08
U234	0.0010	0.0001	0.0002
U235	0.0003	0.0076	0.0043
U236	0.0005	0.0057	0.0051
U238	0.5864	0.9208	0.8283
Np237	0.0038	0.0006	0.0043
Pu238	0.0096	0.0002	0.0060
Pu239	0.0981	0.0060	0.0410
Pu240	0.0890	0.0029	0.0283
Pu241	0.0155	0.0017	0.0146
Pu242	0.0273	0.0008	0.0098
Pu244	1.779EE-07	2.8648E-08	2.1888E-07
Am241	0.0077	6.4427E-05	0.0021
Am242m	0.0005	8.5336E-07	5.0357E-05
Am243	0.0091	0.0001	0.0020
Cm242	0.0004	2.5898E-05	0.0002
Cm243	0.0000	0.0000	1.2639E-05
Cm244	0.0067	8.5616E-05	0.0010
Cm245	0.0017	5.7217E-06	0.0001
Cm246	0.0009	7.2956E-07	6.1406E-06
Cm247	0.0000	0.0000	1.2059E-07
Cm248	4.0265E-06	7.6916E-10	9.1585E-09
Cm250	1.076EE-12	4.2808E-18	3.7338E-17
Cf249	1.6590E-07	1.6499E-12	4.0567E-11
Cf250	9.5219E-09	2.0419E-12	2.9328E-11
Cf251	3.2032E-10	9.8655E-13	1.4479E-11
Cf 252	8.3754E-12	6.5797E-13	7.5346E-12
H3	3.1829E-07	8.5846E-08	1.0269E-07
Kr81	1.5156E-11	4.2168E-11	7.3446E-11
Kr85	0.0000	3.4448E-05	2.0548E-05
Sr90	0.0009	0.0007	0.0004
Tc99	0.0029	0.0011	0.0011
I129	0.0009	0.0002	0.0003
Cs134	0.0001	0.0002	0.0002
Cs135	0.0051	0.0006	0.0009

Table 1: Spent Fuel Compositions

Bibliography

[1] John Conti, Paul Holtberg, Jim Diefenderfer, Angelina LaRose, James T. Turnure, and Lynn Westfall. International Energy Outlook 2016 With Projections to 2040. Technical Report DOE/EIA-0484(2016), USDOE Energy Information Administration (EIA), Washington, DC (United States). Office of Energy Analysis, May 2016. URL https://www.osti.gov/biblio/1296780.

[2] Matthew J. Gidden. *An Agent-Based Modeling Framework and Application for the Generic Nuclear Fuel Cycle*. PhD Thesis, University of Wisconsin, Madison, WI, United States, March 2015.

[3] R Wigeland, T Taiwo, H Ludewig, M Todosow, W Halsey, J Gehin, R Jubin, J Buelt, S Stockinger, K Jenni, and B Oakley. Nuclear Fuel Cycle Evaluation and Screening - Final Report. *US Department of Energy*, 2014. URL https://fuelcycleevaluation.inl.gov/Shared%20Documents/ES%20Main%20Report.pdf.

[4] Nicholas Tsoulfanidis. *The Nuclear Fuel Cycle*. American Nuclear Society, La Grange Park, Illinois, USA, 2013. ISBN 978-0-89448-460-5. 00177.

[5] D. F. Torgerson, Basma A. Shalaby, and Simon Pang. CANDU technology for Generation III+ and IV reactors. *Nuclear engineering and design*, 236(14-16): 1565–1572, 2006.

[6] Harold Feiveson, Zia Mian, M. V. Ramana, and Frank von Hippel. Spent Fuel from Nuclear Power Reactors. An Overview of a New Study by the International Panel on Fissile Materials. *International Atomic Energy Agency*, 2011.

[7] Roald A. Wigeland, Theodore H. Bauer, Thomas H. Fanning, and Edgar E. Morris. Separations and Transmutation Criteria to Improve Utilization of a Geologic Repository. *Nuclear Technology*, 154(1):95–106, April 2006. ISSN 0029-5450. doi: 10.13182/NT06-3. URL https://doi.org/10.13182/NT06-3.

[8] F. Baumgaertner. The Purex process for the reprocessing of nuclear fuels with high Pu content and high burn-up. *Kerntechnik*, 18(6):245–252, 1976.

[9] James J. Laidler, J. E. Battles, W. E. Miller, J. P. Ackerman, and E. L. Carls. Development of pyroprocessing technology. *Progress in Nuclear Energy*, 31(1-2): 131–140, 1997.

[10] Mycle Schneider and Yves Marignac. *Spent nuclear fuel reprocessing in France*. 2008. URL http://www.psr.org/nuclear-bailout/resources/spent-nuclear-fuel.pdf.

[11] A. D. Riley, P. Parkes, and N. H. A. Donaldson. Technology development for future Thorp reprocessing. In *Nuclear recycling. RECOD 98. 5. international nuclear conference on recycling, conditioning and disposal*, 1998.

[12] J. Eddie Birkett, Michael J. Carrott, O. Danny Fox, Chris J. Jones, Chris J. Maher, CÃl'cile V. Roube, Robin J. Taylor, and David A. Woodhead. Recent developments in the Purex process for nuclear fuel reprocessing: Complexant based stripping for uranium/plutonium separation. *CHIMIA International Journal for Chemistry*, 59(12):898–904, 2005.

[13] B. Feng, B. Dixon, E. Sunny, A. Cuadra, J. Jacobson, N. R. Brown, J. Powers, A. Worrall, S. Passerini, and R. Gregg. Standardized verification of fuel cycle modeling. *Annals of Nuclear Energy*, 94:300–312, August 2016. ISSN 0306-4549. doi: 10.1016/j.anucene.2016.03.002. URL http://www.sciencedirect.com/science/article/pii/S0306454916301098.

[14] A. M. Yacout, J. J. Jacobson, G. E. Matthern, S. J. Piet, and A. Moisseytsev. Modeling the Nuclear Fuel Cycle. In *The 23rd International Conference of the System Dynamics Society," Boston*. Citeseer, 2005. URL http://www.inl.gov/technicalpublications/Documents/3169906.pdf.

[15] J. Jacobson, A. Yacout, G. Matthern, S. Piet, D. Shropshire, R. Jeffers, and T Schweitzer. Verifiable fuel cycle simulation model (VISION): a tool for analyzing nuclear fuel cycle futures. *Nuclear Technology*, 172(2):157–178, November 2010.

[16] Robert Gregg and C. Grove. Analysis of the UK Nuclear Fission Roadmap using the ORION fuel cycle modelling code. In *Proc of the IChemE nuclear fuel cycle conference, Manchester, United Kingdom*, 2012.

[17] Carol Shay, Joseph DeCarolis, Dan Loughlin, Cynthia Gage, Sonia Yeh, Samudra Vijay, and E. L. Wright. EPA US national MARKAL database: database documentation. *US Environmental Protection Agency, Office of Research and Development*, 2006.

[18] IAEA. Nuclear Power Reactors in the World. 2018. URL http://www-pub.iaea.org/books/IAEABooks/13379/Nuclear-Power-Reactors-in-the-World.

[19] Jaakko Leppanen. SerpentâĂŞa continuous-energy Monte Carlo reactor physics burnup calculation code. *VTT Technical Research Centre of Finland*, 4, 2013.

[20] Kathryn D. Huff, Matthew J. Gidden, Robert W. Carlsen, Robert R. Flanagan, Meghan B. McGarry, Arrielle C. Opotowsky, Erich A. Schneider, Anthony M. Scopatz, and Paul P. H. Wilson. Fundamental concepts in the Cyclus nuclear fuel

cycle simulation framework. *Advances in Engineering Software*, 94:46–59, April 2016. ISSN 0965-9978. doi: 10.1016/j.advengsoft.2016.01.014. URL http://www.sciencedirect.com/science/article/pii/S0965997816300229. arXiv: 1509.03604.

[21] Kathryn Huff and Brent Dixon. Next Generation Fuel Cycle Simulator Functions and Requirements Document. Technical Report fcrd-sysa-2010-000110, Idaho National Laboratory, July 2010.

[22] Laurent Guerin, Bo Feng, Pavel Hejzlar, Benoit Forget, Mujid S. Kazimi, Luc Van Den Durpel, Abdellatif Yacout, Temi Taiwo, Brent W. Dixon, Grechen Matthern, Lionel Boucher, Marc Delpech, Richard Girieud, and Maryan Meyer. A Benchmark Study of Computer Codes for System Analysis of the Nuclear Fuel Cycle. Technical Report, Massachusetts Institute of Technology. Center for Advanced Nuclear Energy Systems. Nuclear Fuel Cycle Program, April 2009. URL http://dspace.mit.edu/handle/1721.1/75245. Electric Power Research Institute.

[23] Maryan Meyer and Lionel Boucher. New developments on COSI6, the simulation software for fuel cycle analysis. 2009.

[24] L. Van Den Durpel, D. C. Wade, and Abdellatif Yacout. DANESS: a system dynamics code for the holistic assessment of nuclear energy system strategies. *Proceedings of the 2006 System Dynamics Conference*, 2006. URL http://www.systemdynamics.org/conferences/2006/proceed/papers/VAND499.pdf.

[25] V. Tsibulskiy, S. Subbotin, M. Khoroshev, and F. Depisch. DESAE (Dynamic Energy System-Atomic Energy): Integrated Computer Model for Perfoming Global Analysis in INPRO Assessment Studies. In *14th International Conference on Nuclear Engineering*, pages 749–753. American Society of Mechanical Engineers, 2006.

[26] F. ÃĄlvarez Velarde, E. M. GonzÃąlez-Romero, and I. Merino RodrÃŋguez. Validation of the burn-up code EVOLCODE 2.0 with PWR experimental data and with a Sensitivity/Uncertainty analysis. *Annals of Nuclear Energy*, 73:175–188, 2014.

[27] OECD. *Nuclear Fuel Cycle Transition Scenario Studies*. Nuclear Science. OECD Publishing, February 2009. ISBN 978-92-64-10708-3. doi: 10.1787/9789264107083-en. URL http://www.oecd-ilibrary.org/nuclear-energy/nuclear-fuel-cycle-transition-scenario-studies_9789264107083-en.

[28] Mary Lou Dunzik-Gougar, Christopher A. Juchau, Kemal Pasamehmetoglu, Paul P. H. Wilson, Kyle M. Oliver, Paul J. Turinsky, Hany S. Abdel-Khalik, Ross Hays, and Tracy E. Stover. Global Evaluation of Nuclear Infrastructure Utilization Scenarios (GENIUS). In *GLOBAL 2007: Advanced Nuclear Fuel Cycles and Systems,*

September 9, 2007 - September 13, 2007, GLOBAL 2007: Advanced Nuclear Fuel Cycles and Systems, pages 1604–1611, Boise, ID, United states, 2007. American Nuclear Society.

[29] Kerry Dunn, Tae Wook Ahn, Royal Elmore, Kathryn Huff, Kyle Oliver, and Paul P. H. Wilson. GENIUS v2: An Extensible Platform for Modeling Advanced Global Fuel Cycles, November 2009. URL http://digital.library.wisc.edu/1793/38073.

[30] IAEA. Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems INPRO Manual - Overview of the Methodology. November 2008. URL http://www-pub.iaea.org/books/IAEABooks/7787/Guidance-for-the-Application-of-an-Assessment-Methodology-for-Innovative-Nuclear-Energy

[31] Erich A. Schneider, Charles G. Bathke, and Michael R. James. NFCSim: A Dynamic Fuel Burnup and Fuel Cycle Simulation Tool. *Nuclear Technology*, 151(1): 35–50, July 2005.

[32] Nicholas R. Brown, Brett W. Carlsen, Brent W. Dixon, Bo Feng, Harris R. Greenberg, Ross D. Hays, Stefano Passerini, Michael Todosow, and Andrew Worrall. Identification of fuel cycle simulator functionalities for analysis of transition to a new fuel cycle. *Annals of Nuclear Energy*, 96:88–95, October 2016. ISSN 0306-4549. doi: 10.1016/j.anucene.2016.05.027. URL http://www.sciencedirect.com/science/article/pii/S0306454916303383.

[33] Lionel Boucher, F. Alvarez Velarde, E. Gonzalez, B. W. Dixon, G. Edwards, G. Dick, and K. Ono. International comparison for transition scenario codes involving COSI, DESAE, EVOLCODE, FAMILY and VISION. *Actinide and Fission Product Partitioning and Transmutation*, page 61, 2010.

[34] L. Boucher. Benchmark Study on Nuclear Fuel Cycle Transition Scenarios Analysis Codes. Technical Report NEA/NSC/WPFC/DOC(2012)16, OECD, Nuclear Energy Agency, June 2012.

[35] Kathryn A. McCarthy, B. Dixon, Yong-Joon Choi, L. Boucher, Kiyoshi Ono, Francisco Alvarez-Velarde, Enrique Miguel Gonzalez, and B. Hyland. Benchmark Study on Nuclear Fuel Cycle Transition Scenarios Analysis Codes. Technical report, Tech. Rep. NEA/NSC/WPFC/DOC, 2012. URL https://inis.iaea.org/search/search.aspx?orig_q=RN:44089401.

[36] Eva E. Sunny, Andrew Worrall, Joshua L. Peterson, Jeffrey J. Powers, Jess C. Gehin, and Robert Gregg. Transition Analysis of Promising US Future Fuel Cycles Using ORION. Technical report, Oak Ridge National Laboratory (ORNL), Oak Ridge, TN (United States), 2015. URL http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/47/088/47088986.pdf.

[37] Frank CarrÃl’ and Jean-Marie Delbecq. Overview on the French nuclear fuel cycle strategy and transition scenario studies. In *Proceedings of GLOBAL*, Paris, France, 2009. URL https://www.researchgate.net/profile/Frank_

Carre/publication/273751217_Overview_on_the_French_Nuclear_Fuel_Cycle_Strategy_and_Transition_Scenario_Studies/links/55f6ace108ae07629dbae8ea.pdf.

[38] Jerome Serp, Michel Allibert, Ondrej Benes, Sylvie Delpech, Olga Feynberg, VÃl'ronique Ghetta, Daniel Heuer, David Holcomb, Victor Ignatiev, Jan Leen Kloosterman, Lelio Luzzi, Elsa Merle-Lucotte, Jan UhlÃŋÅŹ, Ritsuo Yoshioka, and Dai Zhimin. The molten salt reactor (MSR) in generation IV: Overview and perspectives. *Progress in Nuclear Energy*, 77(Supplement C):308–319, November 2014. ISSN 0149-1970. doi: 10.1016/j.pnucene.2014.02.014. URL http://www.sciencedirect.com/science/article/pii/S0149197014000456.

[39] Transatomic Power Corporation. Technical White Paper. White Paper 2.1, Transatomic Power Corporation, Cambridge, MA, United States, November 2016. URL http://www.transatomicpower.com/wp-content/uploads/2015/04/TAP-White-Paper-v2.1.pdf.

[40] David LeBlanc and Cyril Rodenburg. 18 - Integral molten salt reactor. In Thomas J. Dolan, editor, *Molten Salt Reactors and Thorium Energy*, pages 541–556. Woodhead Publishing, 2017. ISBN 978-0-08-101126-3. doi: 10.1016/B978-0-08-101126-3.00018-X. URL https://www.sciencedirect.com/science/article/pii/B978008101126300018X.

[41] Lars Jorgensen. 19 - ThorCon reactor. In Thomas J. Dolan, editor, *Molten Salt Reactors and Thorium Energy*, pages 557–564. Woodhead Publishing, 2017. ISBN 978-0-08-101126-3. doi: 10.1016/B978-0-08-101126-3.00019-1. URL https://www.sciencedirect.com/science/article/pii/B9780081011263000191.

[42] Zhimin Dai. 17 - Thorium molten salt reactor nuclear energy system (TMSR). In Thomas J. Dolan, editor, *Molten Salt Reactors and Thorium Energy*, pages 531–540. Woodhead Publishing, 2017. ISBN 978-0-08-101126-3. doi: 10.1016/B978-0-08-101126-3.00017-8. URL https://www.sciencedirect.com/science/article/pii/B9780081011263000178.

[43] D. Heuer, E. Merle-Lucotte, M. Allibert, M. Brovchenko, V. Ghetta, and P. Rubiolo. Towards the thorium fuel cycle with molten salt fast reactors. *Annals of Nuclear Energy*, 64:421–429, February 2014. ISSN 0306-4549. doi: 10.1016/j.anucene.2013.08.002. URL http://www.sciencedirect.com/science/article/pii/S0306454913004106.

[44] V. Ignatiev, O. Feynberg, I. Gnidoi, A. Merzlyakov, A. Surenkov, V. Uglov, A. Zagnitko, V. Subbotin, I. Sannikov, A. Toropov, V. Afonichkin, A. Bovet, V. Khokhlov, V. Shishkin, M. Kormilitsyn, A. Lizin, and A. Osipenko. Molten salt actinide recycler and transforming system without and with ThâĂŞU support: Fuel cycle flexibility and key material properties. *Annals of Nuclear Energy*, 64(Supplement C):408–420, February 2014. ISSN 0306-4549. doi: 10.1016/j.anucene.2013.09.004. URL http://www.sciencedirect.com/science/article/pii/S030645491300474X.

[45] J. J. Powers, T. J. Harrison, and J. C. Gehin. A new approach for modeling and analysis of molten salt reactors using SCALE. Sun Valley, ID, USA, July 2013. American Nuclear Society, 555 North Kensington Avenue, La Grange Park, IL 60526 (United States). URL `https://www.osti.gov/scitech/biblio/22212758`.

[46] Jeffrey J. Powers, Jess C. Gehin, Andrew Worrall, T. Jay Harrison, and Eva E. Sunny. An inventory analysis of thermal-spectrum thorium-fueled molten salt reactor concepts. In *PHYSOR 2014*, Kyoto, Japan, October 2014. JAEA-CONF–2014-003.

[47] Benjamin R. Betzler, Sean Robertson, Eva E. Davidson, Jeffrey J. Powers, Andrew Worrall, Leslie Dewan, and Mark Massie. Fuel cycle and neutronic performance of a spectral shift molten salt reactor design. *Annals of Nuclear Energy*, 119:396–410, September 2018. ISSN 0306-4549. doi: 10.1016/j.anucene.2018.04.043. URL `http://www.sciencedirect.com/science/article/pii/S0306454918302287`.

[48] Kathryn D. Huff, Massimiliano Fratoni, and Harris Greenberg. Extensions to the Cyclus Ecosystem In Support of Market-Driven Transition Capability. In *Transactions of the American Nuclear Society*, Anaheim, CA, United States, November 2014. American Nuclear Society.

[49] Jacob J. Jacobson, A. M. Yacout, Gretchen E. Matthern, Steven J. Piet, David E. Shropshire, Robert F. Jeffers, and Tyler Schweitzer. Verifiable fuel cycle simulation model (VISION): a tool for analyzing nuclear fuel cycle futures. *Nuclear Technology*, 172(2):157–178, 2010.

[50] Jaakko Leppanen. Serpent – a Continuous-energy Monte Carlo Reactor Physics Burnup Calculation Code. *VTT Technical Research Centre of Finland, Espoo, Finland*, 2015.

[51] Andrei Rykhlevskii, Alexander Lindsay, and Kathryn D. Huff. Online reprocessing simulation for thorium-fueled molten salt breeder reactor. In *Transactions of the American Nuclear Society*, Washington, DC, United States, November 2017. American Nuclear Society.

[52] The HDF Group. Hierarchical data format, version 5, 1997. URL `https://www.hdfgroup.org/solutions/hdf5/`.

[53] Benjamin R. Betzler, Jeffrey J. Powers, and Andrew Worrall. Molten salt reactor neutronics and fuel cycle modeling and simulation with SCALE. *Annals of Nuclear Energy*, 101(Supplement C):489–503, March 2017. ISSN 03064549. doi: 10.1016/j.anucene.2016.11.040. URL `http://linkinghub.elsevier.com/retrieve/pii/S0306454916309185`.

[54] Terry L. Schulz. Westinghouse AP1000 advanced passive plant. *Nuclear Engineering and Design*, 236(14-16):1547–1557, 2006.

[55] CNE2. Reports of the CNE2. Technical report, Commission Nationae D'Evaluation, June 2015. URL `https://www.cne2.fr/index.php/en/cne-2-2007-to-this-day`.

[56] PRIS IAEA. *Nuclear Power Reactors in the World*. Number 2 in Reference Data Series. IAEA, Vienna, Austria, 2017. URL http://www-pub.iaea.org/books/IAEABooks/12237/Nuclear-Power-Reactors-in-the-World.

[57] World Nuclear Association. Nuclear Power in the European Union - World Nuclear Association, February 2017. URL http://www.world-nuclear.org/information-library/country-profiles/others/european-union.aspx.

[58] Paul L. Joskow and John E. Parsons. The Future of Nuclear Power After Fukushima. Working Paper, MIT CEEPR, February 2012. URL http://dspace.mit.edu/handle/1721.1/70857.

[59] Michael T. Hatch. *Politics and Nuclear Power: Energy Policy in Western Europe*. University Press of Kentucky, January 2015. ISBN 978-0-8131-6307-9. Google-Books-ID: TrwfBgAAQBAJ.

[60] D. Hugelmann and D. Greneche. MELOX fuel fabrication plant: operational feedback and future prospects. In *MOX Fuel Cycle Technologies for Medium and Long Term Deployment (Proc. Symp. Vienna, 1999), C&S Papers Series No*, volume 3, pages 102–108, 1999. URL http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/31/062/31062323.pdf#page=110.

[61] Balendra Sutharshan, Meena Mutyala, Ronald P Vijuk, and Alok Mishra. The AP1000tm Reactor: Passive Safety and Modular Design. *Energy Procedia*, 7:293–302, January 2011. ISSN 1876-6102. doi: 10.1016/j.egypro.2011.06.038. URL http://www.sciencedirect.com/science/article/pii/S1876610211015475.

[62] David Hinds and Chris Maslak. Next-generation nuclear energy: The ESBWR. *Nuclear News*, 49(1):35–40, 2006.

[63] Frederic Varaine, Marie-Sophie Chenaud, Philippe Marsault, Bruno Bernardin, Alain Conti, Pierre Sciora, Christophe Venard, Bruno Fontaine, Laurent Martin, and Gerard Mignot. Pre-conceptual design study of ASTRID core. June 2012. URL https://www.researchgate.net/profile/Frederic_Varaine/publication/282657288_Pre-conceptual_design_study_of_ASTRID_core/links/56166d1908ae37cfe4090bb7.pdf.

[64] C. V. Parks. Overview of ORIGEN2 and ORIGEN-S: Capabilities and limitations. Technical Report CONF-920430-47, Oak Ridge National Lab., TN (United States), February 1992. URL https://www.osti.gov/biblio/10116143.

[65] P. Wilson. The Adoption of Advanced Fuel Cycle Technology Under a Single Repository Policy. Technical report, University of Wisconsin – Madison, 2009.

[66] Matthew J. Gidden and Paul P. H. Wilson. A methodology for determining the dynamic exchange of resources in nuclear fuel cycle simulation. *Nuclear Engineering and Design*, 310:378–394, December 2016. ISSN 0029-5493. doi:

10.1016/j.nucengdes.2016.10.029. URL https://www.sciencedirect.com/science/article/pii/S0029549316304101.

[67] K. TÃűpfer, M. Kleiner, U. Beck, U. Fisher, K. van Donhanyi, A. Gluck, J. Hacker, J. Hambrecht, V. Hauff, and W. Hirche. GermanyâĂŹs Energy TurnaroundâĂŞA Collective Effort for the Future. *Ethics Commission for a Safe Energy Supply. On behalf of Federal Chancellor Dr. Angela Merkel*, 2011.

[68] Bernard Boullis. The French Nuclear Fuel Cycle, May 2015. URL https://www.iaea.org/INPRO/10th_Dialogue_Forum/Day3/Session5/03.Boullis_France.pdf.

[69] Josh Peterson, Bret van den Akker, Riley Cumberland, Paul Miller, and Kaushik Banerjee. UNF-ST&DARDS Unified Database and the Automatic Document Generator. *Nuclear Technology*, 199(3):310–319, 2017.