

Evaluation of Chloride Fuel Salt Lifetime in a Fast-Spectrum, Liquid-Fuel Molten Salt Reactor

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ABSTRACT

Growing interest and investment in new nuclear reactor development span a wide range of designs that include light-water small modular reactors and non-water-cooled reactors. Of these advanced reactors, molten salt reactors (MSRs) represent the most diverse class with respect to fuel forms, neutron spectra, and primary coolant chemistries. Liquid-fuel MSR designs feature nuclear fuel dissolved in the primary system coolant, and this attribute represents a profound departure from current commercial reactor designs that incorporate nuclear fuel in fabricated solid forms. As with any fission reactor, the buildup of fission products—whether in solid or liquid fuel—increases parasitic neutron absorption over time. Maintaining reactor criticality and/or control generally requires intervention, either in the form of replacement of some fraction of solid fuel with fresh fuel (refueling) or removal of some fraction of neutron-absorbing fission products via physical and/or chemical processes. Carrier fuel salts are, to a large degree, radiation resistant; however, without some form of cleanup, the ingrowth of elements and isotopes will limit fuel salt use (or reuse) due to increasing parasitic neutron absorption, changing chemistry and corrosion potentials, and impacts on fuel solubility.

This study investigates how fission product concentrations and compositions evolve over time in the fuel salt of a chloride-based, fast-spectrum, liquid-fuel fast reactor—in essence, a chloride salt fast reactor (CSFR). The study further explores what these fuel salt changes imply for fuel salt sustainability and reuse as well as nuclear fuel cycle management. As a follow-up to preliminary spreadsheet-based calculations, the zero-dimensional reactor physics code, ORIGEN2, is used to calculate the buildup and depletion of individual radionuclides and, by summation, radioelements in a representative CSFR operating on a U/Pu fuel cycle. Results indicate the accumulation of significant fission product concentrations in CSFR fuel salt over the time frames of interest, with potential impacts on reactor performance, waste management, and downstream reuse of fuel salt to start up subsequent CSFR units. More granular analyses and direct results indicating CSFR fuel salt reactivity and criticality effects require, in turn, more detailed CSFR reactor design information and two-dimensional reactor physics calculations not available with the application of ORIGEN2.

Keywords

Advanced reactors Chloride salt fast reactor (CSFR) Fission products Fuel salt sustainability Liquid fuel Molten salt reactor (MSR)



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PRIMARY AUDIENCE: Liquid-fuel molten salt reactor (MSR) developers and vendors

SECONDARY AUDIENCE: Potential owner-operators and other stakeholders with an interest in the design and performance of liquid-fuel MSRs

KEY RESEARCH QUESTION

Liquid-fuel MSR developers and vendors face many design options and uncertainties in their pursuit of a commercially viable advanced reactor technology. Liquid-fuel MSRs represent a dramatic shift away from traditional nuclear reactor technologies, especially with the incorporation of dissolved nuclear fuel circulating throughout the primary system in place of fixed solid fuel forms. This study investigates how fission product concentrations and compositions evolve over time in the fuel salt of a chloride-based, fast-spectrum, liquid-fuel fast reactor—that is, a chloride salt fast reactor (CSFR). The study further explores what these fuel salt changes imply for fuel salt sustainability and reuse as well as nuclear fuel cycle management.

RESEARCH OVERVIEW

Preliminary results generated using spreadsheet-based calculations and equation-based relationships suggest non-trivial fission product concentrations in CSFR fuel salt over the time frames of interest, motivating a more sophisticated and quantitative approach. The zero-dimensional reactor physics code, ORIGEN2, is used to calculate the buildup and depletion of individual radionuclides and, by summation, radioelements in the fuel salt. Developing and executing appropriate ORIGEN2 models requires the development of a sufficiently representative CSFR conceptual design and identification of information for parameterizing the reactor in ORIGEN2.

Information and assumptions used to inform ORIGEN2 model development are derived from review of the open literature. A U-Pu-Na chloride salt composition is assumed for the initial fuel carrier salt. Fuel salt polishing comprises basic physical cleanup processes for removal of insoluble fission product elements, chiefly noble/semi-noble metals and noble gases. To counter increasing reactivity over time due to the conversion of fertile constituents to fissile material, a fraction of fuel salt inventory is periodically removed and replaced with unirradiated salt containing fertile depleted uranium. The resulting ORIGEN2 calculations describe the evolution of fission product concentrations and elemental compositions out to 100 full-power years. Sensitivity analyses—performed by varying key parameters—are used to explore the potential impact of fission product accumulation at equilibrium as well as reactor design attributes such as power density on the long-term performance of fuel carrier salts. Also investigated are the downstream implications following discharge for salt reuse in subsequent reactors or management as waste.



KEY FINDINGS

- The equilibrium fission product concentration in CSFR fuel salt is most sensitive to the fuel salt specific power and, therefore, reactor power density. Power density is mainly a design decision driven by structural material capabilities and economic tradeoffs but can be influenced by fuel salt density and other secondary parameter effects as well.
- The equilibrium fission product concentration in CSFR fuel salt increases from zero to approximately 17 wt% heavy metal for a power density of 300 kW/L. For core power densities of 100 kW/L and 500 kW/L, corresponding fuel salt fission product concentrations are 9 wt% and 26 wt%, respectively.
- For the 300 kW/L scenario at equilibrium, the fission products in CSFR fuel salt account for about 11% of neutron absorption in the fuel salt. For 100 kW/L and 500 kW/L scenarios, fission products account for approximately 3% and 18% of neutron absorption at equilibrium, respectively.
- Approximately 97% of fission product neutron absorption is attributable to Sm, Nd, Eu, Zr, Pm, Gd, Pr, Dy, and Ce. All except Zr are rare earth elements, are not removed by cleanup processing, and have relatively large neutron cross sections. Cleanup processes that remove noble gas precursors of rare earth isotopes can eliminate inventories of corresponding daughter products such as Ce, Zr, and Pr.
- The study does not define fission product concentration limits for sustained use and reuse of CSFR fuel carrier salts due to lack of available information needed to evaluate the evolution of fuel salt physical and chemical properties.
- The study does not directly evaluate the effects of the fission products on CSFR fuel salt reactivity or criticality. Such insights would require, at a minimum, two-dimensional reactor physics calculations not available using the ORIGEN2 application.
- Further analyses and refinement of the CSFR model to address feasibility issues related to reactor physics require a more detailed CSFR reactor design.

WHY THIS MATTERS

The degree to which fuel carrier salts in a liquid-fuel MSR can continue to be used in the reactor or reused to start up subsequent reactor units will ultimately affect plant economics, performance, design, and waste management burden. The greater the impact of fission products on fuel salt sustainability, the greater the complexity and scale of cleanup processes, bulk salt removal and replenishment, waste management, and other fuel cycle operations. An integrated understanding of fuel salt sustainability early in the liquid-fuel MSR design process may help developers and owner-operators avoid downstream design and operational challenges.



HOW TO APPLY RESULTS

This report provides advanced reactor developers, vendors, and other stakeholders with qualitative and quantitative analyses that identify potentially important design limitations and considerations with respect to liquid-fuel MSRs and the impacts of fission products on the useful service life of fuel-bearing salts, with and without application of fuel salt polishing and removal mechanisms. These results may serve as a useful starting point for more detailed, design-specific evaluations such as fuel salt reactivity, which require application of more sophisticated reactor physics codes, multidimensional analysis, and design-specific information.

LEARNING AND ENGAGEMENT OPPORTUNITIES

- EPRI maintains public- and member-facing advisory groups under the Advanced Nuclear Technology (ANT) Program that focus on advanced reactor R&D, demonstration, and commercialization topics. These forums provide opportunities for exchanging information and obtaining input on the direction and nature of EPRI's ANT programmatic focus to support deployment of advanced reactors.
- EPRI continues to seek and welcome collaborative opportunities for the development and application of tools and methods that support commercialization of advanced nuclear technology.

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ACRONYMS AND INITIALISMS

AMORUUUX	advanced mixed oxide reactor using fuel made of recycled plutonium and depleted uranium				
AR	advanced reactor				
CHLOROPHIL	undefined name for the molten chloride fast breeder reactor described in Taube and Ligou [1974]				
cm ³ cubic	c centimeter				
CSFR	chloride salt fast reactor				
DU	depleted uranium				
ENDF/B	Evaluated Nuclear Data File				
EPRI	Electric Power Research Institute				
FHR	fluoride salt-cooled high-temperature reactor				
FLiBe	LiF-BeF ₂ (salt)				
FPY	full-power year				
FSFR	fluoride salt fast reactor				
FSMSR	fast spectrum molten salt reactor				
g	gram				
GW	gigawatt				
GWe	gigawatt-electric				
GWt	gigawatt-thermal				
GW-d	gigawatt-day (unit of energy generation)				
HLW	high-level radioactive waste				
HM	heavy metal				
HTGR	high-temperature gas-cooled reactor				
IS	intermediate spectrum				
kg	kilogram				
kW	kilowatt				

L	liter
LWR	light-water reactor
MCSFR	molten chloride salt fast reactor
mol	mole
mol%	mole percent
MSBR	molten salt thermal breeder reactor fueled thorium-uranium fluoride salt
MSR	molten salt reactor
MTHM	metric ton heavy metal
MW	megawatt
MWe	megawatt-electric
MWt	megawatt-thermal
MWd	megawatt-day (unit of energy generation)
NEA	Nuclear Energy Agency
NFCOC	Nuclear Fuel Cycle Options Catalog
OECD	Organization for Economic Cooperation and Development
ORIGEN2	Oak Ridge isotope generation and depletion code
ORNL	Oak Ridge National Laboratory
REBUS	undefined name for a fast reactor design using molten salt fuel consisting of uranium and transuranic trichlorides dissolved in sodium chloride described in [Mourogov and Bokov 2006]
R&D	research and development
SCALE	Standardized Computer Analyses Licensing Evaluation
S	second
SFR	sodium-cooled fast reactor
SMR	small modular reactor
TRU	transuranic
W	watt
WNA	World Nuclear Association
wt%	weight percent
У	year

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1 INTRODUCTION

Growing interest and investment in new nuclear reactor development span a wide range of designs that include light-water small modular reactors (SMRs) and non-water-cooled reactors. Of these advanced reactors (ARs), molten salt reactors (MSRs) represent the most diverse class with respect to fuel forms, neutron spectra, and primary coolant chemistries. Liquid-fuel MSR designs feature nuclear fuel dissolved in the primary system coolant, and this attribute represents a profound departure from current commercial reactor designs which universally incorporate nuclear fuel in fabricated solid forms. Liquid-fuel designs comprise the majority of MSRs under private sector development for commercialization in the 2030 timeframe.

While immature relative to high-temperature gas-cooled reactors (HTGRs) and sodium-cooled fast reactors (SFRs) that have operated at commercial scales and provided sustained electricity to grids, liquid-fuel MSR technology has been successfully tested and operated at experimental scales; two experimental reactors were constructed and at the Oak Ridge National Laboratory during the 1950s and 60s [Rosenthal et al., 1970; Greene, 2001].

As with any fission reactor, the buildup of fission products—whether in solid or liquid fuel increases parasitic neutron absorption over time. Maintaining reactor criticality and/or control generally requires intervention, either in the form of replacement of some fraction of solid fuel with fresh fuel (refueling) or removal of some fraction of neutron-absorbing fission products via physical and/or chemical processes.

As ionic species, molten fluoride and chloride carrier salts are inherently radiation resistant, aside from neutron absorption reactions that depend on isotopic enrichments of components. Therefore, fuel carrier salts in liquid-fuel MSRs can, in principle, have virtually unlimited lifetimes. However, the irradiation of nuclear fuel-bearing salts introduces other soluble and insoluble species, including actinides via conversion, fission products, and activation products. Therefore, while the carrier salt itself may not be degrade substantially over time, the ingrowth of elements and isotopes may present practical limits on fuel salt use without some form of fuel salt cleanup by increasing parasitic neutron absorption, changing overall primary system chemistry and corrosion potentials, and affecting solubility of the fuel species.

EPRI conducted a preliminary, simplified analysis, to explore trends and magnitudes in fission product ingrowth and removal with respect to sustained fuel salt use and reuse, including implications for contemporary liquid-fuel MSR designs [EPRI, 2021]. Results generated via spreadsheet-based calculations using basic equation-based relationships suggest accumulation of non-trivial fission product concentrations over timeframes of interest, motivating the more quantitative analysis of the study documented in this report. Specifically, the preliminary analysis indicate priority should be given to developing a more complete understanding of how fission product concentrations evolve during reactor operation and the potential efficacy of mitigation measures. Appendix A summarizes this preliminary analysis.

Introduction

This study investigates (1) how fission product concentrations and compositions evolve over time in the fuel salt of a chloride-based fast-spectrum liquid-fuel fast reactor—a chloride salt fast reactor (CSFR)—and (2) what these fuel salt changes imply for fuel salt sustainability, reuse, and nuclear fuel cycle management.

To do so, the study employed a more sophisticated analysis approach than a spreadsheet (but less sophisticated than multi-group reactor physics code systems such as SCALE¹) to understand the rate at which the equilibrium fission product concentration is approached and how the concentration of specific fission product nuclides and elements becomes skewed because of cleanup processes in the fuel salt to form a basis for assessing the impacts of the fission products on reactor design and operation, and out-of-reactor fuel cycle requirements.

The zero-dimensional (point depletion) reactor physics code ORIGEN2 is used to calculate the buildup and depletion of individual radionuclides and, by summation, radioelements in the fuel salt [Croff, 1980]. Developing and executing appropriate ORIGEN2 models, requires the development of a sufficiently representative CSFR conceptual design and identification of information for parameterizing the reactor in ORIGEN2.

Specific topics of inquiry include:

- What is the total fission products concentration in the fuel salt at equilibrium, how long does it take to reach equilibrium, and how does this result compare to the results in the preliminary analysis in the fuel salt? (Section 3)
- How does cleanup processing affect the elemental composition of the fuel salt? (Section 3)
- Could the calculated fission product concentrations constrain sustained reuse of discard fuel salt? (Section 4)

Section 2 documents the representative CSFR concept² and associated parameters used for ORIGEN2 calculations through literature review. Section 3 describes how the CSFR model was implemented, the results of implementing the model, and an evaluation of the results. Section 4 presents conclusions and recommendations.

¹ The SCALE modeling and simulation platform supports nuclear safety analysis and design analyses. SCALE development, maintenance, and testing are managed by Oak Ridge National Laboratory [Rearden and Jessee, 2018].

² **Note:** CSFR is a generic term used in this report to refer a single-fluid, chloride-based, fast-spectrum, liquid-fuel MSR design or concept; other terms can be found in the literature, including molten chloride fast reactor and molten chloride salt fast reactor.

2 REFERENCE MOLTEN CHLORIDE SALT FAST REACTOR (CSFR) MODEL

Analyzing MSR fuel salt sustainability using ORIGEN2 requires a sufficiently representative reactor model. Development of previous ORIGEN2 models for solid-fuel fast reactors [Croff et al., 1983] involved reactor physics analysis of a two-dimensional radial/vertical (R-Z) model of a SFR core based on the physical characteristics and compositional information from a conceptual design of the Prototype Large Breeder Reactor. The analysis involved creating a multi-group cross section library from ENDF/B-IV [Drake, 1970] that was collapsed to a few-group cross section library for use in the depletion calculations and eventually to a library of one-group, spectrum-averaged cross sections for use in ORIGEN2. Creating the libraries and performing the depletion involved using a series of coupled computer codes identified in Croff et al. [1983].

The main limitation of the approach used in the present evaluation is the absence of dimensional representation of the reactor. As a result, the reactor physics calculations performed by ORIGEN2 restricts the ability to provide direct answers on some aspects, such as the impacts of the fission products on fuel salt reactivity.

This section introduces the general attributes required for an adequately representative CSFR design model to be used for subsequent ORIGEN2 calculations, describes relevant information gleaned from existing models and conceptual designs, and describes the parameters and assumptions selected for use in ORIGEN2 calculations using the resulting CSFR model.

2.1 Desired Attributes of a Representative CSFR Model

The approach used in the follow-on evaluation relies on information found in the available literature to supply the assumptions and parameters required by ORIGEN2 without having to perform multi-dimensional reactor physics calculations. Ideally, the reactor(s) described in the resource documents would have a number of attributes relevant to a CSFR so as to provide a solid basis for the ORIGEN2 model. The desirable attributes are listed below along with a brief rationale for including them:

- Fast neutron spectrum and a chloride fuel salt: this combination provides a shorter doubling time than thermal reactors or fluoride salts to better supply the initial fuel salt for subsequent new reactors.
- Single salt system: multiple salt regions or loops (for example, fuel salt and fertile blanket salt) are essentially impossible to analyze with a zero-dimensional depletion codes such as ORIGEN2 unless it is underpinned by the results of multi-dimensional, multi-group reactor physics calculations. Further, while many of the early reactor designs in the literature (1960s to 1980s) have more than one loop, concerns arose about vessel fabrication complications and subsequent replacement of reactor internals—and issue coined the *plumbing problem* by

ORNL. These concerns coupled with advances made during development and design of the fuel cycle of the MSBR [Rosenthal et al., 1972; Robertson, 1971] led ORNL to conclude that a single salt system was preferable even though its neutron economy was not as favorable.

- U-Pu fertile-fissile material: same as first bullet above and avoids issues associated with establishing a new thorium fuel cycle [OECD, 2015].
- Chlorine and uranium isotopic enrichments: natural chlorine is about 75% ³⁵Cl, which (a) captures neutrons without benefit and (b) produces long-lived ³⁶Cl, which can be a significant contributor to repository risk [Croff and Krahn, 2015].
- Uranium isotopic enrichment: the ²³⁵U enrichment of the uranium is not particularly important because the reactor is being fed depleted uranium having a negligible ²³⁵U concentration (an exception may be the case where the fissile material in the initial CSFR fuel salt inventory comprises enriched uranium).
- Fuel salt plutonium concentration and composition, preferably at approximate equilibrium: avoids having to deplete the salt for many cycles to approach the equilibrium composition.
- Reactor physics parameters: one necessary reactor physics parameter is power density or equivalent. All fission energy is produced in the fuel salt in a single-salt design because it contains all the fissionable material. Another necessary parameter is a set of one-group cross sections applicable to a CSFR.
- Fraction of the fuel salt in the core: The power density is multiplied by this parameter to obtain the average power density in the entire fuel salt inventory.
- Fuel salt density: required to convert the power density commonly found in the MSR literature to specific power used as ORIGEN2 input.

2.2 Review of Relevant Fast Spectrum MSR Design Models

A literature search revealed relatively few CSFR models and these have a wide range of assumptions and parameter values related to the attributes. A consolidated starting point is [Betzler et al, 2019] which identifies and briefly summarizes the characteristics of a number of halide (chloride and fluoride) fuel salt fast reactors. The chloride salt fast reactor characteristics from the literature are compared against the desired attributes in Table 2-1.

2.3 Evaluation of Relevant Information in the Literature

Information relevant for parameterizing a representative CSFR model was evaluated for the five reactors summarized in Table 2-1. Key information for each of the attributes leading to the selection of parameter values and assumptions for the ORIGEN2 CSFR model is discussed.

Table 2-1
Comparison of CSFR core designs found in the literature against desired attributes

Attribute	Reactor					
	CHLOROPHIL ^a	Molten Chloride Salt Fast Reactor (MCSFR)	REBUS- 3700ª	Nuclear Fuel Cycle Options Catalog (NFCOC)	Fast Spectrum Molten Salt Reactor (FSMSR)	
Reference	Taube 1972, 1974; Taube & Ligou, 1972	Smith, 1974	Mourogov & Bokov, 2006	Price, 2013	Davidson et al., 2019	
Fast spectrum Chloride salt	Yes	Yes	Yes	Yes	Yes	
Single fluid salt?	No	Yes ^e	Yes	Yes? ^h	Yes	
U-Pu fuel-cycle?	Yes	Yes	Yes	Yes	Yes	
Power density (kW/L)	220	364	100	200	166	
Cross sections provided?	Yes	No	No	No	Yes	
Fraction of fuel salt in core	fuel salt is static	0.44	0.66	not specified	not specified	
Fuel salt composition ^b	U/Na chloride ^c 36/64 mol% Pu/U: 24 mol%	U/Na chloride 38/62 mol% Pu/U: 7.5 mol%	U/Na chlorides 35/65 mol% Pu/U: 55 mol% ^f	U/Na chlorides 48/52 mol% Pu/U: 1.6 mole%	U/Na chlorides 30/70 mol% Pu/U: 3 %	
CI Enrichment as ³⁷ CI	25% (natural)	100%	25% (natural)	75%	100%	
U Enrichment as ²³⁵ U	not specified, depleted U if inferred	not specified	depleted	0.07%	0.12%	
Core Salt Density (g/cm³)	2.34 ^d	3.6	3.6	3.5 ^g	2.115	

^aA definition for this initialism was not identified.

^bThe eutectic mixture of U/Na is 33/67 mol%.

^cThe core fuel salt does not contain uranium, but uranium chloride salt circulates through the core in tubes like a heat exchanger so the core salt and the circulating salt in the core region were assumed to be homogenized for the purpose of obtaining information for a CSFR model.

^dFuel salt only.

eSalt circulates from core to blanket and then back to core in a continuous loop.

^fThe fissile material is a mixture of Pu (>90%) and minor actinides for TRU actinide burning, yielding a low breeding ratio.

^g From Mausolff [2019].

^h The reactor design and performance parameters are not described.

2.3.1 Observations on Reactor Models, Assumptions, and Parameter Values Found in the Literature

The following observations are worth noting to inform interpretation of the information presented in Table 2-1.

- All five reactors addressed in Table 2-1 have a fast neutron spectrum; they all use chloride fuel salt, the uranium-plutonium fertile-fissile pair, and sodium chloride as the carrier salt–all are CSFRs.
- Most of the reactors in Table 2-1 feature a single primary salt system and do not have separate fuel and blanket salts. Exceptions to this observation are:
 - The CHLOROPHIL reactor has two separate salt streams: static Pu/NaCl fuel salt and U/Na chloride blanket salt that circulates through and around the fuel salt in tubes.
 - The design of the CSFR in the NFCOC was not given.
 - The FSMSR design is based on the CHLOROPHIL. However, the analysts homogenized the fuel and blanket salt to yield a FSMSR having a single salt.
- The reported reactor power densities appear plausible and generally fall in the low-tomedium range (100 to 300 kw/L) identified in a preliminary analysis based on a range of literature values for both fluoride and chloride salt reactors [EPRI, 2021]. An important uncertainty associated with reported power densities is that it is not always clear whether the values correspond strictly to the core region itself or represent an average of the core region plus the additional fuel salt inventory that resides in the primary loop but outside of the core at any given moment.
- Cross sections are provided for less than a dozen radionuclides and elements for the CHLOROPHIL reactor including a cross section for a single, lumped fission product. A comprehensive list of cross sections is available for the FSMSR. There are essentially no cross sections mentioned for the other three reactors.
 - Cross-sections for the CHLOROPHIL reactor design are scattered among multiple references cited in Table 2-1. Multi-group cross sections appear to have been processed into one-group cross sections by collapsing 15-group cross sections using a corresponding multi-group neutron spectrum. The means by which the spectrum was determined is not stated.
 - FSMSR fission product cross sections are available in digital form but as the product of a cross section and fission product yield. Extraction of one-group cross sections suitable for use in ORIGEN2 would require determination of the fission product yields used and additional manipulations, which are beyond the scope of the study.
- In the liquid-fuel MSR designs identified in the literature, a fraction of the fuel salt continuously lies outside of the core region during operation and is associated with circulation to and from the primary heat exchanger and cleanup processing to remove insoluble radionuclides. The fraction outside the core does not produce significant amounts of fission energy because it is in a sub-critical configuration and isolated from the core neutron flux by physical distance and shielding. This separation of fuel inventories lowers the average power density of the fuel salt. This fraction is determined by design decisions related

to the size and location of various pieces of equipment that will be fabricated and assembled in the future. There are currently no known theoretical limits to the fraction of the salt outside of the core based on physical and/or chemical reasons. Information on the in-core fraction of fuel salt is not reported for the NFCOC and FSMSR designs.

- The fuel salt elemental composition is characterized by the U/Na ratio and the Pu/U ratio (discussed in the next item). The U/Na ratio is typically within a small range near the U-Na eutectic point of 33/67 mol%. Exceptions and noteworthy features are:
 - The NFCOC U/Na ratio is very high for reasons that are unknown because of the absence of information on the reactor design.
 - The FSMSR composition was obtained by homogenizing the fuel and blanket in the CHLOROPHIL design. However, instead of homogenizing just the fuel salt with the blanket salt within the core, the fuel salt was homogenized with all of the blanket salt including that surrounding the core and circulating in the coolant loop. The result is a low U/Na ratio which means that other parameters such as cross sections may not be representative of more typical single salt CSFRs to an unknown extent
- The Pu/U ratio spans a wide range (1.6 mol%–55 mol%), and there is not enough detail in the source documents to ascertain the cause of the differences (for example, design objectives, design assumptions, operating assumptions, code differences) in most cases. Notable exceptions and features include:
 - The value of 24 mol% for CHLOROPHIL is for the core of a two-fluid system which would be expected to have a high plutonium concentration to breed plutonium in the blanket so it should be discounted.
 - The value of 55% for REBUS-3700 is likely influenced by (a) the presence of <10% of minor actinides; (b) the objective of the reactor being to burn actinides; and (c) its small (just greater than 1.0) breeding ratio.
 - The 1.6% value for the NFCOC design is comparatively low and there is very little information on the design assumptions and how it was modeled.
 - The MCSFR value of 7.5% is plausible but it was calculated using what was available circa the mid-1970s such as (a) cross sections for key isotopes of uranium, plutonium, chlorine, and sodium that do not reflect improved data obtained since that time and (b) improved reactor physics codes for processing and applying the multi-group cross sections.
 - The FSMSR value of 10.3% is also plausible, but this is for homogenized CHLOROPHIL reactor fuel and blanket containing excess sodium in an unknown configuration. This may limit relevance to and representativeness of CSFRs.
- Two designs (CHLOROPHIL and REBUS-3700) assume chlorine having a natural isotopic abundance (75% ³⁵Cl). This is important because (a) ³⁵Cl has a significant neutron capture cross-section that consumes neutrons without benefit and (b) neutron captures in ³⁵Cl produce long-lived ³⁶Cl, which has been calculated to be an important contributor to repository risk [Croff and Krahn, 2015]. Exceptions and noteworthy aspects are:
 - The NFCOC design uses chlorine that is enriched to 75% ³⁷Cl.

- The MCSFR and FSMSR designs assumed chlorine enriched to 100% ³⁷Cl.
- An additional complication is that the ratio of the chlorine anion to the cations in the NFCOC composition (mainly sodium, uranium, and plutonium) indicates that the analysts assumed all cations were monovalent whereas both uranium and plutonium are trivalent. As a consequence, additional chlorine was introduced, and the composition was renormalized as part of the follow-on evaluation.
- The designs identified within the literature typically use depleted uranium, although the ²³⁵U concentration is not always given. Exceptions and noteworthy aspects are:
 - The 235 U concentration for the MCSFR is not stated.
 - The ²³⁵U concentration for the NFCOC is 0.07%, which is very low even for depleted uranium.
 - The FSMSR design uses depleted uranium containing 0.12% ²³⁵U, which is also a low value.
- Three of the fuel salt densities are in a narrow range (3.5-3.6 g/cm³) while the other two are significantly lower than this range and different from each other.
 - Differences in fuel salt temperature can explain some of the differences but the decrease in density is about 0.3-0.4 g/cm³ for the temperature range of 690°C to 1000°C represented by these reactors.
 - A larger contributor to the range is the variation in the uranium-to-sodium ratio because the former is much denser than the latter. For example, the fuel salt density for the FSMSR is the lowest of all the reactors and it also has the lowest U/Na ratio. This is attributable to the decision to blend all of the blanket salt with the core salt to yield a single-fluid system. The converse is true for the NFCOC CSFR which has a high U/Na ratio.
 - None of the documentation provides a source for the claimed densities. Comparison of the claimed densities with a correlation based on laboratory measurements [Desyatnik et al., 1975]–which account for both the salt temperature and U/Na ratio–shows that the MCSFR and REBUS salt densities are 0.4-0.6 g/cm³ too high compared to the lab measurements, and the FSMSR and CHLOROPHIL salt densities are too low by 0.6-0.7 g/cm³.

2.3.2 Evaluation of Parameters and Assumptions Relevant to an ORIGEN2 CSFR Model

The most desirable approach to establishing the ORIGEN2 CSFR reactor model parameter values and assumptions would be to simply adopt one of the reactors in Table 2-1. However, the discussion the previous section supports the view that usage of any single model from the literature is not suitable for this evaluation. This conclusion stems from the review of each identified reactor model candidate available in the literature summarized below.

2.3.2.1 CHLOROPHIL

- CHLOROPHIL is a two-fluid reactor making it difficult to analyze and it has the internal *plumbing problem* that, in part, motivated ORNL to abandon the two-fluid MSBR design in favor of the single-fluid design [Robertson, 1971].
- It has a limited number of one-group cross sections of uncertain provenance, for example, see Taube [1978]. The calculations for this reactor model are based on cross-sections circa 1974, which have been improved over the years, especially in the fast neutron region.
- The Pu/U ratio seems unreasonably high for a continuously fueled reactor.

2.3.2.2 MCSFR

- MCSFR is a single-fluid reactor but the same salt is both the core and blanket.
- It appears to exhibit the traditional MSR *plumbing problem*.
- It uses lead coolant-the effects of which on the reactor physics are unknown.
- Cross sections are not provided.

2.3.2.3 REBUS

- The Pu/U ratio of 55% for REBUS seems unreasonably high. The reactor was designed to be an actinide burner which may account for this but results in a small breeding gain.
- It involves chemical processing and separations to remove more fission products than what are removed by ordinary cleanup processing. How this processing is performed and modeled is not stated.
- Cross sections for this design are not available or identified.

2.3.2.4 NFCOC

- NFCOC initial fuel composition appears to be incorrect because too little chlorine is included in the fuel salt composition, which may impact results in unknown ways.
- The reactor design is not specified, and documentation is very limited.
- The Pu/U ratio seems unreasonably low and the U/Na ratio unreasonably high.
- There are no cross sections available or identified.

2.3.2.5 FSMSR

- FSMSR has a low U/Na ratio due to homogenization of the core and blanket.
- The FSMSR cross sections for the fission products would have to be extracted from a dataset that is in a format that is incompatible with ORIGEN2. This would involve writing software to extract the cross sections, reformat them, and finding a means to verify that it had been done properly.

Reference Molten Chloride Salt Fast Reactor (CSFR) Model

2.4 Reactor and Fuel Parameters for CSFR Model

The previous section concluded that none of the CSFR model reactors found in the literature were suitable as-is to provide the basis for an ORIGEN2 CSFR model. As a consequence, it was necessary to create a new CSFR model for use in this study. This task was facilitated by the combination of (a) the existence of the information gathered in evaluating the five reactor models in Table 2-1 and (b) having developed the simple spreadsheet-based fuel salt sustainability model in the preliminary analysis. In particular, the approach adopted here to develop the ORIGEN2 CSFR model is to incorporate information from both Table 2-1 and the preliminary analysis into this evaluation. The rest of this section does so, one parameter or assumption at a time.

2.4.1 Reactor Type

As is implied by the first three attributes in Table 2-1, it is assumed that the reactor to be modeled is fast-spectrum, fluid-fuel reactor using U-Pu-Na chloride salt. The main reason for this is the CSFR is the only MSR system identified that has the practical capability to breed enough fissile material to form the initial core of follow-on reactors in a timely manner.

2.4.2 Single vs. Multiple Fluid

The model reactor is a single fluid for simplicity—and therefore, ease of modeling—and because most liquid-fuel MSRs currently being developed are single fluid.

2.4.3 Power Density

A survey of the literature for power density values was performed. Data from fluoride salt fast reactors (FSFRs) were included on the basis that the factors limiting the power density (which include heat transfer limitations, structural material limitations due to neutron-induced radiation damage and corrosion) are likely to be similar for halide fuel salt systems.

The CSFR designs found in the literature have power densities that range from 100 kW/L to 364 kW/L but no explanation was found as to how the designers obtained their values (see Table 2-1). Given the relatively wide range of plausible power density values, 300 kW/L is adopted as the intermediate value in this evaluation. Since these results are expected to be sensitive to power density, the sensitivity is explored parametrically bounding values of 100 kW/L and 500 kW/L.

2.4.4 Cross Sections

Only two of the CSFR designs in the open literature have accessible cross sections. The CHLOROPHIL cross sections are limited in number, based on dated cross section data circa 1974, and were processed by using unspecified codes and assumptions. The FSMSR cross sections are comprehensive and based on current data and processing methods documented in [Davidson, et al., 2019]. Consequently, the FSMSR cross section set is generally preferred. However, the FSMSR cross sections would have to be extracted from a dataset in a format that is incompatible with ORIGEN2.

Extracting the fission product cross sections did not seem feasible or to be necessary to accomplishing the objectives of this study. Instead, a hybrid approach that began with an existing ORIGEN2 cross section library for the core of an advanced SFR using U-Pu oxide fuel [Croff et al., 1983, AMORUUUX core region design variant] is used. The cross sections in the SFR library for most radionuclides should be similar to the cross sections for the radionuclides in the CSFR fuel salt because (1) the neutron spectra of a generic SFR and CSFR are similar, as shown in Figure 2-1 [Diamond et al., 2018], and (2) the SFR neutron spectrum used in Croff et al. [1983] shown in Figure 2-2 is similar to the SFR spectrum shown in Figure 2-1.

On this basis, the existing ORIGEN2 SFR cross-section library from Croff et al. [1983] will be used for most radionuclides. However, noting that the purpose of this study is to calculate fission product buildup and removal from the fuel salt, it is desirable to get more recent values of the cross sections for key radionuclides, which are defined here as those in high concentrations and/or those producing the fission products. These radionuclides are ²²Na, ³⁵Cl, ³⁶Cl, ³⁷Cl, ²³⁵U, ²³⁸U, and ²³⁹⁻²⁴²Pu. Cross sections for these key radionuclides were manually extracted from the data set documented in [Davidson et al., 2019] provided by ORNL for the key species and substituted into the existing ORIGEN2 SFR library.

2.4.5 Fraction of Fuel Salt in the Core

A fraction of the fuel salt is continuously outside of the core region because it is being circulated to and from the primary heat exchanger or is undergoing cleanup processing to remove insoluble radioelements. This fraction does not produce significant power because it is in a sub-critical configuration and, in effect, being outside the core for a fraction of the time lowers the average power density of the fuel salt. This fraction is determined by design decisions related to the size and location of various pieces of equipment that will be made well into the future. There are currently no known theoretical limits to the fraction of the salt outside of the core based on physical and/or chemical reasons.

Based on the results of the literature search summarized in Table 2-1 and a cursory survey of other halide-based liquid-fuel fast salt reactor designs that had such information, the fraction ranges from about 0.4 to 0.75. On balance, an intermediate value of 0.6 seems reasonable and it is adopted for this study. Varying this parameter within the identified range is not expected to have a significant effect of the results; therefore, it is not addressed parametrically.



Figure 2-1

Neutron flux spectra for sodium fast reactor (SFR), light water reactor (LWR), fluoride saltcooled high-temperature reactor (FHR), intermediate spectrum (IS) MSR, and CSFR concepts. Courtesy of Brookhaven National Laboratory, U.S. Department of Energy [Diamond et al., 2018]. Legend modified from original for consistent terminology.

2.4.6 Fuel Salt Composition

Parameters that need to be quantified concerning the fuel salt composition are the elemental and isotopic concentrations of the key species which are sodium, uranium, plutonium, and chlorine. The initial fissile material in the CSFR fuel salt may be enriched uranium and then continue to operate on bred plutonium plus depleted uranium feed. However, for simplicity, this study assumes that plutonium is available and is used as the fissile material in the fuel salt from the outset to reduce the number of depletion time steps required to fission an alternative initial fissile material (for example, enriched uranium) and reach the equilibrium fuel salt composition in which plutonium is the only significant fissile material.

2.4.6.1 Fuel Salt Elemental Composition

The fuel salt composition can be initially defined by the U/Na ratio. The eutectic point of U-Na which has the lowest liquidus temperature (453°C for U/Na and 520°C for Pu/Na) is (U/Na or Pu/Na 33/67 mol%) which means unwanted freezing using this composition would be least likely during situations and at locations having relatively low temperatures. There are five values for the U/Na ratio presented in Table 2-1. Of these, two ratios (for NFCOC and FSMSR) are not plausible because they would increase the liquidus temperature by about 100°C.



Figure 2-2 The 126-energy-group neutron spectrum for the core of an advanced SFR using U-Pu oxide fuel. Courtesy of Oak Ridge National Laboratory, U.S. Department of Energy [Croff et al., 1983].

The other three U/Na ratios range from 35/65 mol% to 38/62 mol%. Based on the similarity of the eutectic composition to these three values, the composition of 33 mol% U and 67 mol% Na is adopted for the purpose of determining the fuel salt density.

2.4.6.2 Chlorine Enrichment

The various CSFR designs assumed ³⁷Cl enrichments ranging from natural (about 25%) to 100%, and some other studies [Gregg, 2018; Maulsoff, 2019] have assumed >99.9% ³⁷Cl enrichments. There is currently no evident basis for selecting a value because it depends on a number of unresolved issues. These issues include the availability of technology to achieve the higher enrichments, and the balance between the cost of the higher enrichments and the benefits of an improved neutron economy and reduced ³⁶Cl production. For the present study, an enrichment of 100% (as in the FSMSR model) is assumed for consistency with the cross sections for key FSMSR species, which were based on this value. The results of this study are not expected to be especially sensitive to this parameter; however, reducing the ³⁷Cl enrichment would increase non-productive neutron absorption and lower the breeding ratio. Accordingly, this effect is analyzed parametrically.

2.4.6.3 Uranium-235 Concentration in Depleted Uranium

The ²³⁵U concentrations in the uranium feed that were found in the literature consist of two low values (0.07% and 0.12%). Two other designs just specified depleted uranium but did not quantify the ²³⁵U concentration, and one does not provide any information. Public reporting indicates that current enrichment technology (gas centrifuge) is being operated in western countries to produce tails having ²³⁵U concentrations in the range of 0.18-0.22% [WNA 2020]. Therefore, the average value of 0.2% is adopted. The results of the present study are not

expected to be particularly sensitive to this parameter because the ²³⁵U concentration is small and would only account for a small fraction of the fissions compared to those from plutonium and ²³⁸U. Consequently, this factor is not examined parametrically.

2.4.6.4 Initial Plutonium Isotopic Vector

The plutonium vectors given in the five reactor designs include NFCOC and REBUS that specify LWR reactor grade having a relatively high ²⁴⁰Pu concentration, two (MCSFR and CHLOROPHIL) in which the complete vector is not given, and one (FSMSR) that specifies a relatively high-grade composition having a relatively low Pu-240 concentration. The FSMSR vector is adopted here as the basis for the initial CSFR core because it aligns with the adopted plutonium cross sections. The isotopic vector is: 0.076 wt% ²³⁸Pu; 94.289 wt% ²³⁹Pu; 5.306 wt% ²⁴⁰Pu; 0.242 wt% ²⁴¹Pu; and 0.087 wt% ²⁴²Pu.

2.4.6.5 Plutonium Concentration in Heavy Metal

The plutonium concentrations in the CSFR models found in the literature have a wide range. Based on evaluation in Section 2.3.2, the plutonium concentration values for CHLOROPHIL, REBUS, and NFCOC designs are discounted while the values for MCSFR and FSMSR are similar. At the start of this study, the effects of changing the initial plutonium concentration were not known. Consequently, the FSMSR value of 10 wt% is adopted as the baseline and a range from 5 wt% to 15 wt% is explored parametrically.

2.4.7 Fuel Salt and Heavy Metal Density

Molten salt reactor power production rates are typically expressed in terms of power density (kW/L). However, ORIGEN2 power input is on a mass basis (for example, specific power as kW/kg heavy metal). Accordingly, two additional parameters are needed to convert power density to specific power: the density of the fuel salt (g/cm³) and the fraction of that density attributable to heavy metal (for example, grams of heavy metal per gram of fuel salt). These parameters are quantified in the following two sections.

2.4.7.1 Fuel Salt Density

Information on the density of fluoride salts is not useful when analyzing CSFRs because of the difference in the mass of the chlorine anion as compared to the fluorine anion, and because essentially all of the density information for fluorine fuel salt is for the Th-U fuel cycle salt based on LiF-BeF₂. This means having to rely on information developed for CSFRs such as that in Table 2-1. However, the density data in Table 2-1 is not adequately characterized (for example, the temperature at which the density was determined is not provided) and the density values do not appear consistent with the associated fuel salt compositions. Accordingly, the CSFR fuel salt density value used in this study is derived from Desyatnik et al. [1975] as described below.

The fuel salt density of the initial CSFR fuel salt depends on the proportions of the fuel salt constituents. For this study, the initial fuel salt is assumed to be composed of sodium chloride, uranium trichloride, and plutonium trichloride. Uranium tetrachloride may also be present at a concentration of a few to several percent [Harder et al., 1969] if it is involved in salt redox control but its effect on density is not evaluated in this study due to a lack of data. Based on the discussion of the evaluation of CSFR literature in Section 2.4.6, the U/Na ratio is assumed to be
the eutectic composition (33/67 mole %) for the purposes of establishing the fuel salt density. The density of plutonium trichloride in the fuel salt was assumed to be the same as that for uranium trichloride on the basis that the densities of the solid plutonium and uranium trichlorides are similar (5.71 g/cm³ and 5.5 g/cm³, respectively) [Katz and Sheft, 1960] and the lack of any other information. The effects of fission product buildup on the density are not evaluated because of a lack of data but, in general, increasing fission product concentrations are expected to lower the fuel salt density.

The density of the fuel salt also depends on the temperature of the salt which is assumed to be the average temperature of the fuel salt in the reactor core. As with the fuel fraction, fuel salt temperature is driven by design considerations and not fundamental physical properties. Typically, a higher temperature is preferred because it allows for higher thermal efficiency when producing electricity or higher quality heat if the energy is being used directly in process applications. However, the higher temperatures need to be balanced against the cost of using more refractory materials of construction and higher degradation rates of those materials. Striking this balance is informed by weighing the business or mission need for high temperature operation versus increased operational and performance challenges from material degradation.

Temperatures that were found in the literature ranged from about 700°C to 1000°C with no explanation or justification provided (see Section 3.3). The upper end of this range is currently unrealistic due to material limitations and 900°C appears to be a more reasonable upper bound. The lower end of the range is defined by the need for the fuel salt to remain above the liquidus temperature (~470°C for just uranium, somewhat higher for U/Pu mixtures), plus a margin to ensure that the salt would not freeze in cold spots in the fuel salt loop (add at least 50°C, perhaps 100°C) and the temperature increase in the fuel salt from the salt inlet to the core to the core midplane (again a variety of values were found) so a lower value of 700°C does not seem unreasonable. On this basis, the average of 900°C and 700°C (that is, 800°C) is adopted in this evaluation. The implications of using an average temperature represents an area of uncertainty that remains to be investigated. The results of the present study are not expected to be particularly sensitive to the fuel salt density; therefore, it will not be analyzed parametrically.

With the assumed fuel salt composition and temperature, results from a correlation of the density of fuel salt comprised of various proportions of uranium and sodium chlorides as a function of temperature based on experimental results [Desyatnik et al., 1975] was used to calculate the fuel salt density. From the preceding text, the adopted U/Na ratio is 33:67 mol% and the average temperature is 800°C, which yields a fuel salt density of 2.96 g/cm³.

2.4.7.2 Heavy Metal Density

The fraction of the fuel salt density attributable to heavy metal (uranium and plutonium) is based on the fuel salt composition given in [Davidson et al., 2019] as described in the evaluation of CSFR literature (above). Given that fuel salt composition, a straightforward calculation (multiplying each element by its molecular weight and using the sum of the products to normalize the concentration of each element) leads to a value of 0.5 grams of heavy metal per gram of fuel salt.

2.4.8 Reactor and Fuel Parameters Selected for CSFR Model

Table 2-2 summarizes the reactor and fuel parameters selected for the reference CSFR reactor model.

Parameter	Value or Assumption	Comments
Reactor type	Fast spectrum liquid chloride fuel, U-Pu fuel cycle	
Power density (specific power)	300 kW/L of fuel salt (121.6 kW/kg HM) in the core with parametric variations of 100 kW/L (40.54 kW/kg HM) and 500 kW/L (202.7 kW/kg HM)	
Fuel fraction in core	0.6	
Fuel salt used in ORIGEN2 calculations from [Davidson et al., 2019]	NaCl:UCl₃:PuCl₃ 70:27:3 mol%	Single fuel salt to avoid the <i>plumbing problem</i> anticipated in early MSBR designs
Initial plutonium concentration	10 wt% with parametric variations of 5 and 15 wt%	Pu will be substituted 1:1 for U to maintain 70 mol% Na
Chlorine enrichment	100% ³⁷ Cl	
Depleted uranium feed enrichment	0.2%	
Plutonium vector	0.076 wt% ²³⁸ Pu; 94.289 wt% ²³⁹ Pu; 5.306 wt% ²⁴⁰ Pu; 0.242 wt% ²⁴¹ Pu; 0.087 wt% ²⁴² Pu	
Core salt density and heavy metal concentration	2.96 g/cm ³	50 wt% heavy metal

Table 2-2

Reference	CSFR	nuclear	and fue	parameters

2.5 Fuel Salt Cleanup for Radionuclide Removal

Liquid-fuel MSRs require removal of radionuclides from the fuel salt for three primary reasons:

- To avoid undesirable impacts arising from low solubility of species, such as noble gases and noble metals, that can result in complications, such as gas pockets and precipitation/plate-out, if not removed.
- To maintain fuel salt reactivity and reactor performance by reducing parasitic neutron absorption caused by buildup of certain radionuclides.
- To maintain adequate reactivity control margin as fissile material is produced via conversion reactions.

Conceptually, continuous removal of radioelements by chemical or physical processes is the same as removal by radioactive decay or neutron capture. This is how it is treated in ORIGEN2: another term is added to the Bateman equations that calculate the production and consumption of each radionuclide and, by summation, radioelement. The units of the continuous removal rate used by ORIGEN2 are the same as for a cross section multiplied by the neutron flux or the decay rate: inverse time such as seconds⁻¹ or 1/s.

Early in MSR development, for example, Ferguson et al. [1969], removal rate of radioelements by chemical and physical processes were expressed in terms of a cycle time, the units of which are time (seconds). In essence, the cycle time is the amount of time it takes to process the entire inventory of a salt loop, tank, or other component once. Thus, the cycle time depends on the inventory of the fuel salt and the flow rate of salt through the processing equipment. The removal rate also depends on the efficiency of the removal processes, but this was assumed to be 100% because the specific technologies and their efficiency that underpin selection of cycle times were not found in the literature. This assumption is non-conservative but not likely to be significant because the cycle times used in cleanup processing are short enough keep insoluble fission product concentrations low. The use of cycle time in decay calculations is similar to that of radioactive half-life: the inverse of the cycle time is the cleanup processing removal rate, and it is used just as a radioactive decay constant is used.

The primary objective for evaluating of CSFR fuel salt sustainability is to investigate whether continued recycle and reuse of the fuel salt is feasible, and to assess the changes in the fuel salt elemental composition and their potential impacts as reuse proceeds. The preliminary analysis [EPRI, 2021] was based on a relatively simple CSFR model implemented in a spreadsheet which (a) assumed that all fission products were in the form of the long-lived or stable end-member element of each decay chain, (b) assumed instant removal of 100% of each insoluble radioelement at the end of each fission product decay chain during cleanup processing, and (c) was capable of calculating the equilibrium fraction of fission products in the fuel salt but not the changes in the radionuclide and radioelement composition of the fuel salt.

This study incorporates and accounts for the continuous removal of insoluble fission products on an element-by-element basis plus buildup and depletion on a radionuclide-by-radionuclide basis using ORIGEN2. Doing so requires more detailed information on the composition of the fuel salt than did the preliminary analysis. Section 2.4 discusses the more detailed assumptions and parameter quantification concerning the rate at which each insoluble element is removed from the CSFR fuel salt and the rate at which fresh depleted uranium chloride is fed to the CSFR.

Reference Molten Chloride Salt Fast Reactor (CSFR) Model

2.5.1 Liquid-Fuel MSR Cycle Times

Some information on representative cycle times for liquid-fuel MSRs is available in the literature and described in Appendix B. Key observations are summarized below.

• A single cycle time is assigned to a group of radioelements.

In the literature on cycle times, the elements are grouped according to their chemical family (for example, noble gases, halogens, alkali metals) where they are assumed to behave similarly in postulated but unspecified removal processes. The element groups are relatively standard across the MSR literature and the rows in Appendix B are organized by these groups. These are likely rough-order-of-magnitude estimates. The name of and membership of the relatively standard groups are listed in the first two columns of Table 2-3, the third column contains values for cycle times taken from a single source for a MSBR, and the fourth column describes the MSBR removal process for each element group [Engel et al., 1978; Rosenthal et al., 1972]. The CSFR removal processes are discussed below.

• Most cycle times in literature are for fluoride-salt MSR designs.

Eleven cycle time sets are found that are complete enough for this study. Nine sets are for a thermal-spectrum thorium fuel cycle using a fluoride carrier salt (usually FLiBe) in an MSBR, and one set was for a spectral shift thorium fuel cycle MSBR. Only one set of cycle times corresponds to a fast-spectrum CSFR using a U-Pu fertile-fissile material in a sodium chloride carrier salt.

• Cycle time sets are predominately associated a single source.

Seven of the nine cycle time sets identified are directly linked to one source–ORNL. The extent to which the other two were derived from ORNL research is unknown.

• Cycle time sets vary widely.

There is significant variability of the cycle times in the various cycle time sets. The reasons for the variability are not known because the processes and analyses underlying the cycle times have not been identified. One notable exception is the discard cycle time of 3435 days, which appears often. The importance of the variability of cycle times is mitigated by the fact that the cycle times for specific elements can be increased substantially while still keeping the concentration of the elements at low levels. For example, if only discard were occurring the equilibrium concentration of xenon for the 300 kW/L case would be about 12,000 g/MTHM. A cleanup processing cycle time of 20 seconds results in a xenon concentration of 0.0034 g/MTHM; therefore, even a 100-fold increase in cycle time would not result in a significant xenon concentration increase.

• Cycle time sets are incomplete.

There are many radionuclides for which cycle times are not available from any source. These are mostly activation products or other contaminants from in-leakage and maintenance—for example, nitrogen, oxygen, sulfur. The rate at which these will build up and their importance to reactor operation via corrosion, deposition on surfaces, and other effects, is largely unknown. Consequently, these radionuclides are not considered further given this lack of information.

Group	Elements in Each Group	MSBR Cycle Time	MSBR Removal Process	CSFR Removal Process
Noble gases	Kr, Xe	50 sec	Cleanup processing: inert gas sparging to off-gas system	Same
Semi-noble and noble metals ^{a,b}	Zn, Ga, Ge, As, se, Nb, Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, Sb, Te	2.4 hr	Cleanup processing: plating on surfaces in reactor vessel and heat exchanger or filtration ^c	Same
Uranium	Uranium	10 days	Volatilization in fluorinator, returned to carrier salt and recycled	Not applicable to CSFR
Halogens	Br, I	10 days	Volatilized in fluorinator and scrubbed out in helium recycle system	Not applicable to CSFR
Zr ^c , Pa	Zr, Pa	10 days	Extraction by Bi-Li alloy then hydrofluorination into Pa decay salt	Not applicable to CSFR
Corrosion products	Ni, Fe, Cr	10 days	Extraction by Bi-Li alloy then hydrofluorination into Pa decay salt	Not applicable to CSFR
Trivalent rare earths	Y, La, Ce, Pr, Nd, Pm, Gd, Tb, Dy, Ho, Er	25 days	Extraction into Bi-Li alloy then tranfered into dilute Bi-Li solution	Not applicable to CSFR
Divalent rare earths	Sm, Eu, Sr, Ba	25 days	Extraction into Bi-Li alloy then transferred into dilute Bi-Li solution	Not applicable to CSFR
Alkali metals	Rb, Cs	10 days	Extraction into Bi-Li alloy then accumulation in LiCl	Not applicable to CSFR
Carrier salt	Li, Be, Th	~15 vears	Discard ^d	Same

Table 2-3Salt processing element groups, cycle times, and processes

^a These are sometimes two separate groups

^b These can appear in the reactor gas space [Ferguson et al., 1969] as a result of sparging, presumably in the form of a fog/mist-aerosol, that could be removed by filtration.

° Sometimes included with semi-noble elements in the literature but considered separately here.

^d Discard is defined as removing bulk salt for management by other means including disposal.

2.5.2 Liquid-Fuel MSR Fuel Salt Cleanup Systems

Maintaining the modest breeding gain of a single-fluid MSBR design–between 0.04 to 0.08– requires the incorporation of complex fuel salt processing to minimize parasitic neutron absorption in ²³³Pa, ¹³⁵Xe, and the rare earth elements [Robertson, 1971]. This complexity is evident in Figure 2.3 below, which depicts the single-fluid MSBR fuel processing flowsheet for the original ORNL conceptual design [ORNL, 1971].





Figure 2-3 Single-fluid MSBR fuel salt chemical processing flowsheet. Image courtesy of Oak Ridge National Laboratory, U.S. Department of Energy (as Atomic Energy Commission). [ORNL, 1971]

The situation for a CSFR is different, primarily because essentially all absorption cross sections are much smaller than in a thermal neutron spectrum and a U-Pu fuel cycle has good neutronics properties (for example, neutrons produced per fission) in a fast spectrum. In particular, U-Pu fuel in a fast neutron spectrum reactor exhibits far less unproductive neutron absorption compared to a U-Pu cycle in thermal spectrum MSRs or thorium cycle MSRs, which leads to a substantial neutron excess. The neutron excess allows for a relatively high breeding ratio without having to process the salt to remove rare earth elements and ²³⁹Np—the conceptual equivalent radionuclide in a U-Pu cycle to ²³³Pa in a thorium fuel cycle. This allows for processing to be less complex for a CSFR as compared to a MSBR.

Figure 2-4 represents the same MSBR fuel salt processing operations as those in Figure 2-3; however, Figure 2-4 has been simplified by combining various unit operations to improve clarity. In a MSBR, radionuclides are planned to be removed in five streams: (1) off-gas treatment of mainly noble gases as a part of cleanup processing; (2) fuel salt treatment to remove noble metals and other particulates by filtration as a part of cleanup processing; (3) fluorination that isolates uranium temporarily for removal of rare earths; (4) pyroprocessing to remove the rare earths for disposal and to separate ²³³Pa for decay to ²³³U (outside of a neutron flux); and (5) discard for bulk salt removal for disposal. The two schematic vessels constituting fluorination and reduction-extraction steps account for almost all of the complexity shown in detail in Figure 2-3.



Figure 2-4 Simplified, generic MSR fuel salt treatment and processing flowsheet

The equivalent of the fluorination and pyroprocessing functions are not required in a CSFR because of the favorable CSFR neutron economy. Information found in the literature revealed that designers typically assumed these functions will not be present in a CSFR. Thus, cycle times are not needed for CSFR radioisotope groups that would have been removed by these processes in a MSBR. The radioisotope groups for which cycle times are no longer needed are indicated in the rightmost column of Table 2-3. Consequently, the reference CSFR design is greatly simplified as only cleanup processing and bulk salt discard are needed.

2.5.2.1 Adequacy of Available CSFR Model Cycle Time Information

The only CSFR model having cycle times found in the literature has a U-Pu fuel cycle and a sodium chloride carrier salt [Davidson et al., 2019]. Unfortunately, examination reveals some limitations important for the present study. The fuel salt for Taube and Ligou [1974] is represented as a homogenization of two salts, a static core fuel salt and circulating blanket salt. This results in a fuel salt composition containing too much sodium chloride to be usable for the CSFR because of non-feasibly high liquidus temperature of nearly 600°C. Furthermore, Davidson et al. [2019] assumes a single 3-day cycle time for all radioisotopes except plutonium to coincide with the time step being used in the modified version of SCALE, effectively resulting in 100% removal of all radioelements except plutonium every three days [Davidson et al., 2019]. While this may be adequate for the purposes of Gregg [2018], it is not suitable for the desired evaluation of fuel salt sustainability for the present study, which requires modeling buildup and depletion of individual radioisotopes.

2.5.2.2 Selection of Cycle Times for a CSFR

The discussion above leaves two options. The first option is to adopt the set of cycle times for the Davidson et al. [2019] CSFR model containing one cycle time for all elements except plutonium. The second option is to use cycle times for the seven MSR designs which use thorium-based fuel salt the basis for selecting a set of cycle times to be used for the CSFR. In either case, information on how the cycle times were derived and/or measured is unavailable, as is information relating MSBR cycle times to those for CSFRs. Faced with two non-ideal choices, selection of the second option is justified based on the following rationale:

- MSR and CSFR both use fuel salts having anions that are members of the same chemical group-the halides-so they should behave similarly;
- Reactor designers who are presumably experts in molten salt chemistry put both chlorine and fluorine in same group for purposes of assigning cycle times (see Appendix B);
- The processes used for radioelement cleanup processing and discard relevant to the CSFR are mainly physical (for example, sparging and filtration) and not chemical (for example, precipitation and solvent extraction) so there should be less dependence on the anion, and
- For all elements except plutonium to have the same cycle time in the CSFR model found in the literature does not seem plausible.

As a consequence, the approach used in this study is to select cycle times from the MSR designs shown in Appendix B with a bias toward the more recent ORNL values for MSBRs but excluding [Davidson et al., 2019] which was not considered for reasons stated above. The resulting set of cycle times for a CSFR are given in Table 2-4. The cycle time for *discard* is a placeholder value from the literature because the required discard rate is actually determined by the need to remove plutonium to control the reactivity of the core. This topic is elaborated in the next section.

2.5.3 Radioisotope Removal by Bulk Salt Discard

The fuel salt discard rate is the rate at which bulk fuel salt must be removed to maintain a nearly constant plutonium concentration over years-to-decades in the ORIGEN2 calculation. Secondarily, discard also contributes to limiting the fission product concentration in the fuel salt. The percentage of the bulk fuel salt discarded each year for the parametric range of specific power values considered in the present study and a target plutonium concentration of 10 wt% of the heavy metal are shown in Table 2-5. The feed rate is discussed in the following section. The process by which the salt feed and discard rates were determined is described below Section 3.

Taken alone, these values are not particularly informative. However, they are required to calculate the fuel salt composition as a function of time. And since discard rate is also inversely related to doubling time, a higher discard rate indicates a shorter doubling time.

Group	Elements in Group	Cycle Time	Potential Removal Process(es)
Noble gases	He, Ar, Ne, Kr, Xe, Rn	20 sec	Inert gas sparging to off-gas treatment system
Semi-noble and noble metals ^a	Zn, Ga, Ge, As, Se, Nb, Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, Sb, Te	20 sec	Salt filtration or plating out on surfaces in reactor vessel and heat exchanger ^b
Corrosion Products	Cr, Fe, Ni	3435d	Discard ^c but not included in the model because of inadequate information on metal activation, corrosion rates, and corrosion product concentrations
Carrier salt	All other elements	3435d	Discard

Table 2-4Adopted CSFR cycle times

^a These are sometimes listed as separate groups although typically with the same cycle time.

^b These can also appear in the reactor gas space [Ferguson et al., 1969] as a result of sparging, presumably as a fogmist-aerosol-particulate, that could be removed by filtration in the off-gas system or deposition on various surfaces in the primary loop.

^c Discard is defined as removing bulk salt for management by other means including reuse or disposal. This discard rate is provisional pending ORIGEN2 results (see the following section).

Table 2-5Salt feed rate and fission product discard rates for the adopted parametric range of CSFRpower densities

Average Specific power, kW/kg HM (Core Power density, kW/L)	Biennial fertile salt batch feed rate, % HM	Biennial fuel salt batch discard rate, %	Biennial fuel salt batch retention rate, % ^a
40.54 (100)	15	12	88
121.6 (300)	29	20	80
202.7 (500)	54	31	69

2.5.4 Fuel Salt Replenishment

At a conceptual level, a CSFR is a reactor that converts fertile uranium into excess plutonium (breeding) and fissile uranium and plutonium into fission products. Some fission products are removed at varying rates by cleanup processing and all three species are removed by discard as discussed in the previous two sections. As a result, a CSFR needs to be fed depleted uranium trichloride in a sodium chloride carrier salt to compensate for what is being removed. In the present study, it is assumed that the uranium is fed in the form of the 30 mol% uranium trichloride–70 mol% sodium chloride. Further, as an aid in verifying the ORIGEN2 results, it is assumed that the CSFR feed and discard occur in a single batch every two years.

3 ORIGEN2 MODELING OF REFERENCE CSFR

This section describes the CSFR model in two steps. In the first step the conceptual model adopted for the present study is summarized. In the second step the use of ORIGEN2 to implement the model is described.

3.1 Conceptual CSFR model

In the conceptual CSFR model depicted in Figure 3-1 the hypothetical core is assumed to initially comprise 1.0 MTHM (900 kg depleted uranium + 100 kg plutonium), and 220 kg sodium cation plus 820 kg chlorine anion apportioned among the cations for a total of about 2,000 kg of fuel salt. Irradiation of the fuel salt proceeds as follows (Note, the values below are approximate, rounded, and provided for illustration purposes—they are not study results):

- 1. The fuel salt is irradiated for a set time period (for example, one year) at a constant average specific power corresponding to one of the power densities adopted for the present study multiplied by the fuel fraction in the core.
- 2. For the intermediate case (300 kW/L) the fuel burnup is about 30 GWd/MTHM.
- 3. Since the CSFR is a breeder reactor, the following occurs during the irradiation of one metric ton heavy metal:
 - a. The gross amount of plutonium produced is about 40 kg and the net amount remaining after fissions is about 15 kg HM.
 - b. The gross amount of fission products produced is about 30 kg (25 kg from Pu fissions and 5 kg from U fissions) and the net amount remaining after continuous cleanup processing is about 15 kg.
 - c. The net amount of uranium decreases by 45 kg from about 5 kg of fast fission and producing about 40 kg of gross plutonium.
- 4. To restore the initial plutonium concentration to keep fuel salt reactivity at acceptable levels about 10 kg of plutonium (10% of the initial amount of 100 kg) is removed by discarding 10% of the fuel salt in a single batch.
- 5. The discard of plutonium also removes about 10% of the uranium (about 85 kg) and fission products (about 1.5 kg) in the fuel salt. The discarded salt is assumed to be accumulated in an appropriate storage configuration. When 100 kg of plutonium plus the associated uranium and fission products have accumulated in storage the salt is used as the initial inventory of a subsequent CSFR.
- 6. About 130 kg of depleted uranium-sodium chloride is then added to the fuel salt in a single batch to replenish the uranium removed by fission, plutonium production, and discard.

- 7. The fuel salt now also contains about 13.5 kg of fission products that were not removed by cleanup processing or discard.
- 8. The fuel salt is then irradiated for another time increment (one year in this example) after which steps 3 7 repeat until the equilibrium fission product concentration is reached. The plutonium concentration in the heavy metal at the beginning of each year remains constant at 10 wt% and uranium concentration remains constant at 90 wt%. The fission product concentration increases until the equilibrium concentration is reached.



Figure 3-1 Conceptual CSFR material flows

3.2 ORIGEN2 Implementation

ORIGEN2 modeling assumes constant heavy metal inventory by mass. Cleanup processing occurs continuously but discard and feed occur in discrete batches at rates so that the heavy metal mass in the core, comprising 10 wt% plutonium/90 wt% depleted uranium trichloride in sodium chloride carrier salt, remains constant at the beginning of each biennial (two-year) cycle

after batch discard and feed have occurred.³ A parametric variation in which the amount of fuel is kept constant is described in the portion of this section that discusses reactor physics results (Section 3.4.3).

ORIGEN2 calculations are driven by a simple programming language in which each line is a command for ORIGEN2 to perform some action. This structure provides considerable modeling flexibility. The CSFR model was implemented using the programming language and input parameter values and assumptions described in Section 2 as follows:

- 1. Libraries containing the decay, cross section, and fission product yield data are read and put in a suitable format (a matrix of equations to be solved) for what, in essence, amounts to the Bateman equations. This input includes the cycle times for cleanup processing but not discard rates.
- 2. The composition of the fuel salt constituents (U, Pu, Na, Cl) are read.
- 3. A unit of fuel (1.0 Initial MTHM) is irradiated at constant specific power for two years. During this time cleanup processing (continuous removal of insoluble fission product elements) is performed at the rates defined in Table 2-4 except for discard.
- 4. The heavy metal content is rebalanced biennially by removing plutonium in the form of bulk fuel salt (discard) and adding feed salt composed of depleted uranium-sodium chloride to maintain a mixture of one MTHM of a 90 wt% U–10 wt% Pu.
- 5. The foregoing steps are repeated until the prescribed irradiation duration is achieved. The process is repeated for parametric variations, for example, various specific powers.

While ORNL and others have suggested discard rates (albeit mainly for thorium fuel cycle thermal reactors), it is apparent that while the suggested values are in the ballpark, they do not maintain a constant 10 wt% plutonium concentration at the start of each cycle. Additionally, the appropriate discard and feed rates are not constant but instead are a function of the specific power—a higher specific power causes the neutron flux to increase which consumes the uranium and produces plutonium and fission products more quickly, and so requires higher discard and feed rates.

Determining the appropriate salt discard and feed rates was accomplished by manual iteration using the following approach:

- 1. The discard rate required to obtain a constant plutonium concentration at the end of each two-year cycle of irradiation for a 100 full-power years (FPYs) time horizon was estimated.
- 2. If the plutonium concentration was increasing over time, the discard rate was increased and vice versa for a decreasing plutonium concentration until a constant plutonium concentration was obtained for the 100-year duration.
- 3. Next, the feed rate was adjusted so that the total mass of heavy metal remained constant at about 1.0 MTHM.

³ **Note:** It is recognized that this configuration is unrealistic; in reality, the fuel volume would be held constant in an actual reactor, which would require reduction of heavy metal content to make room for fission products if not removed; however, for the purposes of evaluation, the simplicity of the batch assumption aids in verifying the results.

After some experience, a few trials were adequate to converge on a solution.

Buildup and depletion calculations were performed at average specific powers of 40.54 kW/kg HM, 121.6 kW/kg HM, and 202.7 kW/kg HM. These values correspond to the three adopted core power densities of 100 kw/L, 300 kW/L, and 500 kW/L (see Table 2-2) after converting the power densities to a mass basis using the salt density and heavy metal concentration adopted earlier and adjusting the result downward to account for the fuel salt outside of the core. The ORIGEN2 command input for the intermediate case is provided in Appendix C.

3.3 Limitations and Uncertainties of ORIGEN2 CSFR Model Analysis

The limitations and uncertainties associated with parameterization and analysis methods are important to be recognized in any modeling endeavor. Below, limitations and uncertainties associated with this evaluation are briefly summarized along with a qualitative impact assessment.

3.3.1 Limitations

This study revolves around calculating the buildup and depletion of fission product in CSFR fuel salt and the bulk removal of fission products from the primary system. The adopted approach used the ORIGEN2 point (that is, zero-dimensional) computer code which, by definition, cannot account for spatial effects such as energy and spatial self-shielding when determining radionuclide cross sections, or account for varying neutron fluxes and spectra in different parts of the CSFR. This limitation is unlikely to be important in meeting the objectives of this evaluation, that of elucidating fission product concentrations and compositions, for the following reasons:

- Self-shielding is important in thermal reactors where there is a substantial neutron flux in the resonance region, but it is not particularly important in fast reactors where most of the neutrons have an energy higher than the resonance region.
- The focus of the present study is on fission products. With few exceptions, the production and destruction of fission products depend on burnup, fission product yields, and fission product half-lives. This is particularly relevant for fast spectrum reactors, in which neutron-induced production and destruction rates of fission products are small relative to fission product removal rates, which depend on half-lives and cycle times.
- The one-group cross sections used in ORIGEN2 were obtained from multi-group, multidimensional reactor physics calculations and reflect self-shielding to the extent it is important, albeit primarily for a solid U-Pu oxide fuel in a fast reactor.
- The ORIGEN family of codes has been used and verified for decades and has been generally shown to accurately reflect its input assumptions (feed composition, specific power or flux, irradiation and decay duration) although information that would allow verification for fuel salt reactors could not be found. Such verification is desirable to increase confidence in the results.
- The fraction of the fuel salt outside the core is accounted for in the *fuel salt fraction* parameter which accounts for the small burnup average power density being lower than the core power density per se. This parameter is design-specific and will not be known until a detailed CSFR design is available.

The fact that ORIGEN2 is zero-dimensional limits the ability to evaluate the neutronic impacts of fission product buildup (fission product poisoning) on the reactivity of the CSFR fuel salt as a function of time. The change in the fraction of neutrons absorbed by the fission products was used as a qualitative surrogate for reactivity.

3.3.2 Uncertainties

The major uncertainties in this study are discussed in this section to provide perspective for readers that may use the results as a basis for planning, research and development (R&D) activities, decision-making, and design.

3.3.2.1 Absence of a Suitable CSFR Design

A CSFR design containing all of the information needed for this study could not be found. The reasons for this are detailed in the evaluation of CSFR literature in Section 2. In short, models found in the literature search were unsuitable for one or more of the following reasons: incomplete description of the reactor and reactor parameters, unrealistic parameter assumptions, limited or no cross section information, designs that are not representative of current MSR concepts, apparent errors in calculations, or unexplained or indefensible assumptions in determining fuel salt compositions and other parameters. This situation led to the creation of a high-level, CSFR pre-conceptual design and quantification of the necessary parameters based on information from a number of literature sources. The model used herein adopted what appear to be reasonable and internally consistent values for each parameter required from the literature. However, the reasonableness of the adopted parameters is based on our review of the preponderance of the evidence found in the literature, general scientific and engineering knowledge, logic, and expert judgment.

This uncertainty in the results introduced by having to create a CSFR model to be evaluated was addressed in part by parametric variation of key parameters. For other parameters that had a relatively narrow range of values, a fixed value was adopted. In the course of conducting the analyses underlying this study, some insights were obtained that appear to mitigate the magnitude of the uncertainty introduced by this approach. In particular, for a given set of nuclear data (half-lives, fission product yields, cross sections) the fission product concentration depends on the average burnup of the fuel salt (MWd/MTHM or similar). The average burnup depends on the following hierarchy of parameters:

- 1. Core specific power (kW/kg HM) which is the input parameter for ORIGEN2. The specific power was varied parametrically in ORIGEN2 calculations and depends on:
 - a. The fuel salt power density in the core (kW/L) which was quantified based on information in the literature and—within limits that will ultimately be defined mainly by materials capabilities and economic considerations—can apparently have a wide range of values and so it was varied parametrically
 - b. The fuel salt density (for example, kg/L) which depends on:
 - i. The fuel salt elemental composition which was assumed to be the well-known eutectic composition (67:33 mol% NaCl:UCL₃). The extent to which the actinide content can be changed is limited to a relatively narrow range because increasing or

decreasing it from the eutectic point rapidly increases the melting point from the eutectic point.

- ii. The isotopic composition of fuel salt constituents which would have a small influence on density and depends on:
 - 1. The ²³⁵U concentration in the DU, which changes the mass per mole and was held constant.
 - 2. The isotopic enrichment of the chlorine which changes the mass per mole and was held constant.
 - 3. The isotopic vector of the initial Pu in the core which was taken from the literature but which is irrelevant soon after startup because the isotopic composition of the Pu is determined by cross sections and half-lives that define equilibrium.

Note: Because the isotopic masses in each element are within 6% (for example, ³⁵Cl and ³⁷Cl) or less of each other, varying them will have a small effect on density.

- c. The average temperature of the fuel salt was based on information in the literature. The density has a weak dependence on temperature and was assumed to be constant.
- d. The density of the eutectic composition, which was taken from a literature correlation, was assumed to be constant. Uncertainties in the correlation used to calculate the density were provided and were small.
- e. The fraction of the fuel salt density attributable to the heavy metal, that is, excluding sodium and chlorine.
- f. Irradiation duration (on the order of days) which is an independent variable but assumed to be constant at 100 FPY, at which time equilibrium had been reached in all cases.
- g. The fraction of the fuel salt being irradiated in the core (dimensionless) which has the effect of reducing the average power density. This fraction is essentially a design decision. The literature values range from 0.4 to 0.7. A value of 0.6 was adopted and assumed to be constant.

A key point of the foregoing hierarchy reinforces that the topmost parameter (the core specific power for a constant irradiation time), which this study found to have a major effect on the fission product concentration, was varied over a broad range (approximately 82 kW/kg HM \pm 67%). However, the variability of the specific power could be viewed as reflecting the cumulative uncertainty in all of the parameters on which it depends as listed above, not just the power density selected by the designer per se, especially because the range of most of the parameters is relatively small. Additionally, the uncertainty in the subsidiary parameters may not act in the same direction—for example, a lower average temperature increases fuel salt density, but this might compensate for a lower actinide fraction in the fuel salt which decreases fuel density. Taken together, this indicates that the primary uncertainty among the parameters discussed in this section is the design decision leading to the selection of the target specific power.

3.3.2.2 Cross Section Input Data

The single set of cross sections for a CSFR identified in the literature review of Davidson et al. [2019] was in a format where fission product cross sections would have had to be extracted from the product of cross section and yield in the data set, but the set of fission product yields used in Davidson et al. was not stated. The hybrid approach that was adopted here used fission product cross sections taken from an ORIGEN2 data set for an advanced oxide SFR on the basis that the neutron spectrum of the SFR was similar to that of a CSFR. Then, cross sections for high-concentration fuel salt constituents in the CSFR (U, Pu, Na, Cl) were extracted from the Davidson, et al. data set and manually substituted into the ORIGEN2 SFR cross-section set. Comparison of the cross sections for the key materials from the SFR and CSFR showed most were within about 20% of each other although ²³⁸U fast fission, sodium, and chlorine isotopes were further apart. In any case, the impact of these differences in cross sections should not be overly important to the objectives of this evaluation which are driven by the average fuel salt burnup to determine fission product concentration and the discard rate.

3.3.2.3 Cleanup Processing Cycle Times and Efficiencies

While several sets of cycle times for cleanup processing were found in the literature, none of the references described how the cycle times were derived. Since cleanup processing involves using unit operations such as filtration and carbon bed absorption, it would seem likely that evaluations were performed to determine: (a) the rate at which the material stream (fuel salt or off-gas) went through each unit operation and (b) the efficiency of that operation for removing the various chemical species in these streams. However, the available literature only reports the cycle time values for various element groups and provides no discussion of process efficiencies was found. Further, it does not seem plausible that nearly two dozen elements (noble gases, noble metals, and semi-noble metals) would have the same cycle time: 20 seconds. The potential impact of a wider range of cycle time values for the various elements is mitigated in this analysis by the fact that a sensitivity study (Section 2.5.1) showed that changing the cycle times by factors of 10 or even 100 makes little difference in the result–all reduce the insoluble elements to very small concentrations.

The ORIGEN2 results provide element-by-element details of fission product distributions resulting from cleanup processing and should be accurate enough to inform key aspects of early-stage reactor development such as allowing simulated fuel salts to be synthesized for material testing and providing the basis for measuring and/or calculating physicochemical properties of the evolving fuel salt composition. However, at this point the potential impacts of the evolving fuel salt composition on materials, components, and reactor thermohydraulic design are essentially unknown. Consequently, determining whether various fission product concentrations would constrain sustained fuel salt recycle is not possible.

3.4 ORIGEN2 Modeling Results and Discussion

ORIGEN2 modeling results describing CSFR fuel salt evolution with respect to lifetime and sustainability are presented and discussed below.

3.4.1 Total Fission Product Concentration as a Function of Time

3.4.1.1 Results

One key result of interest is the rate at which fission products build up in the fuel salt. The fundamental output from ORIGEN2 is the fuel salt composition (gram-atoms) as a function of time which ORIGEN2 can convert to a variety of units such as mass, decay heat, etc. and aggregate this nuclide-by-nuclide result by element and totals thereof. The relevant portions of the ORIGEN2 output of the fission product mass as a function of time for the three power levels were imported into a spreadsheet and are displayed in graphical form in Figure 3-2.



Figure 3-2 Buildup of bulk fission products in CSFR fuel salt as a function of time for three power densities

The time required to approach half of the equilibrium concentration—which is a measure of how long the fission product concentration remains relatively low—is of interest because the fuel salt may be reusable for a longer time if it has a low concentration. The result is that the 100 kW/L scenario reaches half-equilibrium in 8 FPY, the 300 kW/L in 6 FPY, and the 500 kW/L scenario in 3 FPY. Note that if fuel salt is removed from a parent reactor before fission product concentrations will continue to build up. Moreover, this approach will likely require supplementation of the parent reactor with additional fissile material. An example of calculated ORIGEN2 fission product concentrations as a function of time for the intermediate (300 kW/L) scenario is given in Appendix D.

3.4.1.2 Discussion

The buildup of fission products in the fuel salt proceeds as expected with the concentration increasing rapidly from several years out to a few decades and then asymptotically approaching equilibrium values where the combination of cleanup processing and bulk fuel salt discard balance the production rate. The equilibrium fission product concentrations are 8.5 wt% (100 kW/L), 16.8 wt% (300 kW/L), and 26.3 wt% (500 kW/L).

The equilibrium concentration of fission products for the 300 kW/L power density in this study is somewhat lower than that used in the preliminary analysis [EPRI, 2021]. This difference is attributed to the need to set fuel salt discard rates in the preliminary analysis assuming doubling times gleaned from the literature. In the current study, discard rates were adopted to maintain plutonium concentrations at 10 wt%. The preliminary analysis also employs a bounding approach and, therefore, does not consider scenarios comparable to the 100 kW/L and 500 kW/L power density scenarios evaluated herein. For context, the fission product concentration in typical (50 GWd/MTHM burnup) LWR fuel is about 5 wt% and in some solid fuel advanced reactors such as SFRs where the core fuel is planned to reach 150 GWd/MTHM the fission product concentration would be about 15%.

3.4.2 Effect of Cleanup Processing on the Fission Product Elemental Composition

3.4.2.1 Results

The element-by-element impact of cleanup processing could not be examined in the preliminary analysis [EPRI, 2021] because the computational approach (a simple spreadsheet) could not track individual radionuclides which are summed to yield the radioelement concentrations. However, ORIGEN2 was designed to do so and yields the results required to examine the distribution of fission product elements in the salt after sustained operation and cleanup processing. The relevant information was extracted from the ORIGEN2 output for the 300 kW/L scenario and put into graphical form which is shown in Figure 3-3. The element-by-element evolution of the fission product composition with cleanup processing for the 300 kW/L scenario out to 100 FPY is provided in Appendix C, Table C.1.

3.4.2.2 Discussion

The impact of a multi-element distribution that evolves over time on fuel salt characteristics (such as corrosivity, heat capacity, thermal conductivity, viscosity, solubility limits and volatility), along with how these changes would affect the continued use and reuse of the fuel salt are unknown because relevant information was not found in the literature. Addressing these unknowns will likely require an R&D program involving complementary experimental and computational components to provide the information needed by planners and designers and required by regulators.

3.4.3 Reactor Physics

3.4.3.1 Results

At equilibrium for the 300 kW/L scenario, the fission products account for about 11% of the neutron absorptions in the fuel salt; uranium and plutonium account for about 87% of neutron absorptions. At equilibrium neutron absorption by fission products would be about 3% for the 100 kW/L scenario and about 17.5% for the 500 kW/L scenario. About 97% of fission product neutron absorption is attributable to Sm, Nd, Eu, Zr, Pm, Gd, Pr, Dy, and Ce, all of which except Zr are rare earth elements that are not removed by cleanup processing and have relatively large cross sections. Concentrations of Ce, Pr, and Zr are reduced to the extent that precursor noble gas isotopes are removed by cleanup processing. The concentrations of fission products that are

ORIGEN2 Modeling of Reference CSFR

directly removed by cleanup processing in the fuel salt are negligible from a neutronics perspective because the assumed cycle times are sufficiently short (for example, CSFR values in Table 2-4) so as to make their concentrations negligible.

In the 300 kW/L scenario the carrier salt accounts for about 2% of the neutron absorptions at equilibrium. In a parametric variation, natural chlorine (25% 37 Cl) is substituted for the baseline 100% 37 Cl used in this study and neutron absorption by the carrier salt increased from 2% to about 10% of the total.

To investigate the effect of varying the initial plutonium concentration, a baseline CSFR ORIGEN2 model using an intermediate power density and its associated feed and discard rates (per Table 2-2) is exercised assuming initial plutonium concentrations of 5 wt%, 10 wt% (baseline), and 15% of the initial heavy metal at a power density of 300 kW/L. The results in terms of parameters such as fission product concentrations, element distribution and the flux from using the high and low plutonium concentrations are then compared to the baseline.

In short, varying the plutonium concentration has no effect on total fission product buildup or the fission product element distribution out to three significant figures. This result is obtained because the specific power and irradiation duration are the same in the variant scenarios, which means the total burnup is the same. This is consistent with the *rule of thumb* that producing 1 MWd of thermal energy requires fissioning 1 g of an actinide thereby producing 1 g of fission products [Stewart, 1985], the inevitable outcome is that the total fission product mass produced and the total mass remaining in the fuel salt does not vary with initial plutonium concentration.

Discussion of ORIGEN2 implementation in Section 3 noted that the scenarios in the present study are somewhat unrealistic because the amount of heavy metal was kept constant at 1.0 MTHM whereas in an actual reactor the amount of fuel salt (that is, volume of fuel salt) would need to be held constant so the amount of heavy metal must be lowered to make room for fission products that are not removed. A single parametric study was conducted in which the mass of fuel salt was held constant for the 300 kW/L scenario. The result was that the amount of heavy metal decreased, and the equilibrium fission product concentration increased from 16.8 wt% to 21 wt%.

ORIGEN2 has two capabilities that are particularly relevant to liquid-fuel MSRs: continuous feed of specified nuclear materials (elements having a particular isotopic vector) and continuous removal of specified elements from the fuel salt at a specified rate. These capabilities are in addition to, or in lieu of, the batch addition of feed materials and batch removal of materials such as discard. ORIGEN2 calculations in this study are performed by assuming continuous radionuclide removal for cleanup processing but batch fresh fuel salt feed and irradiated fuel salt discard. Continuous cleanup processing is assumed to avoid accumulation of insoluble fission products, which would be unreaslistic, particularly for the noble gases. Batch feed and discard are assumed for transparency to facilitate verification of ORIGEN2 results because continuous removal rates are not readily visible in the results just as half-lives and cross sections are not.

The MSRE design involved continuous cleanup processing by sparging with helium to remove noble gases but employed batch feed and irradiated fuel salt removal using a device called a sampler-enricher [Haubenreich and Engle, 1970]. While continuous feed and discard would facilitate continuous reactor operation, it is likely that reactor shutdowns will be required for maintenance.

Reasons for adopting batch feed and discard might include avoiding reactor neutronic or flow perturbations during reactor operation and facilitating safeguards for the plutonium-laden irradiated fuel salt. Additionally, the amount of salt to be fed and discarded is likely to be no more than a few liters per day for a 1000 MWt reactor (depending on the power density) which means designing a low-flow system to feed and remove the salt while keeping it molten. However, to elucidate the impact of modeling salt cleanup processing, feed, and discard as continuous processes, the feed and discard rates in Table 2 5 for the 300 kW/L scenario were included in an ORIGEN2 calculation as continuous events which changed the equilibrium fission product concentration from 16.8 wt% to 16.1 wt%.

3.4.3.2 Discussion

Skewing the fission product elemental composition to favor high-cross-section rare earth elements as discussed in the reactor physics results above increases non-productive neutron absorption as would the use of lower chlorine enrichments. These effects would reduce the breeding ratio to an extent that depends on the power density (higher power density leads to greater reduction in breeding ratio) and the nearness of the fission product concentration to equilibrium (closer to equilibrium leads to greater reduction). The 10% increase in neutron absorption each for higher fission product absorption and lower chlorine enrichment suggests that at equilibrium the breeding ration might be reduced by 0.2 because 10% fewer neutrons are available for breeding due to these two mechanisms in the 300 kW/L scenario. This behavior is to be expected because the plutonium concentration is defined by (a) the nuclear properties (e.g., half-lives, cross sections) of the radionuclides involved, and (b) the feed and discard rates, both of which were held constant at the values for the baseline 10 wt% plutonium concentration in the heavy metal.

The essential absence of impacts to fission product concentration from varying initial plutonium concentration and the minimal impacts from using a fixed amount of fuel salt instead of a fixed amount of heavy metal is unsurprising. This result is obtained because the specific power and irradiation duration are the same in the variant scenarios, which means the total burnup is the same. Using the approximation of 1 MWd of thermal energy requires fission of 1g of an actinide thereby producing 1g of fission products, the outcome is that the total fission product mass produced and the total mass remaining in the fuel salt does not vary with initial plutonium concentration.

ORIGEN2 Modeling of Reference CSFR



Figure 3-3

CSFR fission product element distribution after sustained feed, discard, and cleanup processing compared to distribution after sustained feed and discard but without cleanup processing at 100 FPY

4 SUMMARY AND CONCLUSIONS

Three lines of inquiry were followed in this evaluation of fuel salt sustainability for a chloridebased liquid-fuel MSR:

- What is the equilibrium fission product concentration in the fuel salt, how long does it take to reach equilibrium, and how does this result compare to the results in the preliminary analysis in the fuel salt?
- How does cleanup processing affect the elemental composition of the fuel salt?
- Could the calculated fission product concentrations constrain sustained reuse of discard fuel salt?

4.1 Fully Sustained Salt Reuse

With respect to constraints on CSFR fuel salt sustainability and reuse, the question can be refined to ask whether fission product concentrations could constrain fully sustained salt reuse, in which actinide-rich fuel salt removed from a CSFR is accumulated and used in its entirety to startup subsequent CSFRs. In this scenario, none of the fuel salt is chemically processed to separate and recover species such as U, Pu, or Cl or is managed as waste.

While uncertainties remain, the main condition that favors full sustained salt reuse is to maintain the power density at sufficiently low levels so that fission product concentrations do not build up to levels that might constrain continued CSFR operation. The constraint could be driven by factors related to economics, reactor physics, or the compositional and physicochemical properties of the fuel salt becoming unacceptable.

4.1.1 Economic Factors

The existence of a maximum acceptable power density and, thus, a maximum acceptable equilibrium fission product concentration, would seem to be inevitable because of limits on heat transfer, material properties, etc. The maximum acceptable power density that would not constrain full sustained salt reuse is presently unknown and will require a more detailed CSFR design, for example, a two-dimensional representation, and an analysis based on such a design, and a better understanding of the fuel salt chemistry.

As discussed in the results in Section 3, the 100 kW/L scenario has a maximum fission product concentration of approximately 8 wt %, which is comparable to contemporary light water reactor fuel at discharge. From a reactor physics perspective, it may not be unreasonable to achieve full sustained salt reuse for CSFRs having power densities in the 50 kW/L to100 kW/L range. Similarly, the 300 kW/L case has an equilibrium fission product concentration of about 17 wt % which is not substantially greater that what might be found in SFR core fuel having a burnup of 150 GWd/MTHM. Consequently, full sustained salt reuse might be possible, at least from a

Summary and Conclusions

reactor physics perspective. At 500 kW/L the equilibrium fission product concentration is about 26 wt % and this exceeds what has been postulated even for advanced commercial solid fuels and, thus, raises concerns about feasibility on an intuitive basis.

Accordingly, the chances of fully sustained salt reuse would conceivably be enhanced by lower power densities. Such a scenario would have potentially important advantages such as avoiding the need to reprocess or dispose of unusable spent fuel salt and the ability to provide the initial fuel salt inventory for an expanding CSFR fleet. The lower fission product concentration associated with lower power densities may also have lesser adverse impacts on materials and have more readily manageable chemistry.

However, a low power density CSFR capable of fully sustained salt reuse would be economically disadvantaged. One disadvantage is that if a lower power density is adopted, a larger amount of fuel salt will be needed to produce a given amount of energy. A larger amount of fuel salt means the CSFR equipment sizes and costs will increase, and the cost of the fuel salt itself—fissile material for the initial core and possibly enriched chlorine—will increase. Another disadvantage is that a lower power density leads to longer doubling times which could constrain the growth rate of a CSFR fleet and the economics of the CSFR fleet.

The existence of a limiting high-power density would seem to be inevitable but fully sustained salt reuse at lower power densities does not seem unreasonable. Such an approach might be a practical way to initiate a fully sustained salt reuse fuel cycle and could offer some important advantages like minimizing repository waste; however, it would also entail economic penalties. Economic tradeoff analyses require a more detailed CSFR design for an adequate basis.

4.1.2 Reactor Physics

The foregoing results concerning the reactor physics of the CSFR identify adverse neutronics impacts of fission product buildup in the fuel salt. Initial results suggest that parasitic neutron absorptions in the fission products do not reach levels that would make sustained operation of a CSFR infeasible from a reactor physics perspective a priori although more detailed studies at high power densities and long irradiation times using a more detailed reactor design will be needed to confirm this.

Fully sustained salt reuse may be feasible from a reactor physics perspective and the conditions under which it is feasible are favored by lower fission product concentrations (lower power densities) and use of enriched chlorine. As with the economic factors, evaluating how high is too high awaits an integrated conceptual CSFR design and an analysis thereof.

4.1.3 Physicochemical Property Factors

Given knowledge of the fresh fuel salt feed and power density, the evolving concentration and composition of the fuel salt can be calculated with sufficient accuracy to support an analysis of the impacts of the changing composition on the physicochemical properties of the fuel salt. However, even though the evolving composition can be calculated, there is essentially no information available concerning how the physicochemical properties change in response. Such an assessment requires details of the CSFR design, as well as additional information on fuel salt chemistry.

Whether the evolving fuel salt fission product concentration and composition constitutes an impediment to fully sustained salt reuse is unknowable at this time because there is insufficient information to evaluate the potential impacts of the effects of the changes on fuel salt physical and chemical properties and, in turn, how the changes affect materials degradation, reactor performance, and other considerations.

4.2 Limited Fuel Salt Reuse

If fully sustained salt reuse is ultimately determined to be infeasible or undesirable, CSFRs may be operated in a limited salt reuse mode in which some of the fuel salt discard is managed by means other than reuse. Limited salt reuse may be required for reasons discussed in the previous section or if the CSFR fleet is not expanding rapidly enough to use all of the discarded fuel salt. This possibility leads to considering how to manage the (excess) fuel salt that is not reused. Options that could be considered for managing excess fuel salt are:

- Chemical processing of the fuel salt to recover potentially valuable materials such as uranium, plutonium, and enriched chlorine (if used). This would address the issue of increased fission product concentration but entails additional cost and could raise safeguards and security issues.
- Enhancing cleanup processing to remove more fission product elements (for example, the rare earths). This would reduce the fission product concentration and improve the neutron economy but the acceptability of the physicochemical properties of the remaining fission product mix would still have to be evaluated, and it would add complexity and cost to the CSFR system.
- Treating the fuel to convert it to a chemical state acceptable for conversion to a waste form (if required), immobilize it, and dispose of it as waste.

The decision on whether to pursue fully sustained or limited salt reuse scenarios may ultimately be an economic decision but that decision will need to be underpinned by more complete technical data, for example, salt physicochemical properties and salt interaction with materials, and reactor system analyses (both in-reactor and reactor fleet) leading to candidate CSFR designs that can be compared in terms of costs, risks, and benefits.

Review of the literature suggests that relatively little work has been done on technology to manage excess MSR halide fuel and, based on the information identified in this study, none of the options listed can be shown to be either preferred or infeasible. Identification and evaluation of the status and technology gaps in options for managing excess fuel salt and undertaking R&D to fill the gaps should be considered.

4.3 Conclusions

None of the CSFR model reactors found in the literature were suitable individually to provide the basis for an ORIGEN2 CSFR model to be used for the follow-on evaluation. Therefore, a conceptual model for reference CSFR was developed based on a composite of information drawn from multiple sources. Important limitations and uncertainties associated with the parameterization and application of this model remain.

Summary and Conclusions

Results from ORIGEN2 calculations are consistent with the previous preliminary analysis described in Appendix A.

Fission product concentrations are primarily a function of the average fuel salt burnup– equivalent to the specific power for a fixed irradiation duration—and the discard rate required to maintain the plutonium concentration at an acceptable level which also removes bulk fission products.

- Other ORIGEN2 input parameters are secondary to the average burnup in their effect on equilibrium fission product concentrations.
- Calculated fission product concentrations and elemental distributions are not sensitive to the assumed cleanup processing cycle times because the cycle times are sufficiently short to reflect essentially instantaneous removal of all insoluble elements. This remains the case even if the cycle times are increased substantially by one to two orders of magnitude.

Whether the advantages of fully sustained salt reuse outweigh the disadvantages cannot be ascertained until more detailed and integrated CSFR designs along with better understanding of fuel salt properties and the associated impacts are available.

4.4 Recommendations

- Further work to evaluate the potential impacts of the combination of increasing total fission product concentration in the fuel salt and evolving elemental distribution should be considered at some point. The initial step should be a literature survey leading to gap identification and a R&D plan to obtain the information required to fill the gaps, to support an assessment of the impacts of the increased fission product concentration and evolving and equilibrium composition on materials performance and fuel salt properties. Such an effort would involve a combination of experiments and computation.
- Identification and evaluation of approaches for managing continuous and end-of-life fuel salt discard streams should be considered. Again, the initial step should be a literature search leading to gap identification and an R&D plan to fill the gaps.

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A PRELIMINARY ANALYSIS OF CSFR FUEL SALT SUSTAINABILITY

A simplified, preliminary analysis was performed to calculate the extent to which the concentration of soluble species in the fuel salt could increase and to qualitatively assess the potential impacts on sustained reuse of the fuel salt in chloride-based liquid-fuel MSRs operating on a U-Pu fuel cycle. The analysis was focused on a conceptual chloride salt fast reactor (CSFR) design. CSFR designs are being pursued by multiple private-sector developers and are of interest because they produce a substantial excess of neutrons that can be used to generate surplus fuel inventories to allow for startup of subsequent CSFRs. Calculated CSFR breeding ratios are reported to range up to about 1.5 [Nelson et al., 1967], whereas breeding ratios for a fluoride-salt fast reactor operating on a Th-U cycle is perhaps a bit over 1.1 [Merle-Lucotte, 2013] and the breeding ratio of a fluoride-salt thermal reactor operating on a Th-U fuel cycle have the disadvantage of requiring fuel salt chemical processing to achieve desired breeding ratios. A fluoride-based liquid-fuel thermal reactor operating on a U-Pu fuel cycle has a breeding ratio less than 1.0.

The preliminary analysis is based on simplified equation-based spreadsheet calculations and information in the publicly domain from available literature. More complex reactor physics considerations are beyond the study scope. The preliminary analysis and results are summarized below. Full study results are documented in an internal EPRI letter report [EPRI, 2021].⁴

A.1 Description of the Preliminary Analysis

A postulated fuel reuse scenario is described as follows:

- After irradiation, the fissile/fissionable material concentration (primarily isotopes of plutonium) in a CSFR will increase to the point that reactivity control mechanisms such as control rods do not have adequate worth for managing reactor reactivity control within an acceptable margin of safety. Consequently, fuel salt is bled/removed from the system to maintain fissile concentrations at an acceptable level.
- Concurrent with fuel salt removal, fresh fertile material comprising depleted uranium and the chloride carrier salt (some combination of NaCl, KCl, and/or MgCl₂) is added to compensate for the inventory of fuel salt removed.
- The fuel salt removed from the CSFR is stored in a criticality-controlled vessel.
- Once enough discarded fuel salt has been accumulated, it can be used as the initial fuel salt inventory for a second CSFR. This fleet expansion process can be repeated as long as additional reactors are needed provided that the fuel salt is suitable for reuse.

⁴ The letter report was initially issued in 2019 and was revised in 2021.

Preliminary Analysis of CSFR Fuel Salt Sustainability

If additional reactors are not needed at some point or if the fuel salt becomes unsuitable for reuse in subsequent reactors for any reason, the fuel salt would require management as irradiated nuclear fuel via disposal or chemical separations.

If sustained reuse is feasible, it would have two major advantages. First, it would not require chemical separations often associated with thermal-spectrum MSRs and other reactors involving actinide recycling. Second, activities and volumes of resulting radioactive waste streams requiring management and eventual disposal would be limited mainly to stabilized forms of insoluble fission product gases and solids and contaminated equipment.

The analysis was performed by first developing a simple conceptual model of a single-reactor CSFR fuel cycle using a uranium-plutonium-sodium chloride fuel salt. Next, an equation for calculating the average fission product concentration at equilibrium using typical reactor design parameters (power density, fuel composition, fissile material doubling time, etc.) was developed. Each parameter in the equation was then quantified as a single value or a range that is based information found in the open literature. The equation was then exercised over the parameter ranges to yield a matrix of equilibrium (maximum) fission product concentrations in the fuel salt for three scenarios:

- Scenario 1: Bulk fuel salt removal to maintain acceptable fissile material concentrations in the CSFR.
- Scenario 2: Scenario 1 plus removal of insoluble fission products by continuous fuel salt cleanup processing.
- Scenario 3: Scenario 2 plus limiting the range of certain parameter values because they are related to others by algebraic equations—that is, they are not independent (e.g., the doubling time depends on the specific power being produced by the fuel salt).

Scenario 3 was then extended to include calculation of the rate at which the equilibrium fission product concentration would be approached for a few combinations of parameters.

A.2 Summary

Preliminary analysis of fuel salt sustainability for a conceptual CSFR indicates equilibrium fission product concentrations could reach levels that would need to be addressed in the design, operation, and nuclear fuel cycle of a chloride-based liquid-fuel fast MSR. The results suggest that fission product removal rates for typical solubility-based fuel salt polishing processes and bulk salt discard (for activity control) may not be fully adequate for sustaining reactor performance or allowing for startup of subsequent reactors from accumulated inventories of discarded fuel salt. Therefore, additional means to remove fission products may be needed or desirable.

In the preliminary analysis, approximately 50% of fission products remain soluble in the fuel salt and are not removed by typical fuel salt cleanup processes alone, for example, via sparging of gases and filtration of solids. Soluble fission products are removed by discard to maintain reactivity control at a rate depending on the doubling time. Additional fission product removal options include:

- Increased removal (discard) of bulk fuel salt, requiring additions of makeup fissile and fertile material.
- Enhanced cleanup processing of the fuel salt to remove soluble and/or semi-volatile fission products.
- Separating and recycling of actinides and removal and management of fission products as a waste stream.

Based on preliminary results, design opportunities to reduce fission product concentrations appear limited. In terms of fission product concentration in the fuel salt, the only variable that having a substantial impact on the equilibrium concentration in the preliminary analysis is the breeding ratio. The important controllable design parameters for increasing the breeding ratio are decreasing parasitic neutron capture and reducing neutron leakage via use of larger cores, fertile blankets, and/or neutron reflectors.

Recovery and recycle of enriched chlorine may be important and cost-effective. Increasing breeding ratios and reducing long-lived, mobile ³⁶Cl production favor use of chlorine enriched in ³⁷Cl. As enriched chlorine is expected to be costly, at least initially, the extent to which ³⁷Cl-encrich fuel salt can be reused represents an attractive option for preserving that investment, as with other isotopically enriched chemicals used in industry like boron.

Fuel salt isotopic compositions will be complex and fission product concentrations will increase with time until equilibrium is achieved. Bulk fuel salt removal to maintain reactivity control or for external fuel salt separations or disposal do not alter the isotopic and chemical composition of the remaining fuel salt. Likewise, while typical fuel salt polishing results in low concentrations of insoluble fission products, soluble fission products are not affected, and their concentrations will continue to increase.

Accounting for enhanced fuel cleanup processing is beyond the scope of the preliminary analysis as the time-dependent removal of individual fission product elements during cleanup processing cannot be calculated in a simple spreadsheet. The increasing concentration of soluble fission products poses a number of potential challenges to reactor design that include:

- Changing physical properties related to the thermal-hydraulic characteristics of the fuel salt;
- Increased parasitic neutron absorption and the consequent decrease in breeding ratio;
- The need to select and qualify salt-wetted materials to withstand interaction with an evolving mix of fission products;
- The need for understanding fuel salt performance over a range of fission product concentrations and compositions; and
- Designing systems to control the chemistry of the fuel salt without unintended adverse consequences such as precipitation of solids that become insoluble in the evolving fuel salt composition.

Preliminary Analysis of CSFR Fuel Salt Sustainability

Simplified calculations provided useful insights into the potential magnitude of impacts from fission product accumulation on sustainability of fuel salt in a conceptual liquid-fuel MSR. The next logical step in the evaluation of fuel salt sustainability is to obtain a more complete understanding of how fission product concentrations and compositions evolve and evaluate potential options for mitigation measures.

A.3 References

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Merle, 2017	E. Merle. <i>Concept of Molten Salt Fast Reactor</i> . Presented to GENIV International Forum Molten Salt Reactor Workshop, Paul Scherrer Institute, Villigen, Switzerland: May 23, 2017.
Merle-Lucotte, 2013	E. Merle-Lucotte. <i>Introduction to the Physics of the Molten Salt Fast Reactor</i> . Present to Thorium Energy Conference 2013 (ThEC13) – CERN, Geneva, Switzerland. October 27-31, 2013.

B SUMMARY OF LITERATURE SURVEY FOR LIQUID-FUEL MSR CYCLE TIMES

This appendix summarizes the results of a literature search for MSR cycle times. Tables B-1 through B-4 define the labels, initialisms, and references used in the consolidation of MSR cycle time information presented in Table B-5.

Table B-1Legend for the headers in Table B-5

Header Label	Definition
Z	Atomic number
Elem	Element symbol
Elem Group	Group of elements that have the same cycle time. These are defined in Table B-2
Cycle times for fluoride salt MSRs	Information in the columns below is for MSRs having fluoride fuel salt and is taken from the numbered reference which is identified in Table B-2. The reactor spectrum is also identified.
Cycle times for chloride salt MSRs	Information in the one column below is for MSRs having chloride fuel salt and is taken from the numbered reference which is identified in Table B-2. The reactor spectrum is also identified.
Rem mech	The suggested mechanism(s) by which each element is removed from the fuel salt for fluoride and chloride salts. These differ because CSFRs have fewer removal mechanisms (i.e., no reprocessing). These are defined in Table B-4

Summary of Literature Survey for Liquid-Fuel MSR Cycle Times

Table	B-2				
Refere	ences	cited	in	Table	B-5

Reference No.	Reference
1	D. E. Ferguson, K.B. Brown, R.G. Wymer, R.E. Blanco, M.E. Whatley, H.E. Goeller, and R.E. Brooksbank. <i>Chemical Technology Division Annual Progress Report for Period Ending May 31, 1969.</i> Oak Ridge National Laboratory, Oak Ridge, TN: ORNL-4422: October 1969. ORNL-4422.
2	R. C. Robertson (Ed.). <i>Conceptual Design Study of a Single-Fluid Molten-Salt Breeder Reactor</i> . Oak Ridge National Laboratory, Oak Ridge, TN: June 1971. ORNL-4541.
3	M. W. Rosenthal, R. B. Briggs, and P. N. Haubenreich. <i>Molten Salt Reactor Program</i> <i>Semiannual Progress Report for Period Ending August 31,1971.</i> February 1972. ORNL- 4728.
4	J. R. Engel, W.R. Grimes, W.A. Rhoades, and J.F. Dearing. Molten-Salt Reactors for Efficient Nuclear Fuel Utilization without Plutonium Separation. Oak Ridge National Laboratory, Oak Ridge, TN: August 1978. ORNL/TM-6413.
5	K. Nagy, J.L. Kloosterman, D. Lathouwers, and T. van der Hagen. Parametric Studies on the Fuel Salt Composition in Thermal Molten Salt Breeder Reactors. <i>International</i> <i>Conference on the Physics of Reactors "Nuclear Power: A Sustainable Resource"</i> Casino-Kursaal Conference Center, Interlaken, Switzerland: September 14-19, 2008.
6	J. J. Powers, A. Worrall, J.C. Gehin, T.J. Harrison, and E.E. Sunny. Parametric Analyses of Single-Zone Thorium-Fueled Molten Salt Reactor Fuel Cycle Options. Proceedings of Global 2013, Salt Lake City, UT: September 29-October 3, 2013.
7	V. Ignatiev, O. Feynberg, I Gnidoi, and A. Merzlyakov. Molten Salt Actinide Recycler and Transforming System without and with Th–U Support: Fuel Cycle Flexibility and Key Material Properties. <i>Annals of Nuclear Energy</i> 64 , 408–420. February 2014. DOI:10.1016/j.anucene.2013.09.004
8	J. C. Gehin and J. J. Powers. Liquid Fuel Molten Salt Reactors for Thorium Utilization. <i>Nuclear Technology</i> 194 , 152–161. May 2016.
9	B. R. Betzler, F. Heidet, B. Feng, C. Rabiti, T. Sofu, and N.R. Brown. Modeling and Simulation Functional Needs for Molten Salt Reactor Licensing. <i>Nuclear Engineering and Design</i> 355 , 110308. August 28, 2019.
10	E. Davidson, B. Betzler, R. Gregg, and A. Worrall. Modeling a Fast Spectrum Molten Salt Reactor in a Systems Dynamics Fuel Cycles Code. Annals of Nuclear Energy 133 , 409-424. 2019.
Initialism	Element Group Definition
------------	---
AM	Alkali metal
AE	Alkaline earth
СР	Corrosion product solids
DV	Di-valent rare earth
HAL	Halogen
NG	Noble gas
NM	Noble metal solids
SNM	Semi-noble metal solids
NMC	Element is both a noble metal and a corrosion product
TVA	Tetravalent actinide
TRIA	Trivalent actinide
TRIR	Trivalent rare earth

Table B-3Definition of element group initialisms in Table B-5

Table B-4Definition of removal mechanism initialisms in Table B-5

Initialism	Removal Mechanism Definition
D	Discard of bulk fuel salt in batch mode or continuously, or when the salt is no longer useful
CP	Cleanup processing that removes insoluble gases in an off-gas system
CPD	Cleanup processing that removes insoluble solids by deposition on internal surfaces and/or filtration
RW	Waste from reprocessing of the fuel salt (fluoride fuel salt only) using pyroprocessing techniques
DRV	Volatilization during reprocessing of the fuel salt as part of fluorination (fluoride fuel salt only)

Summary of Literature Survey for Liquid-Fuel MSR Cycle Times

Table B-5

Results of literature search for MSR cycle times. Labels, initialisms, and references are defined in Tables B-2 to B-5 above.

													Cycle times for		
														chloi	ride salt
						Cycle	times for	fluoride s	alt MSRs	, seconds				MSRs.	seconds
Z	Elem	Elem	Rem	Ref. 1	Ref. 2	Refs. 3, 4	Ref. 5	Ref. 5	Ref. 5	Ref. 6	Ref. 7	Ref. 8	Ref. 9	Rem	Ref. 10
		Group	Mech	Thermal	Thermal	Thermal	Thermal	Thermal	Thermal	Thermal	Thermal	Thermal	Spectral	mech	Fast
				IVISK	IVISK	IVISK	IVISK Fast Cycle	NISK	NISK	IVISK	IVISK	IVISK	SHIT		IVISK
							Time	Cycle	Cycle				IVISIA		
							·····e	Time,	Time,						
								Case 1	Case 2						
3	Li	AM	D	2.59E+08	2.97E+08	8.64E+05				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
11	Na	AM	D	2.59E+08	2.97E+08	8.64E+05				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
19	к	AM	D	2.59E+08	2.97E+08	8.64E+05				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
37	Rb	AM	D	2.59E+08	2.97E+08	8.64E+05				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
55	Cs	AM	D	2.59E+08	2.97E+08	8.64E+05				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
87	Fr	AM	D	2.59E+08	2.97E+08	8.64E+05				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
4	Ве	AE	D	2.59E+08	2.97E+08	2.16E+06				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
12	Mg	AE	D	2.59E+08	2.97E+08	2.16E+06				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
20	Ca	AE	D	2.59E+08	2.97E+08	2.16E+06				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
38	Sr	AE	D	2.59E+08	2.97E+08	2.16E+06				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
56	Ва	AE	D	2.59E+08	2.97E+08	2.16E+06				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
88	Ra	AE	D	2.59E+08	2.97E+08	2.16E+06				2.97E+08	2.37E+08	2.97E+08	2.97E+08	D	2.59E+05
24	Cr	СР	CPD	5.00E+01	2.97E+08	8.64E+05			9.46E+07		6.31E+07		2.97E+08	CPD	2.59E+05
26	Fe	СР	CPD	5.00E+01	2.97E+08	8.64E+05			9.46E+07		6.31E+07		2.97E+08	CPD	2.59E+05
28	Ni	СР	CPD	5.00E+01	2.97E+08	8.64E+05			9.46E+07		6.31E+07		2.97E+08	CPD	2.59E+05
62	Sm	DV	RW	1.94E+07	4.32E+06	2.16E+06	4.32E+07	3.15E+07		4.32E+07	6.31E+07	4.32E+07	4.32E+06	D	2.59E+05
63	Eu	DV	RW	1.94E+07	4.32E+07	2.16E+06	4.32E+07	3.15E+07	9.46E+07	4.32E+07	6.31E+07	4.32E+07	4.32E+07	D	2.59E+05
9	F	HAL	RW	2.59E+08	2.97E+08	2.16E+06	5.18E+06			2.97E+08		2.97E+08	2.97E+08	D	2.59E+05
17	Cl	HAL	DRV	2.59E+08	2.97E+08	2.16E+06	5.18E+06			2.97E+08		2.97E+08	2.97E+08	D	2.59E+05
35	Br	HAL	DRV	4.32E+06	5.18E+06	8.64E+05	5.18E+06			5.18E+06		5.18E+06	5.18E+06	D	2.59E+05
53	I	HAL	DRV	4.32E+06	5.18E+06	8.64E+05	5.18E+06			5.18E+06		5.18E+06	5.18E+06	D	2.59E+05
85	At	HAL	DRV	2.59E+08	2.97E+08	2.16E+06	5.18E+06			2.97E+08		2.97E+08	2.97E+08	D	2.59E+05
2	He	NG	СР	5.00E+01	2.00E+01	5.00E+01	5.00E+01	5.00E+01	5.00E+01	2.00E+01	5.00E+01	2.00E+01	2.00E+01	СР	2.59E+05
10	Ne	NG	СР	5.00E+01	2.00E+01	5.00E+01	5.00E+01	5.00E+01	5.00E+01	2.00E+01	5.00E+01	2.00E+01	2.00E+01	СР	2.59E+05
18	Ar	NG	СР	5.00E+01	2.00E+01	5.00E+01	5.00E+01	5.00E+01	5.00E+01	2.00E+01	5.00E+01	2.00E+01	2.00E+01	СР	2.59E+05
36	Kr	NG	СР	5.00E+01	2.00E+01	5.00E+01	5.00E+01	5.00E+01	5.00E+01	2.00E+01	5.00E+01	2.00E+01	2.00E+01	СР	2.59E+05
54	Xe	NG	СР	5.00E+01	2.00E+01	5.00E+01	5.00E+01	5.00E+01	5.00E+01	2.00E+01	5.00E+01	2.00E+01	2.00E+01	СР	2.59E+05
86	Rn	NG	СР	5.00E+01	2.00E+01	5.00E+01	5.00E+01	5.00E+01	5.00E+01	2.00E+01	5.00E+01	2.00E+01	2.00E+01	СР	2.59E+05
41	Nb	NM	CPD	1.10E+02	2.00E+01	8.64E+03	2.00E+01	8.64E+03	8.64E+03	2.00E+01	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
43	Тс	NM	CPD	1.10E+02	2.00E+01	8.64E+03	2.00E+01	8.64E+03	8.64E+03	2.00E+01	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
44	Ru	NM	CPD	1.10E+02	2.00E+01	8.64E+03	2.00E+01	8.64E+03	8.64E+03	2.00E+01	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
45	Rh	NM	CPD	1.10E+02	2.00E+01	8.64E+03	2.00E+01	8.64E+03	8.64E+03	2.00E+01	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
46	Pd	NM	CPD	1.10E+02	2.00E+01	8.64E+03	2.00E+01	8.64E+03	8.64E+03	2.00E+01	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
47	Ag	NM	CPD	1.10E+02	2.00E+01	8.64E+03	2.00E+01	8.64E+03	8.64E+03	2.00E+01	5.00E+01	2.00E+01	2.00E+01	CPD	2.59E+05

Table B-5 (continued) Results of literature search for MSR cycle times (continued). Labels, initialisms, and references are defined in Tables B-2 to B-5 above.

														Cycle times for	
														chlor	ide salt
						Cycle t	imes for	fluoride s	alt MSRs,	seconds				MSRs,	seconds
Ζ	Elem	Elem	Rem	Ref. 1	Ref. 2	Refs. 3, 4	Ref. 5	Ref. 5	Ref. 5	Ref. 6	Ref. 7	Ref. 8	Ref. 9	Rem	Ref. 10
		Group	Mech	Thermal	Thermal	Thermal	Thermal	Thermal	Thermal	Thermal	Thermal	Thermal	Spectral	mech	Fast
				MSR	MSR	MSR	MSR Fact Curls	MSR	MSR	MSR	MSR	MSR	Shift		MSR
							Time	SIOW	SIOW				IVISK		
							mile	Time,	Time,						
								Case 1	Case 2						
48	Cd	NM	CPD	1.10E+02	1.73E+07	8.64E+03	1.73E+07	8.64E+03	8.64E+03	1.73E+07	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
49	In	NM	CPD	1.10E+02	1.73E+07	8.64E+03	1.73E+07	8.64E+03	8.64E+03	1.73E+07	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
50	Sn	NM	CPD	1.10E+02	1.73E+07	8.64E+03	1.73E+07	8.64E+03	8.64E+03	1.73E+07	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
51	Sb	NM	CPD	1.10E+02	2.00E+01	8.64E+03	2.00E+01	8.64E+03	8.64E+03	2.00E+01	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
52	Те	NM	CPD	1.10E+02	2.00E+01	8.64E+03	2.00E+01	8.64E+03	8.64E+03	2.00E+01	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
42	Mo	NMC	CPD	1.10E+02	2.00E+01	8.64E+03	2.00E+01	8.64E+03	8.64E+03	2.00E+01	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
34	Se	SNM	CPD	1.73E+07	2.00E+01	8.64E+03		8.64E+03	2.00E+01	2.00E+01	1.08E+04	2.00E+01	2.00E+01	CPD	2.59E+05
30	Zn	SNM	CPD	1.73E+07	1.73E+07	8.64E+03		8.64E+03		1.73E+07	1.08E+04	1.73E+07	1.73E+07	CPD	2.59E+05
31	Ga	SNM	CPD	1.73E+07	1.73E+07	8.64E+03		8.64E+03		1.73E+07	1.08E+04	1.73E+07	1.73E+07	CPD	2.59E+05
32	Ge	SNM	CPD	1.73E+07	1.73E+07	8.64E+03		8.64E+03		1.73E+07	1.08E+04	1.73E+07	1.73E+07	CPD	2.59E+05
33	As	SNM	CPD	1.73E+07	1.73E+07	8.64E+03		8.64E+03		1.73E+07	1.08E+04	1.73E+07	1.73E+07	CPD	2.59E+05
89	Ac	TVA	D	2.59E+08	2.97E+08	2.16E+06					9.46E+08			D	2.59E+05
90	Th	TVA	D	2.59E+08	2.97E+08	4.73E+08				2.97E+08	9.46E+08			D	2.59E+05
91	Ра	TVA	D	2.59E+08	2.59E+05	4.73E+08				2.97E+08	9.46E+08			D	2.59E+05
92	U	TVA	D	2.59E+08	2.97E+08	4.73E+08				2.97E+08	9.46E+08			D	2.59E+05
93	Np	TRIA	RW	2.59E+08	5.05E+08	2.16E+06					4.32E+06			D	2.59E+05
94	Pu	TRIA	RW	2.59E+08	5.05E+08	2.16E+06					4.32E+06			D	2.48E+08
95	Am	TRIA	RW	2.59E+08	5.05E+08	2.16E+06					4.32E+06			D	2.59E+05
96	Cm	TRIA	RW	2.59E+08	5.05E+08	2.16E+06					4.32E+06			D	2.59E+05
39	Y	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
40	Zr	TRIR	RW	1.73E+07	1.73E+07	8.64E+05	1.73E+07	3.15E+07	9.46E+07	1.73E+07	4.32E+06	1.73E+07	4.32E+06	D	2.59E+05
57	La	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
58	Ce	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
59	Pr	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
60	Nd	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
61	Pm	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
64	Gd	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
65	Tb	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
66	Dy	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
67	Но	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05
68	Er	TRIR	RW	4.32E+06	4.32E+06	2.16E+06	4.32E+06	3.15E+07	9.46E+07	4.32E+06	4.32E+06		4.32E+06	D	2.59E+05

C ORIGEN2 COMMAND INPUT AND UPDATED CROSS SECTIONS

```
-1
-1
RDA
      ******* FUEL SALT SUSTAINABILITY ******
RDA
RDA
      This version uses the Davidson fuel formulation
      19 0.01 20 0.01
                         21 0.01 22 0.01
CUT
                                                 -1
      0 0 0
LIP
      0 1 2 3 321 322 323 9 50 0 1 0
LIB
              101 102 103 10
RDA
      PHO
        ONE METRIC TON OF INITIAL HEAVY METAL IN A CSFR
BAS
RDA
        READ FUEL SALT COMPOSITION EXCEPT U AND PU; AND PUT IN -15
RDA
        THE C1 AND NA ARE THE AMOUNT REQUIRED PER MTIHM OF DU
RDA
        IN THE FEED
                                                                    Source:
        -15 1 -1 -1 1 1
TNP
                                                                    Table 2-2
       READ DU FEED COMPOSITION AND PUT IN -16
RDA
INP
        -16 1 -1 -1 1 1
                                                                      These
RDA
       READ PU FEED COMPOSITION AND PUT IN -17
                                                                    are innut
INP
        -17 1 -1 -1 1 1
RDA
     READ CONTINUOUS RADIONUCLIDE REMOVAL RATES
       -18 0 -1 3 1 1
INP
                                                              Source: Table 2-4
RDA
       ZERO VECTORS 12 AND 1; USE AS SCRATCH
MOV
       1 12 0 0.0
RDA
       COMPOSE INITIAL FUEL SALT
MOV
       -15 1 0 1.0
                                                 An irradiation step follows each IRP
ADD
       -16 1 0 0.9
                                                 command. There are five irradiation
ADD
        -17
            1 0 0.10
                                                    segments each for 20y in 11
       COMPOSE DU-Na CHLORIDE FEED SALT
RDA
                                                        irradiation steps.
ADD
        -15 -16 0 1.0
TIT
        IRRADIATION OF CSFR FUEL SALT
BUP
RDA
      ****** THIS BEGINS THE FIRST IRRADIATION SEGMENT *******
          1.0 121.6 -1 2 5 2
IRP
RDA
          ZERO OUT VECTOR 12 ONCE
                                                                     Specific
 MOV
          1 12 0 0.0
                                                                     Power
           -16 12 0 0.2891 🔻
 MOV
                                                                     Source:
  ADD
           2 12 0 0.8050
IRP
          2.0 121.6
                          12 3
                                5 0
                                                                    Table 2-5
           -16 12 0 0.2891
  MOV
  ADD
           3 12 0 0.8050
RDA MOV
             1 12 0 0.0
                                                                     DU Salt
TRP
          4.0 121.6
                          12 4 5 0
                                                                    Feed Rate
 MOV
           -16 12 0 0.2891
                                                 Discard
  ADD
           4 12 0 0.8050
                                                                     Source:
                                                  Rate
             1 12 0 0.0
RDA MOV
                                                                    Table 2-5
                                                 Source:
IRP
          6.0 121.6
                          12 5 5 0
                                                Table 2-5
           -16 12 0 0.2891
 MOV
```

ORIGEN2 Command Input and Updated Cross Sections

```
5 12 0 0.8050
   ADD
            1 12 0 0.0
 RDA MOV
           8.0 121.6
 IRP
                       12 6 5 0
           -16 12 0 0.2891
   MOV
           6 12 0 0.8050
   ADD
 RDA OUT
          12 1 -1 0
 RDA MOV
             1 12 0 0.0
 IRP
          10.0 121.6
                       12 7 5 0
           -16 12 0 0.2891
   MOV
   ADD
            7 12 0 0.8050
 RDA MOV
             1 12 0 0.0
 IRP
        12.0 121.6
                       12 8 5 0
   MOV
           -16 12 0 0.2891
   ADD
            8 12 0 0.8050
 RDA MOV
          1 12 0 0.0
        14.0 121.6
 IRP
                       12 9 5 0
   MOV
           -16 12 0 0.2891
   ADD
           9 12 0 0.8050
 RDA MOV 1 12 0 0.0
IRP 16.0 121.6
                     12 10 5 0
           -16 12 0 0.2891
   MOV
           10 12 0 0.8050
   ADD
 RDA MOV
            1 12 0 0.0
      18.0 121.6
IRP
                       12 11 5 0
 RDA TEMPORARILY SAVE OUTPUT VECTOR 1 TO STORAGE VECTOR 18
   MOV 1 -19 0 1.0
   MOV
           -16 1 0 0.2891
   ADD
           11 1 0 0.8050
IRP
         20.0 121.6
                        1 12 5 0
 RDA MOVE OUTPUT VECTOR 1 CONTENTS BACK TO THAT VECTOR
   MOV -19 1 0 1.0
 BUP
 HED
         1 EOC FUEL
 OPTL 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OPTA 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OPTF 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OUT
        12 1 -1 0
RDA *****SAVE RESULTS AT SELECTED TIMES
 MOV 1 -20 0 1.0
 MOV 6 -21 0 1.0
 MOV 12 -22 0 1.0
RDA SAVE COMPOSITION AT LAST TIME ON THIS OUTPUT PAGE
 MOV 12 -14 0 1.0
 STP 1
 1 110230 220130.0 170350 0.0 170370 820306.0 0 0.0
 0
                                                                 These are the input
 2 922340 0.0
                922350 2000.0 922380 998000.0 0 0.0
                                                                material compositions
 0
 2
   942360 0.0
                 942370 0.0
                              942380 760.0
                                            942390 942890.0
    942400 53060.0 942410 2420.0 942420 870.0
                                               952410 0.0
 2
 0
                  0.05
0.05 6 0.05
              12
                       4
                                               These are the cleanup
02 10 18 36
              54
                  86
                                                  processing
41 42 43 44
              45
                  46 47 48 49 50 51
                                        52
                                               radioelement removal
30
  31
      32 33
                                               rates from
-1
 -1
```

```
-1
     ****** THIS BEGINS THE SECOND IRRADIATION SEGMENT *******
RDA
     19 0.01 20 0.01 21 0.01 22 0.01
 CUT
                                             -1
 LIP
      0 0 0
      0 1 2 3 321 322 323 9 50 0 1 0
 LIB
 RDA
     PHO 101 102 103 10
 BAS
        ONE METRIC TON OF INITIAL HEAVY METAL IN A CSFR
 RDA READ CONTINUOUS RADIONUCLIDE REMOVAL RATES
      -18 0 -1 3 1 1
 INP
        -14 1 0 1.0
 MOV
 TIT
        IRRADIATION OF CSFR FUEL SALT
 BUP
   MOV
           -16 12 0 0.2891
   ADD
          1 12 0 0.8050
 IRP
          22.0 121.6 12 2 5 0
           -16 12 0 0.2891
   MOV
   ADD
           2 12 0 0.8050
 IRP
         24.0 121.6
                         12 3 5 0
           -16 12 0 0.2891
   MOV
   ADD
           3 12 0 0.8050
 IRP
          26.0 121.6
                      12 4 5 0
           -16 12 0 0.2891
   MOV
           4 12 0 0.8050
   ADD
          28.0 121.6
                      12 5 5 0
 IRP
   MOV
           -16 12 0 0.2891
   ADD
           5 12 0 0.8050
 IRP
          30.0 121.6
                         12 6 5 0
           -16 12 0 0.2891
   MOV
   ADD
           6 12 0 0.8050
          32.0 121.6 12 7 5 0
 IRP
   MOV
           -16 12 0 0.2891
   ADD
           7 12 0 0.8050
 IRP
         34.0 121.6
                         12 8 5 0
   MOV
           -16 12 0 0.2891
           8 12 0 0.8050
   ADD
        36.0 121.6 12 9 5 0
IRP
           -16 12 0 0.2891
   MOV
   ADD
           9 12 0 0.8050
IRP
       38.0 121.6
                   12 10 5 0
          -16 12 0 0.2891
   MOV
   ADD
          10 12 0 0.8050
       40.0 121.6
IRP
                       12 11 5 0
   MOV 1 -19 0 1.0
           -16 1 0 0.2891
   MOV
   ADD
          11 1 0 0.8050
IRP
        42.0 121.6
                       1 12 5 0
RDA MOVE OUTPUT VECTOR 1 CONTENTS BACK TO THAT VECTOR
   MOV -19 1 0 1.0
 BUP
 OPTL 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OPTA 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OPTF 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OUT
        12 1 -1 0
RDA *****SAVE RESULTS AT SELECTED TIMES
 MOV 4 -23 0 1.0
 MOV 8 -24 0 1.0
 MOV 12 -25 0 1.0
```

ORIGEN2 Command Input and Updated Cross Sections

```
RDA SAVE COMPOSITION AT LAST TIME IN THIS SEGMENT
 MOV 12 -14 0 1.0
 STP 1
0.05 6
       0.05 12 0.05 4
02 10 18 36 54 86
41 42 43 44 45 46 47 48 49 50 51 52
30 31 32 33
-1
 -1
 -1
RDA
      ****** THIS BEGINS THE THIRD IRRADIATION SEGMENT *******
 BAS
        ONE METRIC TON OF INITIAL HEAVY METAL IN A CSFR
 RDA
       COPY COMPOSITION AT LAST TIME FROM STORAGE VECTOR
 RDA READ CONTINUOUS RADIONUCLIDE REMOVAL RATES
      -18 0 -1 3 1 1
 INP
        -14 1 0 1.0
 MOV
 TIT
        IRRADIATION OF CSFR FUEL SALT
 BUP
           -16 12 0 0.2891
   MOV
           1 12 0 0.8050
   ADD
          44.0 121.6
 TRP
                          12 2 5 0
           -16 12 0 0.2891
   MOV
           2 12 0 0.8050
   ADD
         46.0 121.6
 IRP
                        12 3 5 0
   MOV
           -16 12 0 0.2891
   ADD
           3 12 0 0.8050
 IRP
          48.0 121.6
                         12 4 5 0
   MOV
           -16 12 0 0.2891
   ADD
           4 12 0 0.8050
          50.0 121.6 12 5 5 0
 IRP
   MOV
           -16 12 0 0.2891
   ADD
           5 12 0 0.8050
          52.0 121.6
 IRP
                          12 6 5 0
   MOV
           -16 12 0 0.2891
           6 12 0 0.8050
   ADD
          54.0 121.6
                          12 7 5 0
 IRP
           -16 12 0 0.2891
   MOV
   ADD
           7 12 0 0.8050
 IRP
         56.0 121.6
                      12 8 5 0
   MOV
           -16 12 0 0.2891
   ADD
          8 12 0 0.8050
        58.0 121.6
                        12 9 5 0
IRP
           -16 12 0 0.2891
   MOV
           9 12 0 0.8050
   ADD
IRP
        60.0 121.6
                      12 10 5 0
   MOV
           -16 12 0 0.2891
   ADD
           10 12 0 0.8050
IRP
       62.0 121.6
                       12 11 5 0
   MO V 1 -19 0 1.0
           -16 1 0 0.2891
   MOV
   ADD
           11 1 0 0.8050
IRP
        64.0 121.6 1 12 5 0
RDA MOVE OUTPUT VECTOR 1 CONTENTS BACK TO THAT VECTOR
  MOV -19 1 0 1.0
 BUP
 OPTL 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OPTA 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
```

OPTF 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8 OUT 12 1 -1 0 RDA *****SAVE RESULTS AT SELECTED TIMES MOV 6 -26 0 1.0 MOV 12 -27 0 1.0 RDA SAVE COMPOSITION AT LAST TIME IN THIS SEGMENT MOV 12 -14 0 1.0 STP 1 0.05 6 0.05 12 0.05 4 02 10 18 36 54 86 41 42 43 44 45 46 47 48 49 50 51 52 30 31 32 33 -1 -1 -1 RDA ****** THIS BEGINS THE FOURTH IRRADIATION SEGMENT ******* BAS ONE METRIC TON OF INITIAL HEAVY METAL IN A CSFR RDA COPY COMPOSITION AT LAST TIME FROM STORAGE VECTOR RDA READ CONTINUOUS RADIONUCLIDE REMOVAL RATES INP-18 0 -1 3 1 1 MOV -14 1 0 1.0 IRRADIATION OF CSFR FUEL SALT TIT BUP MOV -16 12 0 0.2891 ADD 1 12 0 0.8050 66.0 121.6 IRP 12 2 5 0 MOV -16 12 0 0.2891 2 12 0 0.8050 ADD 68.0 121.6 IRP 12 3 5 0 -16 12 0 0.2891 MOV ADD 3 12 0 0.8050 70.0 121.6 IRP 12 4 5 0 MOV -16 12 0 0.2891 ADD 4 12 0 0.8050 IRP 72.0 121.6 12 5 5 0 -16 12 0 0.2891 MOV 5 12 0 0.8050 ADD 74.0 121.6 IRP 12 6 5 0 MOV -16 12 0 0.2891 ADD 6 12 0 0.8050 IRP 76.0 121.6 12 7 5 0 MOV -16 12 0 0.2891 7 12 0 0.8050 ADD 78.0 121.6 IRP 12 8 5 0 MOV -16 12 0 0.2891 ADD 8 12 0 0.8050 IRP 80.0 121.6 12 9 5 0 MOV -16 12 0 0.2891 ADD 9 12 0 0.8050 82.0 121.6 IRP 12 10 5 0 -16 12 0 0.2891 MOV 10 12 0 0.8050 ADD 84.0 121.6 IRP 12 11 5 0 MOV 1 -19 0 1.0 MOV -16 1 0 0.2891 ADD 11 1 0 0.8050 IRP 86.0 121.6 1 12 5 0

```
RDA MOVE OUTPUT VECTOR 1 CONTENTS BACK TO THAT VECTOR
   MOV -19 1 0 1.0
 BUP
 OPTL 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OPTA 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OPTF 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OUT
         12 1 -1 0
RDA *****SAVE RESULTS AT SELECTED TIMES
 MOV 6 -28 0 1.0
 MOV 12 -29 0 1.0
RDA MOV 10 -23 0 1.0
RDA SAVE COMPOSITION AT LAST TIME IN THIS SEGMENT
 MOV 12 -14 0 1.0
 STP 1
       0.05 12 0.05 4
0.05 6
02 10 18 36
              54
                  86
41 42
      43
          44
              45
                  46 47 48 49 50 51 52
30 31 32 33
-1
 -1
 -1
       ****** THIS BEGINS THE FIFTH IRRADIATION SEGMENT *******
 RDA
 BAS
        ONE METRIC TON OF INITIAL HEAVY METAL IN A CSFR
        COPY COMPOSITION AT LAST TIME FROM STORAGE VECTOR
 RDA
 RDA READ CONTINUOUS RADIONUCLIDE REMOVAL RATES
 INP
      -18 0 -1 3 1
                        1
 MOV
        -14 1 0 1.0
         IRRADIATION OF CSFR FUEL SALT
 TIT
 BUP
           -16 12 0 0.2891
   MOV
   ADD
           1 12 0 0.8050
           88.0 121.6
                           12 2 5 0
 IRP
           -16 12 0 0.2891
   MOV
   ADD
            2 12 0 0.8050
          90.0 121.6
 IRP
                         12 3 5 0
           -16 12 0 0.2891
   MOV
            3 12 0 0.8050
   ADD
          92.0 121.6
 TRP
                          12 4 5 0
   MOV
           -16 12 0 0.2891
           4 12 0 0.8050
   ADD
           94.0 121.6
 IRP
                          12 5 5 0
           -16 12 0 0.2891
   MOV
   ADD
           5 12 0 0.8050
           96.0 121.6
 IRP
                           12 6 5 0
   MOV
           -16 12 0 0.2891
   ADD
           6 12 0 0.8050
           98.0 121.6
 TRP
                           12 7 5 0
           -16 12 0 0.2891
   MOV
            7 12 0 0.8050
   ADD
 IRP
         100.0 121.6
                         12 8 5 0
 BUP
 OPTL 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OPTA 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OPTF 2*8 8 8 2 8 8 11*8 8 8 8 8 2*8
 OUT
         8 1 -1 0
RDA *****SAVE RESULTS AT SELECTED TIMES
 MOV 4 -30 0 1.0
```

MOV	8 -	-31	0	1.0					
RDA	MOVE	SAV	ED (COMPOS	SITIONS	TO (DUTPUI	VEC	CTOR
MOV	-20	1 0	1.	0					
MOV	-21	2 0	1.	0					
MOV	-22	3 0	1.	0					
MOV	-23	4 0	1.	0					
MOV	-24	5 0	1.	0					
MOV	-25	60	1.	0					
MOV	-26	7 0	1.	0					
MOV	-27	8 0	1.	0					
MOV	-28	90	1.	0					
MOV	-29	10	0 1	.0					
MOV	-30	11	0 1	.0					
MOV	-31	12	0 1	.0					
TIT	CSI	FR F	UEL	SALT	IRRADIA	ATIOI	J FOR	100	YEARS
OUT	-	12 1	-1	0					
END									

Table C-1 Updated cross sections used in follow-on evaluation

Nuclide	Capture cross section, barns	Fission cross section, barns
²³ Na	0.00023	
³⁵ Cl	0.0053	
³⁶ Cl	0.0099	
³⁷ Cl	0.00074	
²³⁵ U	0.51	1.43
²³⁸ U	0.26	0.025
²³⁸ Pu	0.58	
²³⁹ Pu	0.42	1.46
²⁴⁰ Pu	0.47	
²⁴¹ Pu	0.39	2.17
²⁴² Pu	0.39	

D ORIGEN2 OUTPUT OF THE FISSION PRODUCT CONCENTRATION IN FUEL SALT FROM 0 TO 100 FULL-POWER YEARS OF IRRADIATION

ORIGEN2 Output of the Fission Product Concentration in Fuel Salt from 0 to 100 Full-Power Years of Irradiation

Table D-1 Equilibrium fission product element mass in one MTHM CSFR fuel salt as a function of irradiation time at 300 kW/L

	BOC FUEL	8.0YR	20.0YR	26.0YR	34.0YR	42.0YR	52.0YR	64.0YR	74.0YR	86.0YR	92.0YR	100.0YR
Н	0.000E+00	5.512E-01	7.253E-01	7.432E-01	7.510E-01	7.531E-01	7.538E-01	7.541E-01	7.541E-01	7.540E-01	7.540E-01	7.542E-01
LI	0.000E+00	2.530E-03	3.022E-03	3.051E-03	3.062E-03	3.064E-03	3.065E-03	3.066E-03	3.066E-03	3.066E-03	3.066E-03	3.066E-03
BE	0.000E+00	1.247E-03	1.904E-03	2.020E-03	2.093E-03	2.124E-03	2.139E-03	2.144E-03	2.146E-03	2.146E-03	2.146E-03	2.146E-03
CO	0.000E+00	8.197E-12	8.185E-12	8.187E-12	8.186E-12	8.188E-12	8.186E-12	8.188E-12	8.190E-12	8.188E-12	8.188E-12	8.189E-12
NI	0.000E+00	3.686E-09	3.698E-09	3.700E-09	3.699E-09	3.700E-09	3.699E-09	3.700E-09	3.701E-09	3.700E-09	3.700E-09	3.701E-09
CU	0.000E+00	3.792E-08	3.806E-08	3.808E-08	3.808E-08	3.808E-08	3.807E-08	3.809E-08	3.810E-08	3.808E-08	3.809E-08	3.809E-08
ZN	0.000E+00	5.063E-07	5.070E-07	5.073E-07	5.073E-07	5.074E-07	5.073E-07	5.074E-07	5.076E-07	5.074E-07	5.075E-07	5.075E-07
GA	0.000E+00	1.612E-06	1.613E-06	1.614E-06	1.614E-06	1.614E-06	1.614E-06	1.615E-06	1.615E-06	1.614E-06	1.615E-06	1.615E-06
GE	0.000E+00	1.385E-05	1.386E-05	1.387E-05	1.387E-05	1.387E-05	1.387E-05	1.387E-05	1.388E-05	1.387E-05	1.387E-05	1.388E-05
AS	0.000E+00	3.202E-05	3.201E-05	3.203E-05	3.203E-05	3.203E-05	3.203E-05	3.204E-05	3.205E-05	3.203E-05	3.204E-05	3.204E-05
SE	0.000E+00	1.106E+02	1.665E+02	1.762E+02	1.821E+02	1.846E+02	1.857E+02	1.862E+02	1.863E+02	1.863E+02	1.863E+02	1.863E+02
BR	0.000E+00	2.358E+01	2.925E+01	2.972E+01	2.992E+01	2.998E+01	3.000E+01	3.001E+01	3.001E+01	3.001E+01	3.001E+01	3.001E+01
KR	0.000E+00	9.061E-04	9.062E-04	9.068E-04	9.068E-04	9.070E-04	9.067E-04	9.070E-04	9.073E-04	9.070E-04	9.071E-04	9.072E-04
RB	0.000E+00	1.534E+02	2.285E+02	2.412E+02	2.489E+02	2.521E+02	2.536E+02	2.541E+02	2.542E+02	2.543E+02	2.543E+02	2.543E+02
SR	0.000E+00	1.306E+03	1.841E+03	1.917E+03	1.958E+03	1.973E+03	1.979E+03	1.981E+03	1.982E+03	1.982E+03	1.982E+03	1.982E+03
Y	0.000E+00	6.161E+02	8.832E+02	9.286E+02	9.562E+02	9.674E+02	9.726E+02	9.745E+02	9.750E+02	9.750E+02	9.750E+02	9.752E+02
ZR	0.000E+00	2.094E+04	3.166E+04	3.354E+04	3.471E+04	3.518E+04	3.541E+04	3.550E+04	3.552E+04	3.552E+04	3.552E+04	3.553E+04
NB	0.000E+00	2.693E-03	2.723E-03	2.730E-03	2.733E-03	2.735E-03	2.735E-03	2.736E-03	2.737E-03	2.736E-03	2.736E-03	2.737E-03
MO	0.000E+00	4.945E-03	4.953E-03	4.957E-03	4.956E-03	4.957E-03	4.956E-03	4.958E-03	4.960E-03	4.958E-03	4.958E-03	4.959E-03
TC	0.000E+00	1.231E-03	1.235E-03	1.236E-03	1.236E-03	1.236E-03	1.236E-03	1.236E-03	1.237E-03	1.236E-03	1.236E-03	1.236E-03
RU	0.000E+00	8.896E-04	8.930E-04	8.937E-04	8.936E-04	8.938E-04	8.936E-04	8.939E-04	8.942E-04	8.939E-04	8.940E-04	8.941E-04
RH	0.000E+00	1.500E-04	1.508E-04	1.509E-04	1.509E-04	1.509E-04	1.509E-04	1.509E-04	1.510E-04	1.509E-04	1.509E-04	1.510E-04
PD	0.000E+00	1.274E-04	1.280E-04	1.281E-04								
AG	0.000E+00	2.804E-05	2.808E-05	2.809E-05	2.809E-05	2.810E-05	2.809E-05	2.810E-05	2.811E-05	2.810E-05	2.810E-05	2.811E-05
CD	0.000E+00	6.401E-05	6.404E-05	6.407E-05	6.407E-05	6.408E-05	6.406E-05	6.408E-05	6.411E-05	6.408E-05	6.409E-05	6.410E-05
IN	0.000E+00	4.157E-05	4.159E-05	4.161E-05	4.161E-05	4.162E-05	4.161E-05	4.162E-05	4.163E-05	4.162E-05	4.162E-05	4.163E-05
SN	0.000E+00	9.095E-04	9.104E-04	9.110E-04	9.109E-04	9.111E-04	9.109E-04	9.112E-04	9.115E-04	9.111E-04	9.112E-04	9.114E-04
SB	0.000E+00	1.398E-03	1.400E-03	1.401E-03	1.401E-03	1.401E-03	1.401E-03	1.401E-03	1.402E-03	1.401E-03	1.401E-03	1.401E-03
TE	0.000E+00	2.793E-03	2.797E-03	2.799E-03	2.799E-03	2.800E-03	2.799E-03	2.800E-03	2.801E-03	2.800E-03	2.800E-03	2.801E-03
I	0.000E+00	3.429E+00	3.609E+00	3.621E+00	3.624E+00	3.625E+00	3.625E+00	3.626E+00	3.627E+00	3.625E+00	3.626E+00	3.626E+00
XE	0.000E+00	5.405E-03	5.413E-03	5.417E-03	5.417E-03	5.418E-03	5.417E-03	5.418E-03	5.420E-03	5.418E-03	5.418E-03	5.419E-03
CS	0.000E+00	1.300E+03	1.836E+03	1.911E+03	1.950E+03	1.964E+03	1.969E+03	1.971E+03	1.971E+03	1.971E+03	1.971E+03	1.971E+03
BA	0.000E+00	3.021E+03	4.708E+03	5.029E+03	5.238E+03	5.328E+03	5.373E+03	5.391E+03	5.395E+03	5.396E+03	5.396E+03	5.397E+03
LA	0.000E+00	4.417E+03	6.591E+03	6.950E+03	7.165E+03	7.249E+03	7.288E+03	7.301E+03	7.305E+03	7.305E+03	7.305E+03	7.306E+03
CE	0.000E+00	1.656E+04	2.442E+04	2.578E+04	2.662E+04	2.697E+04	2.713E+04	2.719E+04	2.721E+04	2.721E+04	2.721E+04	2.721E+04
PR	0.000E+00	8.141E+03	1.138E+04	1.180E+04	1.202E+04	1.209E+04	1.212E+04	1.213E+04	1.213E+04	1.213E+04	1.213E+04	1.213E+04
ND	0.000E+00	2.813E+04	4.375E+04	4.651E+04	4.823E+04	4.893E+04	4.926E+04	4.939E+04	4.942E+04	4.942E+04	4.942E+04	4.943E+04
PM	0.000E+00	8.400E+02	9.588E+02	9.814E+02	9.959E+02	1.002E+03	1.005E+03	1.006E+03	1.007E+03	1.006E+03	1.006E+03	1.007E+03
SM	0.000E+00	9.303E+03	1.508E+04	1.614E+04	1.679E+04	1.706E+04	1.718E+04	1.723E+04	1.724E+04	1.724E+04	1.724E+04	1.724E+04
EU	0.000E+00	1.083E+03	1.731E+03	1.869E+03	1.959E+03	1.997E+03	2.014E+03	2.021E+03	2.022E+03	2.022E+03	2.022E+03	2.023E+03
GD	0.000E+00	1.392E+03	3.197E+03	3.709E+03	4.103E+03	4.294E+03	4.395E+03	4.434E+03	4.443E+03	4.446E+03	4.447E+03	4.448E+03
TB	0.000E+00	7.453E+01	1.797E+02	2.204E+02	2.563E+02	2.759E+02	2.872E+02	2.919E+02	2.932E+02	2.936E+02	2.936E+02	2.937E+02
DY	0.000E+00	1.145E+02	3.631E+02	4.831E+02	6.091E+02	6.908E+02	7.449E+02	7.708E+02	7.782E+02	7.810E+02	7.814E+02	7.818E+02
HO	0.000E+00	2.423E+00	1.179E+01	1.833E+01	2.703E+01	3.419E+01	4.008E+01	4.353E+01	4.471E+01	4.524E+01	4.534E+01	4.542E+01
ER	0.000E+00	3.891E+00	2.015E+01	3.450E+01	5.728E+01	7.995E+01	1.027E+02	1.193E+02	1.263E+02	1.299E+02	1.307E+02	1.314E+02
TM	0.000E+00	1.166E-02	8.975E-02	1.766E-01	3.476E-01	5.565E-01	8.111E-01	1.037E+00	1.148E+00	1.214E+00	1.230E+00	1.243E+00
YB	0.000E+00	9.811E-03	1.496E-01	3.363E-01	7.736E-01	1.422E+00	2.397E+00	3.499E+00	4.179E+00	4.680E+00	4.828E+00	4.955E+00
TOTAL	0.000E+00	9.753E+04	1.490E+05	1.583E+05	1.641E+05	1.666E+05	1.678E+05	1.682E+05	1.683E+05	1.684E+05	1.684E+05	1.684E+05



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