MoSS Reactor – Molten Static Salt Reactor

Alicia Borel, Jaden Minnick, Matthew Louis, Nathan Tanzosch

Georgia Tech

Background

History of Molten Salt Reactor Concepts

Aircraft Reactor Experiment at Oak Ridge National Lab in 1954

- First successful use of circulating liquid fuel in a reactor
- Notably this reactor was designed to counter Xenon poisoning

Molten Salt Reactor Experiment at Oak Ridge National Lab 1960s

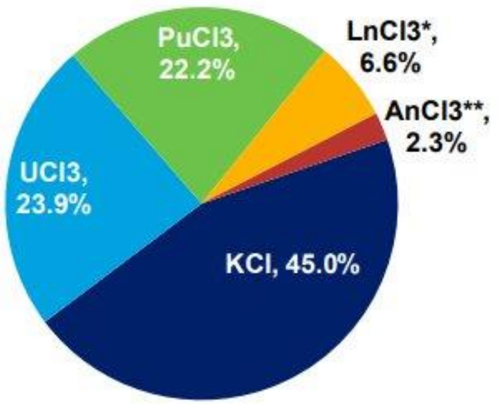
- Fluoride fuel salt flows inside graphite moderating channels
- Successfully went critical in 1965

Key takeaways

- High outlet temperature of over 600°C
- Fission products could be separated from the system while online
- Issues with corrosion and low Chromium alloys (Hastelloy-N)

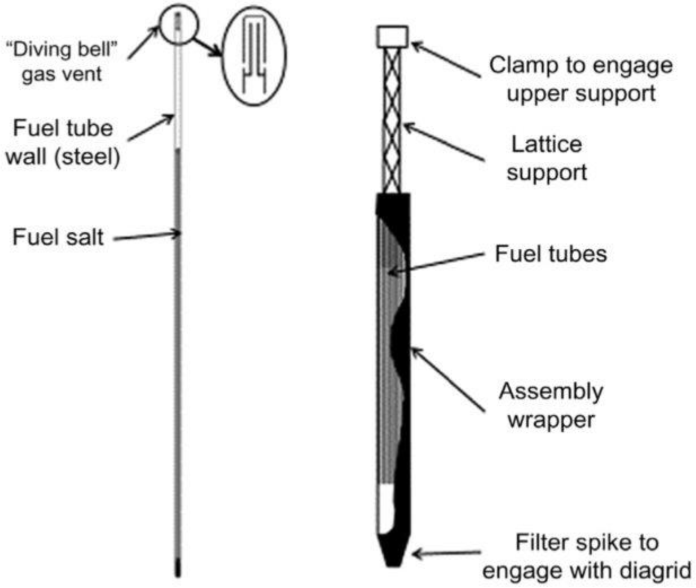
Moltex Stable Salt Design

Chloride fuel with high impurity level



Pu vector is approximately 2/3 fissile based on CANDU spent fuel composition. U vector is below natural enrichment levels.

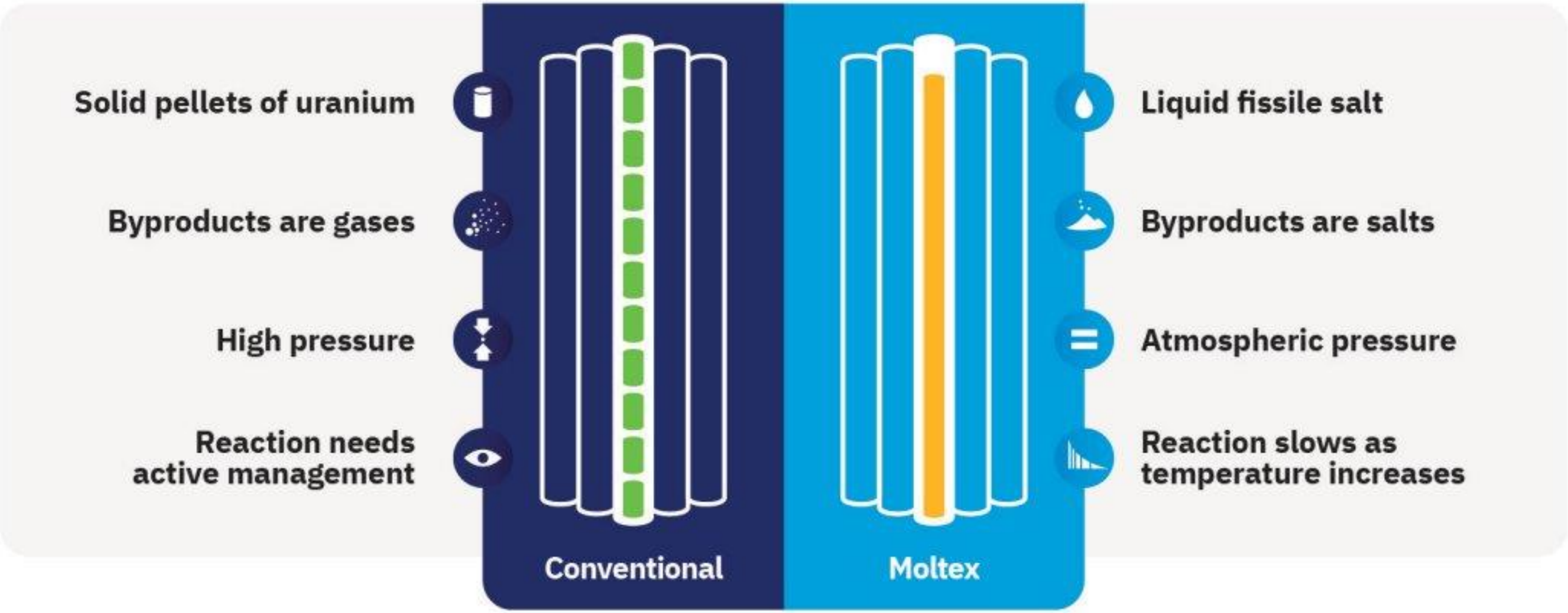
*Ln=Lanthanides; **An=Actinides



- Configuration is based on the Fuel Assembly
- Patented the fuel tube design (expiration in 2035)
- Stable Salt Reactors (SSR-W & SSR-U)
- Fuel Salt – Potassium Chloride
- Coolant Salt – Magnesium Chloride or Sodium Chloride
- 750 MWt SSR-W licensed and being constructed in Canada using reprocessed CANDU fuel

[1] O'Sullivan, Rory. *Moltex Technology: An overview for the National Academies of Sciences, Engineering and Medicine committee on fuel cycles and waste aspects of advanced nuclear reactors*. February 23, 2021.

Comparison of Fuel Pins



[1] O'Sullivan, Rory. *Moltex Technology: An overview for the National Academies of Sciences, Engineering and Medicine committee on fuel cycles and waste aspects of advanced nuclear reactors*. February 23, 2021.

Perceived Weaknesses of SSR-W

Utilizes low-burnup CANDU Waste

<10% of all nuclear waste

<0.4% Plutonium Content

Lanthanides in fuel salt

Requires active pumping for salt circulation

Project Scope

Project Design

★ Optimize reactor operating parameters for transuranic burning in a novel molten salt fuel tube design by repurposing spent legacy fuel

1. The fuel tube resembles a traditional LWR fuel rod except it contains molten salt fuel compared to solid oxide fuel.

2. The fuel composition is composed of elements found in legacy LWR fuel waste

Objectives

Optimize salt composition for transuranic burning such that MoSS Reactor fuel waste reaches natural uranium radiotoxicity

Show that fuel tubes can be adequately cooled and find optimal geometry for heat transfer

Show that structural materials can handle anticipated corrosion and high temperature

Determine if this design can be completely pumpless

Demonstrate safety of fission product gas venting into coolant

Discover "Break-even" cost to charge for reprocessing waste

Societal Impact

Burning legacy nuclear waste

- Fewer proliferation concerns
- Reduces long-term nuclear storage concerns
- Incineration of weapons grade plutonium

Clean energy able to replace current fossil systems

- Coal and natural gas infrastructure is compatible with high core outlet temperature

Inherent safety through passive systems

- Negative temperature reactivity coefficient
- No pressurization
- Gaseous products captured in reactor vessel head

Jaden

Project Scope

In Scope	Out of Scope
Fast Spectrum	Thermal Spectrum/Moderated Core
3 Batch Reload	Secondary Loops
Chloride Fuel Salt	Detailed Valve Design
Fluoride Coolant Salt	Multiple Reprocessing Steps
Corrosion Resistance	
Fission Product Solubility	
Fluid Dynamics & Convective Heat Transfer	

Success Metrics

A successful design will:

- Meet Current Safety Regulations
- Reduce Radiotoxicity of Spent Fuel Waste
- Have a low break-even cost to charge for fuel reprocessing

Project Requirements

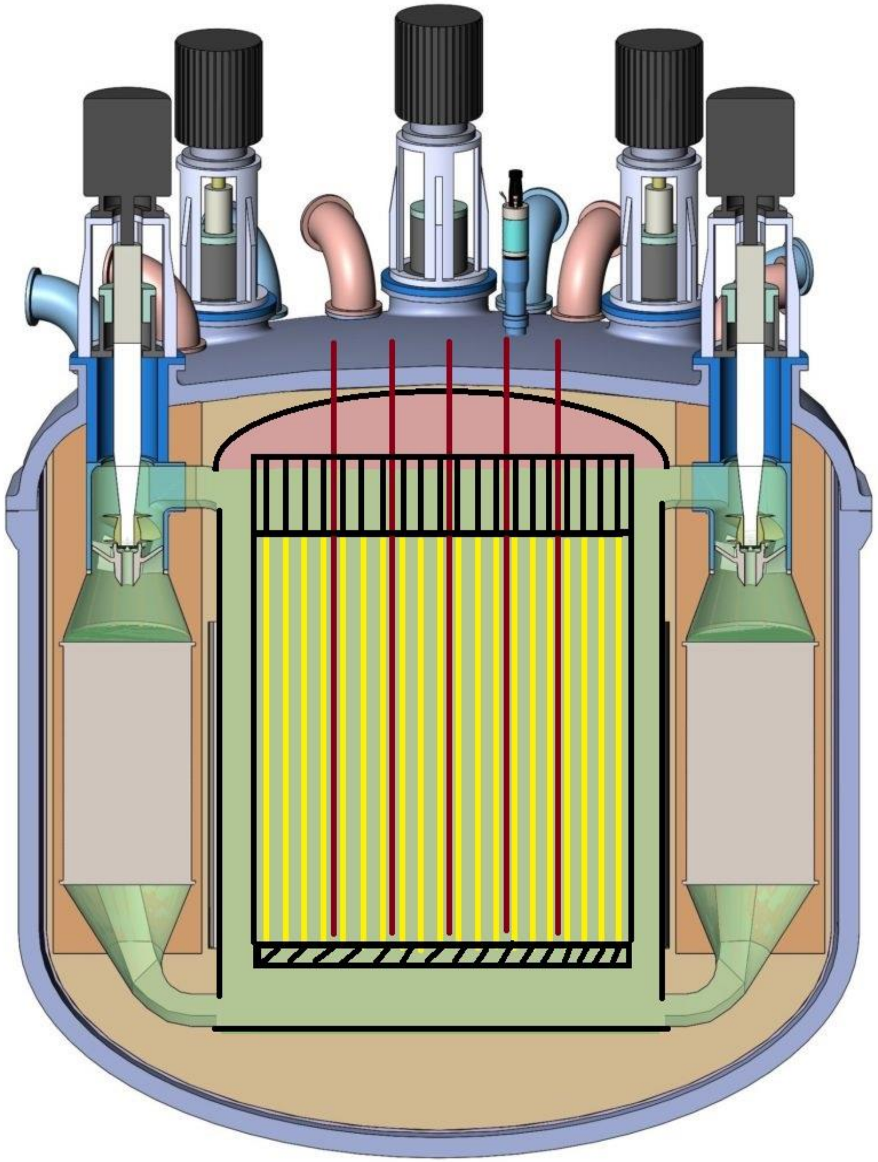
- Structural Integrity Corrosion Analysis
- Fuel and Coolant Salt Optimization
- Neutronics based Core Geometry Optimization
- Coupled Thermohydraulic and Flow Dynamic Analysis
- Safety and Accident Scenario Analysis
- Fuel Waste Radiotoxicity Comparison
- Economic Estimates of Fuel Cycle

Team Responsibilities

- **Nathan**: Team Lead, Modeling & Fuel Cycle
- **Matthew**: Member, Heat Transfer & CFD, Corrosion Resistance Analysis
- **Jaden**: Member, Neutronics, Modeling, & Economic Analysis
- **Alicia**: Member, Fission Product Analysis, Safety, Meeting Notes

Reactor Design

Core Graphic

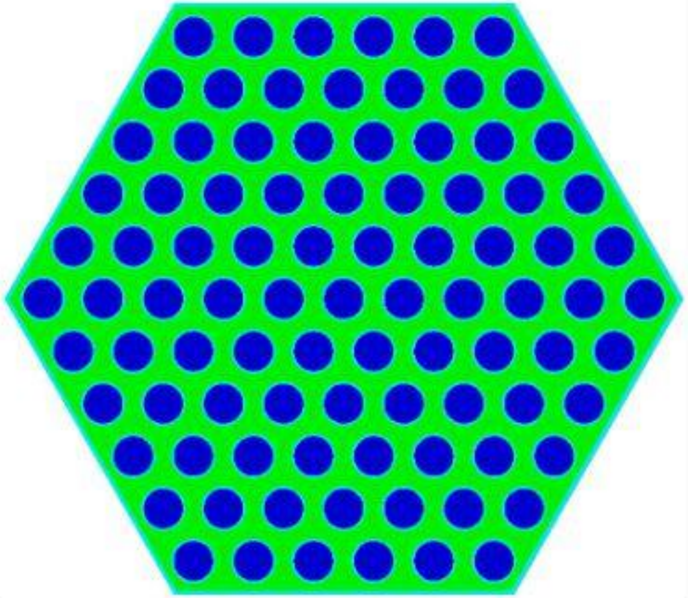


Core Parameters

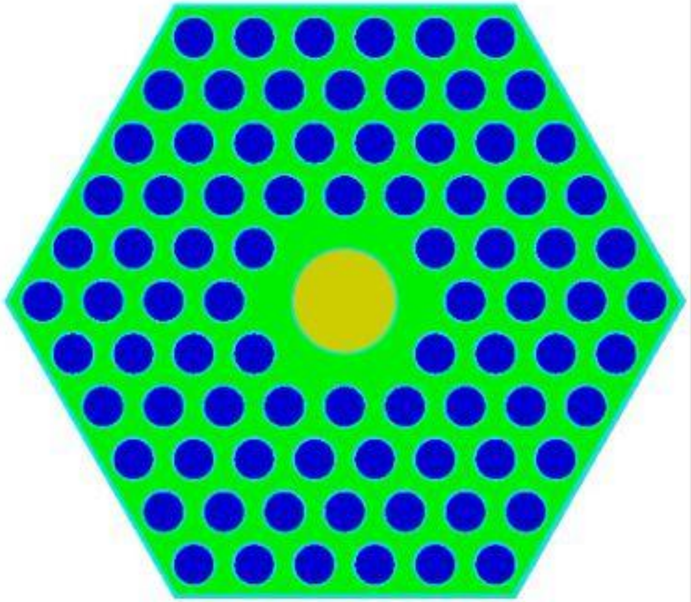
Parameter	Value
Power density (of fuel salt)	100 kW/L
Core Power	426 MWt
Lattice type	Hexagonal
Fuel tube radius	5.5 mm
Fuel tube thickness	0.6 mm
Redox control	Zr insert in fuel salt, ZrF2 additives in coolant
Clad material	SS 316
Vessel material	SS 316
Fuel tube venting	Diving Bell above coolant level

Assembly Models

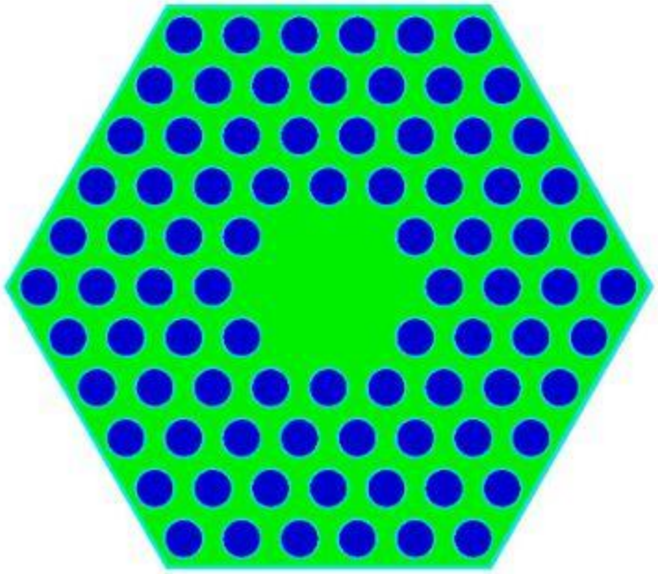
Filled Assembly
91 fuel rods



Control Rod Assembly
84 fuel rods

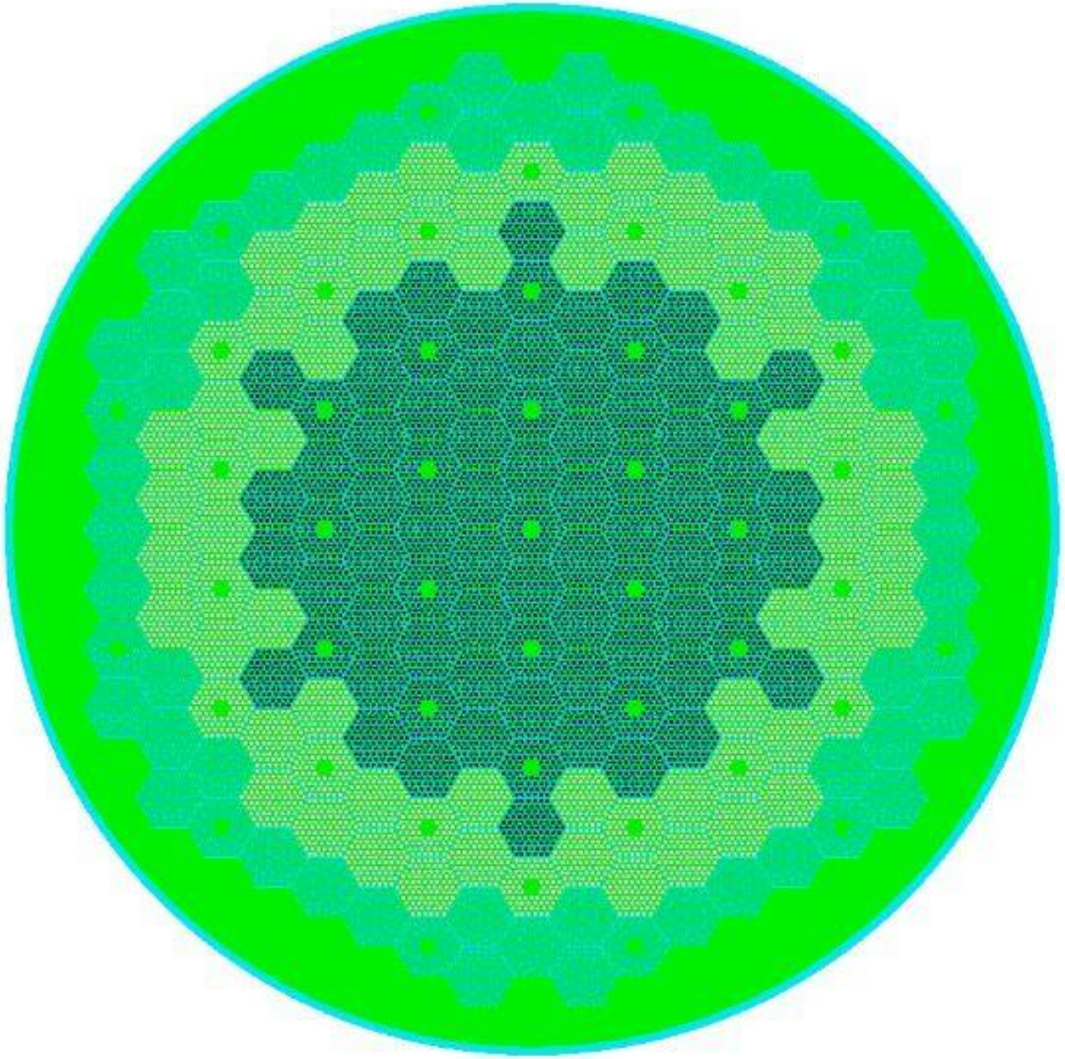


Empty Guide Tube Assembly
84 fuel rods



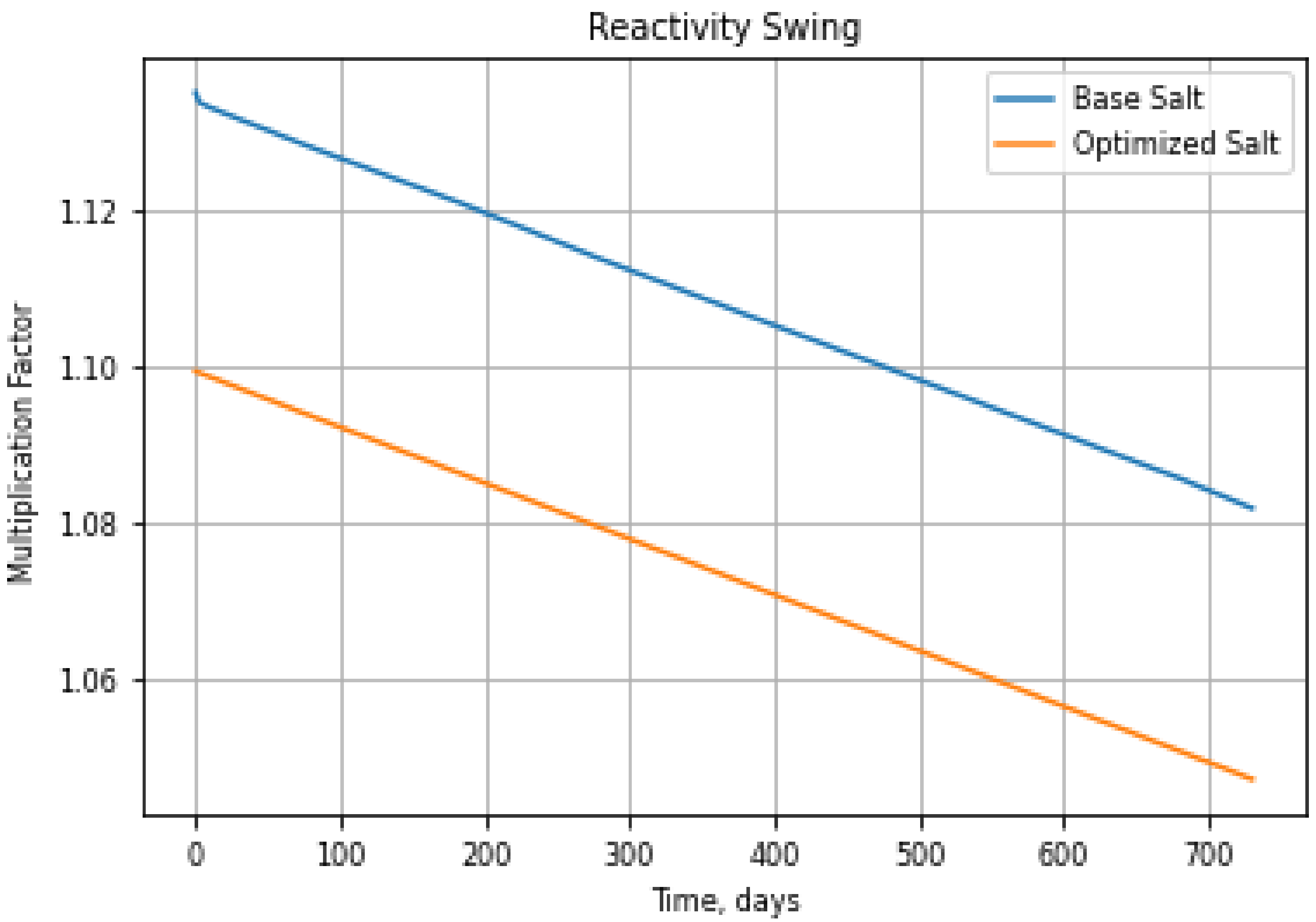
Core

Total Assemblies	211
Total Rods	18858
Zone 1 Assemblies (Fresh)	79 Total 19 GT – 60 Filled
Zone 2 Assemblies (Once Burned)	72 18 GT – 54 Filled
Zone 3 Assemblies (Twice Burned)	60 12 GT – 48 Filled
Core Radius	1.25 meter
Active Height	3m
Total Height	5m



3-Batch Solution

- Iteratively depleted average region isotopic compositions until converged
- Final fuel cycle reactivity swing measured and extrapolated to beginning optimized salt composition
- Two-year cycle
  - 6.22 pcm loss per day
- Use of boron burnable absorbers shown to limit MSR cycle reactivity swing to less than 1000 pcm [1]



[1] Eunhyug Lee, Taesuk Oh, and Yonghee Kim. *Burnable Absorber Design Study for Fast Spectrum Molten Salt Reactors*. Korean Nuclear Society, October, 2023.

Reactivity Coefficients

	dT	α	σ
Coolant 900K	50	0.0000162	0.0000017
Coolant 1000K	150	0.0000174	0.0000006
Coolant 1100K	250	0.0000173	0.0000004
Fuel 1550K	50	-0.0001749	0.0000022
Fuel 1600K	150	-0.0001197	0.0000007
Fuel 1700K	200	-0.0001869	0.0000005

- Weakly positive coolant temperature reactivity coefficient in the one sigma range of 1.4 to 1.8 pcm/K
- Strong negative fuel temperature reactivity coefficient in the one sigma range of -11.9 to -18.7 pcm/K

Control Rod Worth

	ARO (All Rods Out)	ARI (All Rods In)
K_eff	1.16003 + or - 0.00011	0.985433 + or - 0.000085
Dryout K_eff	1.176280 + or - 0.000063	1.02451 + or - 0.00010

- At startup, control rods fully control excess reactivity
- Positive reactivity insertion when all coolant is drained
- Supercritical in case of LOCA accident, however, this did not account for other factors that reduce multiplication factor
  - Axial leakage
  - 3-batch fuel load
  - Optimized salt compositions
  - Burnable Absorbers

Corrosion and Materials

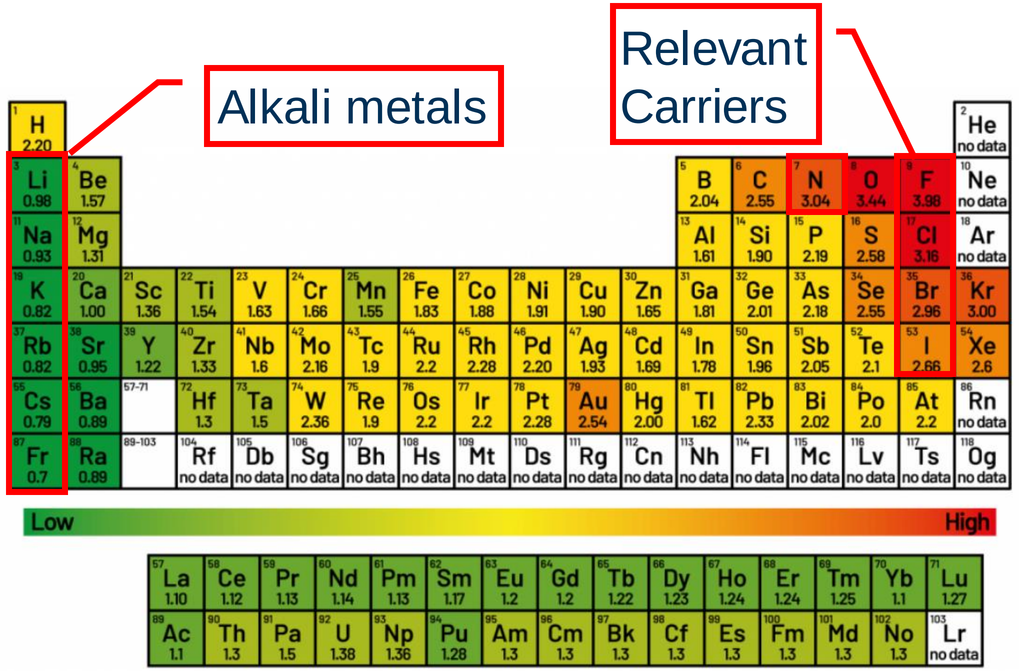
Matthew

Molten Salt Fundamentals

Salt Chemistry:

- Typically, an alkali metal that forms an ionic bond with an electronegative halogen
- Ionic bonds typically occur between atoms with very different electronegativities
  - Electron **donors** and **acceptors**
- Halogen referred to as the salt "carrier"
  - Carrier can be either *organic* or *inorganic*
- For industrial use, common carriers are
  - N, F, Cl, I, Br
- Industrial salts typically mixes of several *pure salts, e.g. FLiNaK*
  - LiF−NaF−KF (46.5-11.5-42 mol %)

Endmembers



Choice of Fuel and Coolant Carrier

Exclusion of Bromides and Iodides

- Recently being considered for nuclear applications, but
    - Lack of data
    - Expensive
    - Generally inferior for wasteburning

Coolant Carrier

- Fluorine:
    - Superior thermophysical properties and better for heat transfer (hence why salt cooled designs generally use F)

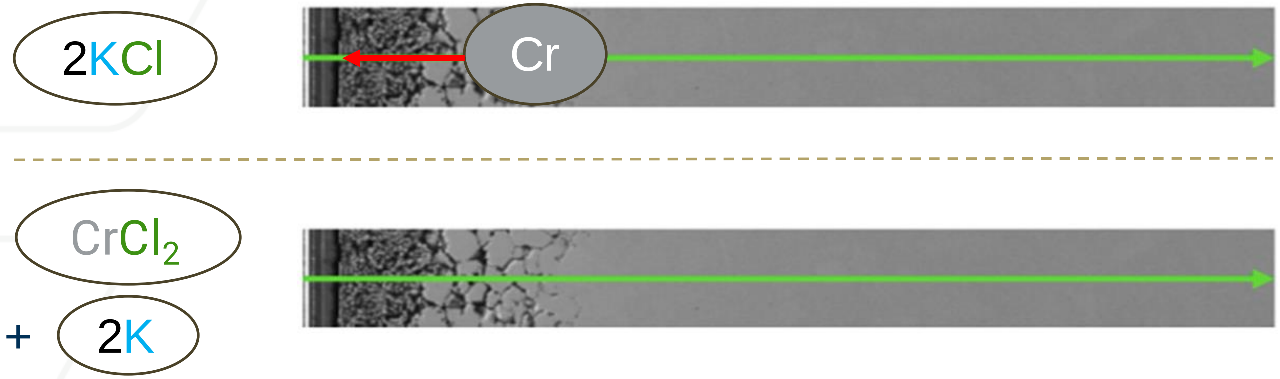
Fuel Carrier

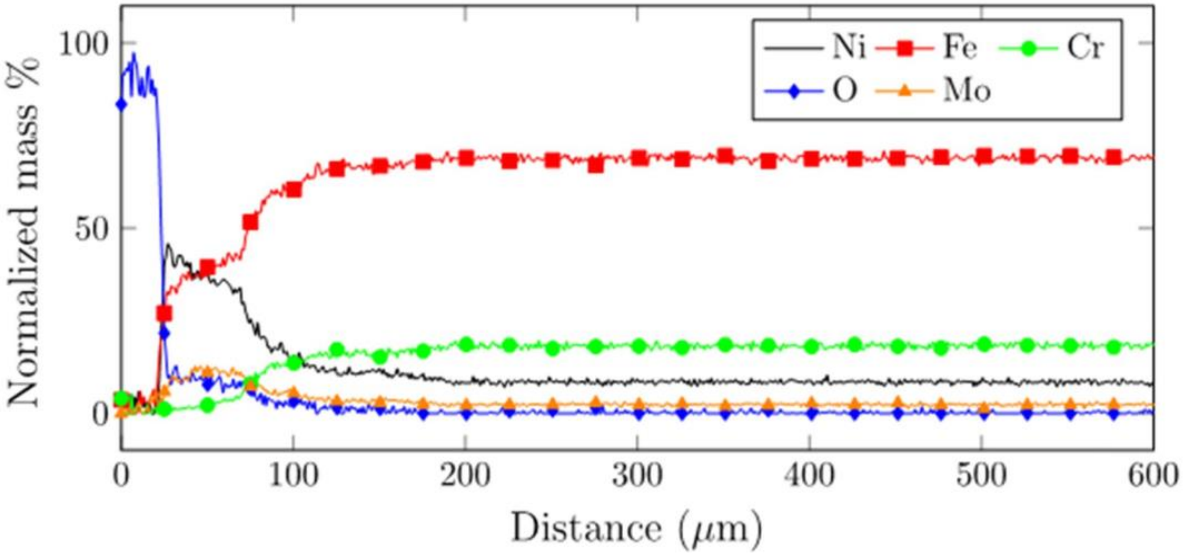
- Chlorine
    - High actinide solubility
    - Higher A better for fast systems
    - Typically considered for fast MSRs (excluding e.g. SAMOFAR)

Introduction to Corrosion

Molten Salt

Metal (e.g. Stainless steel 316)





- Molten salts are notoriously corrosive
  - Special corrosion resistant **Hastelloy N** used historically (in the MSRE) to withstand corrosion
- Higher rates of corrosion at higher temperatures
- **Corrosion in Molten Salts:**
  - Normally, corrosion is mitigated by formation of a passivating oxide layer
  - In molten salts, these oxide layers are generally soluble, and dissolve
  - Corrosion is essentially *thermodynamically* driven
- **The Oxidation Reduction Reaction**
  - $xM + yAC \rightleftharpoons M_xC_y + yA$
  - M = Pure metal (Cr)
  - A = Alkali metal (K)
  - C = Salt carrier (F)

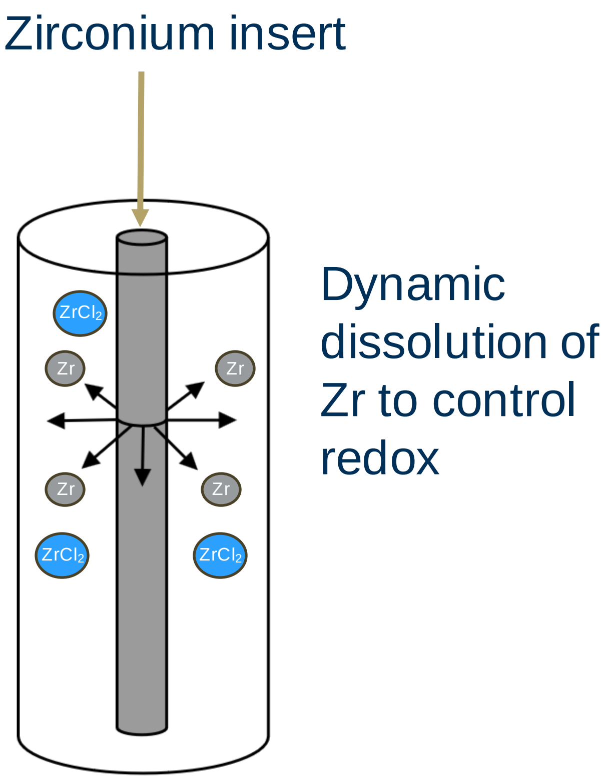
Corrosion Control and Calculation Methodology

Redox Control

- The redox potential (an electrostatic potential in V) characterizes the **rate** of corrosion
- Can lower the redox potential via additions of a reducing element to the salt
  - Scott in [1] suggests the use of a Zr sacrificial metal in the fuel salt (an insert that dissolves and maintains the salt in a highly reducing state)
  - Controlling the ratio of ZrF_2/ZrF_4 in the coolant

Modelling Corrosion

- Worst-case estimate by modelling metal dissolution in **thermodynamic equilibrium**
  - given a potentially infinite amount of time for the corrosion reaction to stabilize in the forward and reverse directions
- Equilibrium thermochemistry by Gibbs energy minimization
  - Thermochimica
  - Molten Salt Thermal Properties Database (MSTDB)



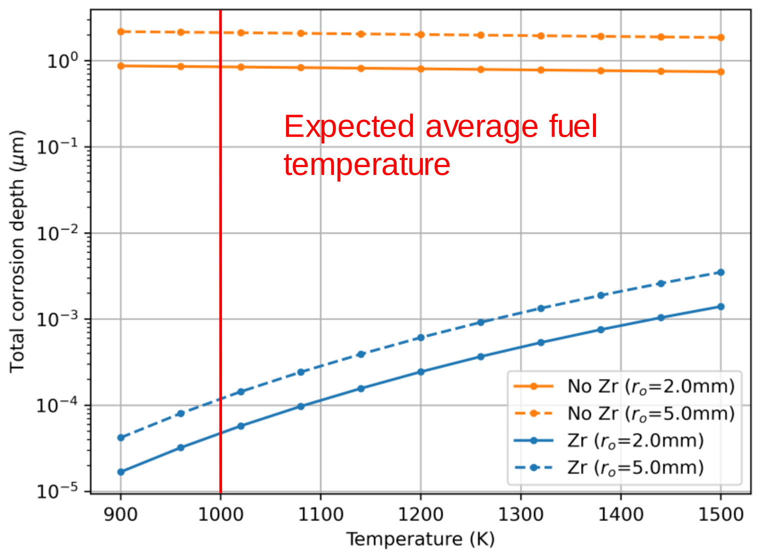
Cannot be used in pumped salts due to thermal gradient driven corrosion

- Causes fouling of pumps

[1] Chapter 14: Static liquid fuel reactors: T. J. Dolan. Molten salt reactors and thorium energy. Woodhead Publishing, 2017.

Fuel Corrosion

- No code for commercial use of Hastelloy N in nuclear reactors [1]
- Wanted to see if stainless steel 316H could be used with sufficient redox control
- Used reference chloride salt because exact fuel salt composition not yet determined

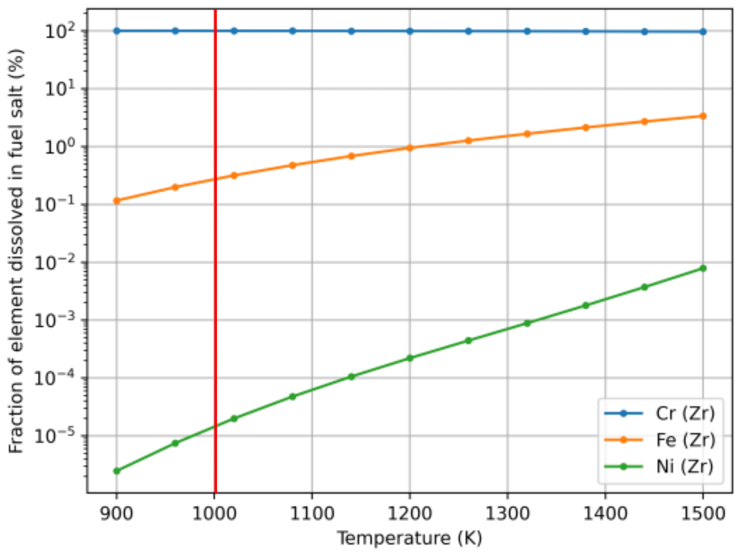


Element	Range (wt%)	Average (wt%)	Average (mol%)
Cr	16-18	17	18.1
Ni	10-14	12	11.3
Mo	2-3	2.5	1.4
Mn	2	2	2
Si	0.75	0.75	1.5
C	0.08	0.08	0.37
Ti	-	-	-
Nb	-	-	-
Fe	Remaining	67.67	65.2

Stainless Steel 316H Composition

Element	Moles of element
Pu	0.289756607
U	0.200000000
Am	0.027444106
Np	0.023517078
Cm	0.001890731
K	0.449999550
Cl	2.100000000

Fuel Salt Composition



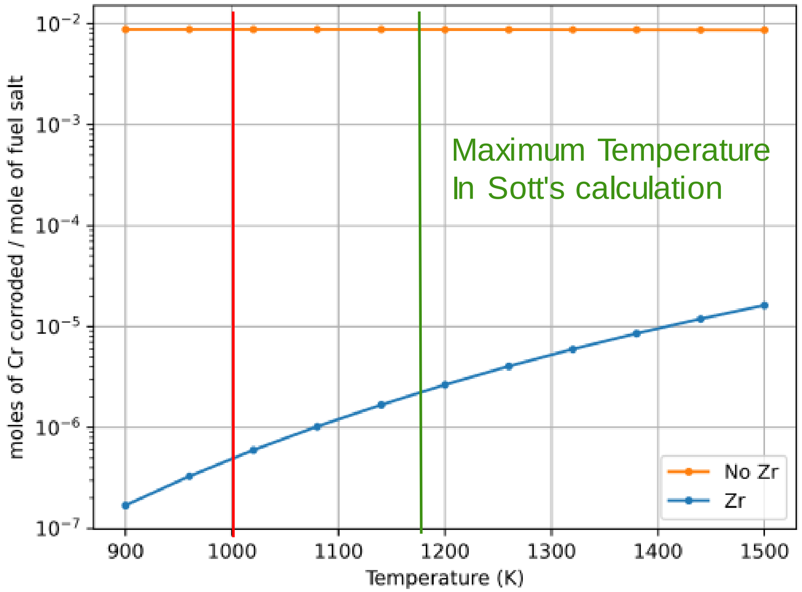
[1] Ren, Weiju. "Considerations of alloy N code extension for commercial molten salt reactor development and deployment." *Pressure Vessels and Piping Conference*. Vol. 51685. American Society of Mechanical Engineers, 2018.

Fuel Corrosion (Continued)

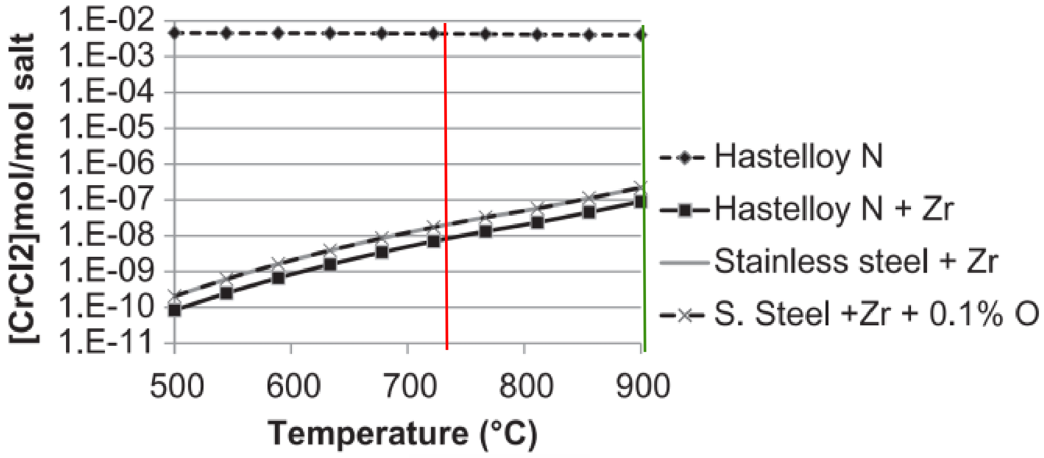
- Hard to find criteria on acceptable corrosion depth
- Instead compared chromium dissolution with a reference [1] to determine sufficiency of corrosion resistance
- Scott claims that a chromium dissolution of 0.5 ppm is acceptable (dissolution in Scott's calculation at max temp)
  - Near calculated chromium dissolution at operating temperature

Conclusion:

- Stainless steel 316H sufficient on fuel side



Thermochimica Calculation

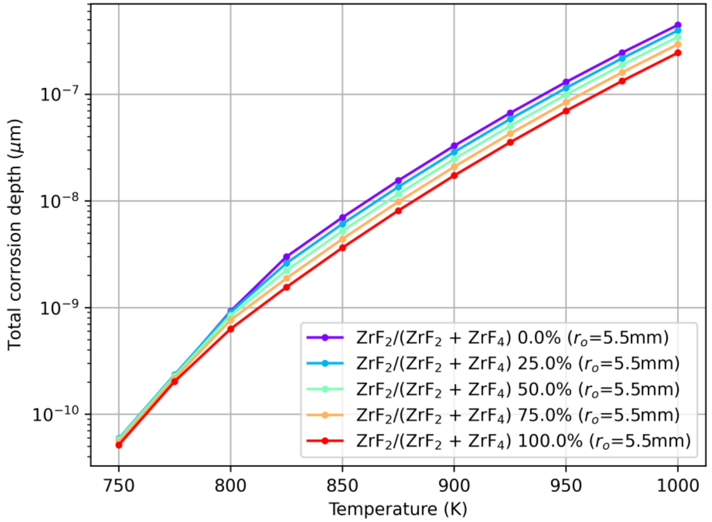


A reference calculation done by Scott [1]

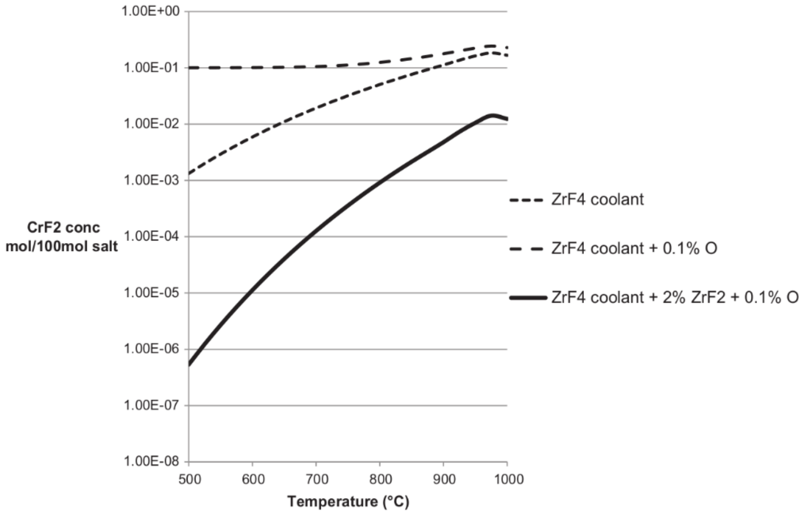
[1] Chapter 14: Static liquid fuel reactors: T. J. Dolan. Molten salt reactors and thorium energy. Woodhead Publishing, 2017.

Coolant Corrosion

- For all expected coolant temperatures (less than 1000 K), corrosion depth *much* lower than for fuel-side
- Relatively insensitive to ZrF_2/ZrF_4 perhaps due to low Zr fraction in the coolant salt
  - Scott shows a *much* stronger dependence
- Since less than fuel side, which was deemed acceptable -> stainless steel 316H is acceptable on the coolant side as well.



Thermochimica Calculation



A reference calculation done by Scott [1]

Endmember	Mole Fraction (%)
NaF	13.2
ZrF_4	3.2
KF	41.09
LiF	41.77

Coolant Composition

[1] Chapter 14: Static liquid fuel reactors: T. J. Dolan. Molten salt reactors and thorium energy. Woodhead Publishing, 2017.

Salt Selection and Optimization

Motivation

- Even when constrained to a specific carrier (i.e. F or Cl), design space of salts is *very* large
- Thermophysical properties depend on endmember fractions
  - *Large* range of thermophysical properties
- Different operating temperatures, and heat transfer mechanisms, so it's not clear apriori that the optimal salt will be one from the literature
- Discrepancies in thermophysical properties etc provided by Scott
- Proper choice of endmember fractions could have *large* effect on feasibility of design

Methodology:

- Constrained optimization on endmember fractions to determine optimal coolant and fuel salts

Calculating Thermophysical Properties

- Thermophysical properties important for safety and performance
- Severe lack of experimental data for higher order molten salt systems
  - Especially regarding composition dependence
- Molten salts are *expensive* to measure (high melting points, possibly volatile at atmospheric pressure)
- Need to rely on theoretical interpolation methods instead
- **Ideal approximations**
  - Total is some weighted combination of thermophysical properties of endmembers
  - Does not account for enthalpy of mixing
- **Redlich Kister Expansion:** Accounts for non-ideal mixing, but only expansion coefficients for density
- Developed custom code for calculating thermophysical properties using these methods

Coolant Salt Optimization

- **Optimization Criteria:**
- For fixed ΔT, height, and radius, heat transfer to the coolant dependent only on heat transfer coefficient
- Constrained optimization of heat transfer coefficient
  - Higher melting point means smaller ΔT, and larger heating requirements
  - Constrained melting point to be below 750 K
- Considered the endmembers (based on database availability):
  - NaF, KF, ZrF_4, CaF_2, LiF
  - LiF undesirable from a neutronic viewpoint, requires enrichment in ^7Li to avoid tritium production, but *necessary* for achieving a reasonably low melting point
  - Lithium enrichment historically environmentally damaging, new methods like electromigration can mitigate this

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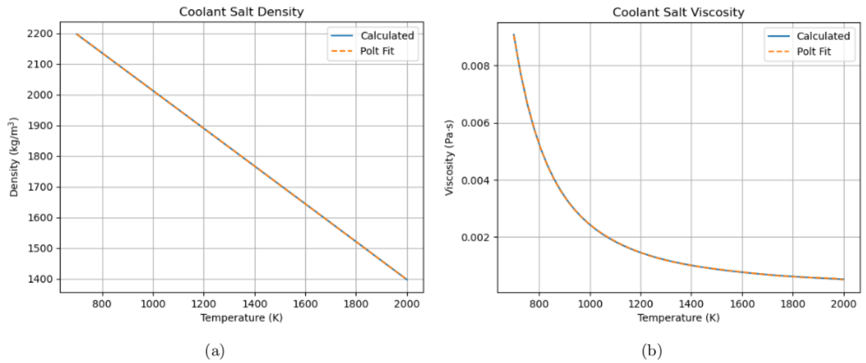
Matthew

Optimized Coolant Salt

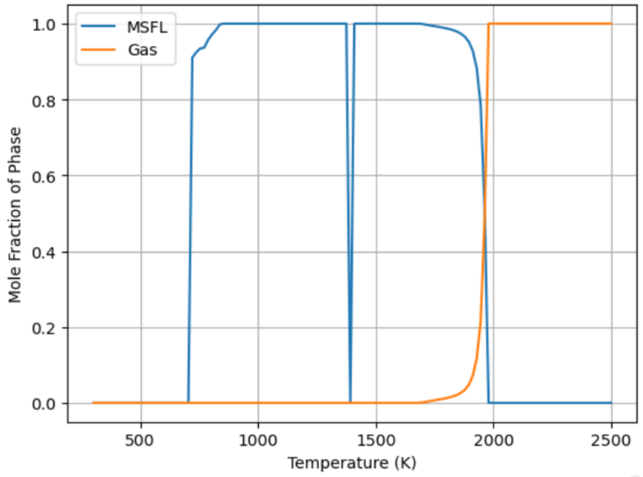
- Final melting and boiling points
  - T_m = 721 K
  - T_b = 1980 K
- Solution had very small mole fraction of CaF_2, so excluded from final solution
- Objective function value: **-42548.0**
  - only marginally higher than guess salt, but guess salt was fine tuned
  - Common choice of coolant: LiF-NaF-KF (46.5-11.5-42.0 mole %) gives objective value of **-36979.6**
  - Salt chosen by Scott: ZrF4 -NaF-KF (42-10-48 mole %) gives **-36558**



Thermophysical Properties

Endmember	Mole Fraction (%)
NaF	13.2
ZrF_4	3.2
KF	41.09
LiF	41.77

Final Coolant Salt Composition



Coolant Salt Phase Transition

Fuel Salt Optimization

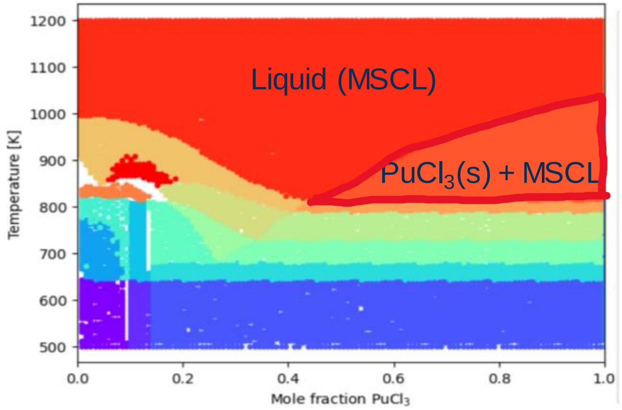
- Assuming minimal natural convection *within* the fuel tube, heat transfer dominated by *conduction*
- Objective function: Thermal conductivity
- Constraints:
  - *Maximum* melting temperature: 800 K
  - *Minimum* boiling point: 2000 K
- Actinide fraction and composition constrained by neutronics:
- 55 mole% of fuel is: $PuCl_3$ -$NdCl_3$ -UCl_3 (39.8-0.2-15.0 mole %)
  - Remaining 45 mole% unconstrained.
- Non-actinide components considered:
  - NaCl, KCl, $ZrCl_4$, AlCl

Optimized Fuel Salt

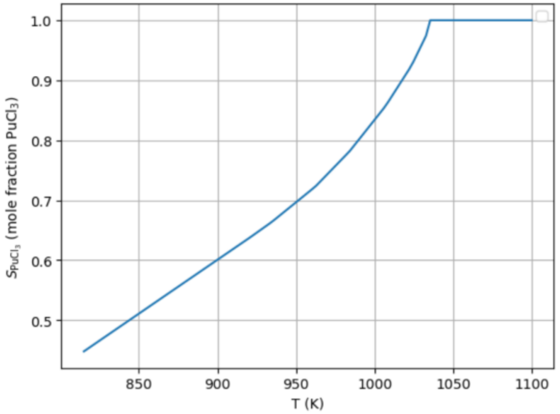
- Final melting and boiling points
  - **T_m = 788.6 K**
  - **T_b = 2164.4 K**
- *Large* operational temperature range
- Assumed 55 mole% actinide fraction would be soluble in fuel salt
  - Need to verify
  - Calculated $PuCl_3$ solubility from pseudo binary phase diagram using Thermochimica
- PuCl3 solubility most constraining
- 39.8 mole% soluble at all temperatures above melting point

Endmember	Mole Fraction (%)
NaCl	25.0
KCl	14.1
$ZrCl_4$	5.9
$PuCl_3$	39.8
$NdCl_3$	0.2
UCl_3	15.0

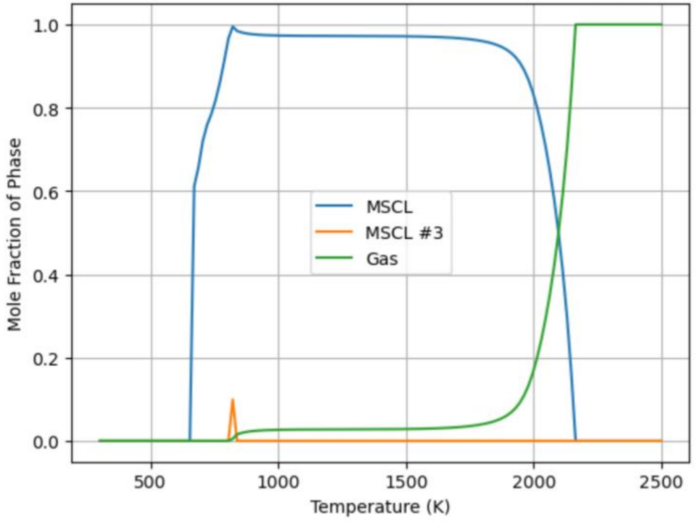
Final Fuel Salt Composition



$PuCl_3$ Pseudo-Binary Phase Diagram



$PuCl_3$ Solubility in Final Fuel Salt



Fuel Salt Phase Transition

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Thermal Hydraulics

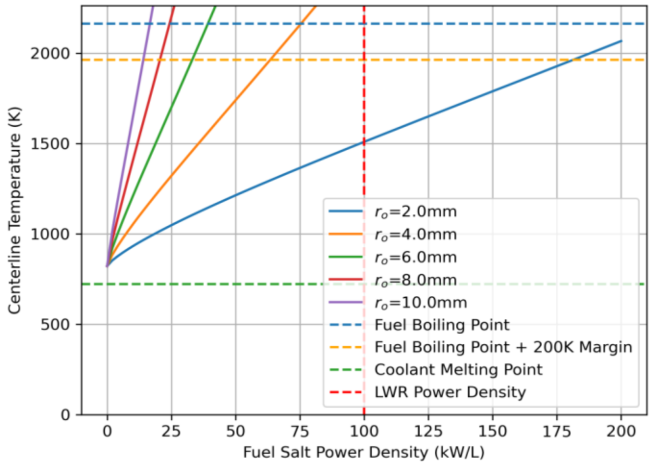
Background and Motivation

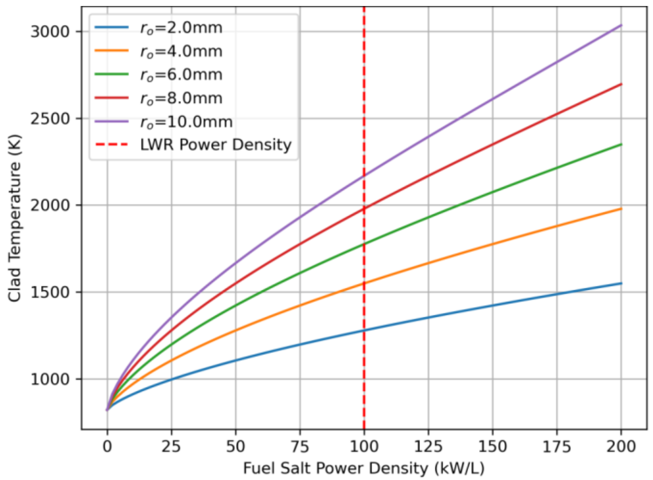
- This concept was originally considered during the Aircraft Reactor Experiment (ARE) but deemed infeasible from a TH standpoint
  - Modern CFD tools didn't exist
  - Convection within fuel tubes couldn't be relied on because reactor was designed to go on an aircraft
  - Adopted a design where fuel salt is pumped through a primary loop
    - Inspired the design of the MSRE
- Scott in [1] suggests this could not only be feasible, but that adequate heat transfer is possible with *purely* natural convection for a small modular variant of the design
  - Wanted to investigate this for our larger design
  - Would allow us to bypass the issues associated with the development of pumps for MSRs
    - Reliability and maintenance
    - Relying only on natural convection during operation allows for maximum passive safety

[1] Chapter 14: Static liquid fuel reactors: T. J. Dolan. Molten salt reactors and thorium energy. Woodhead Publishing, 2017.

The Need for Computational Fluid Dynamics (CFD)

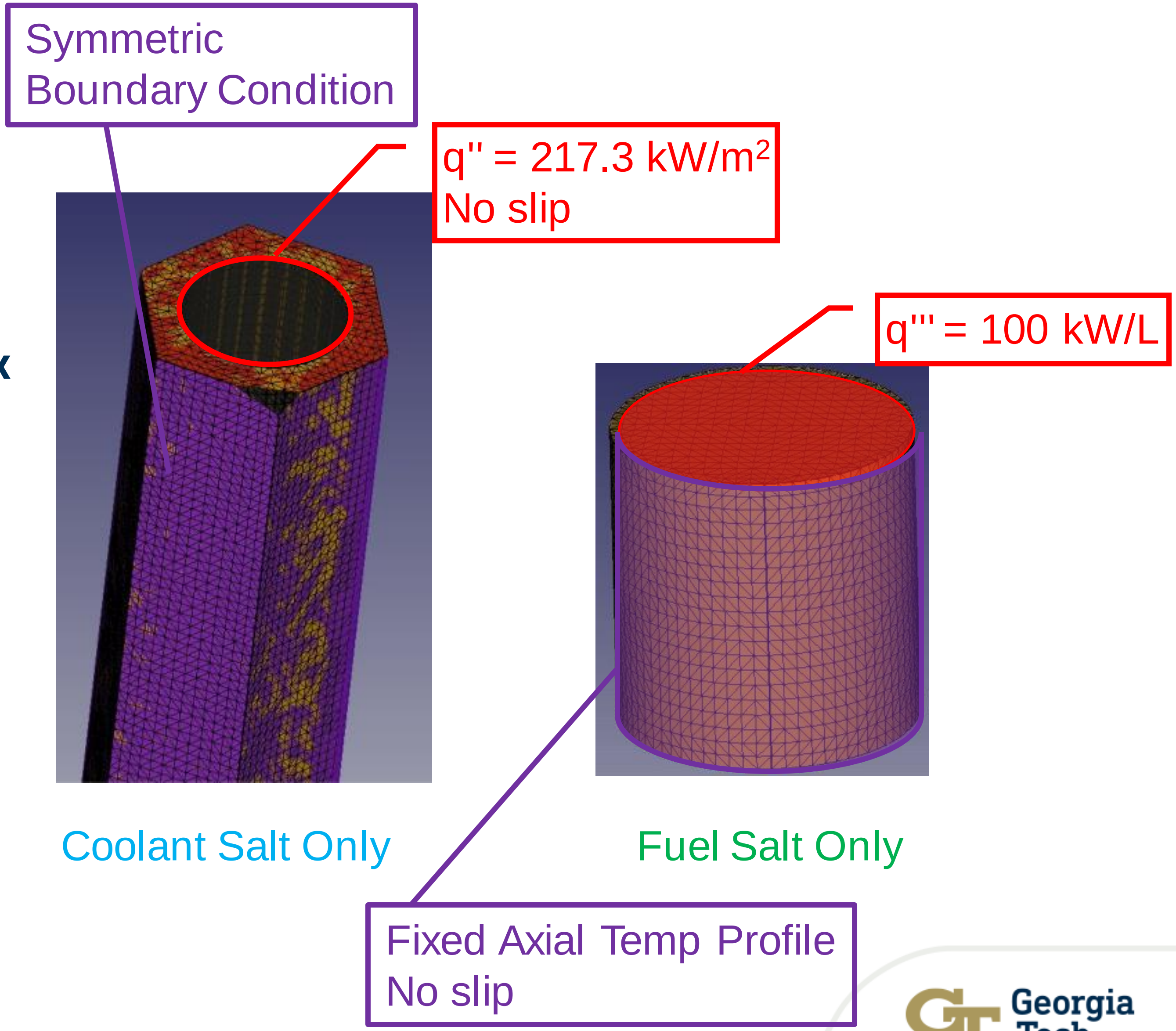
- Core height scoping with optimized coolant salt and constraints from neutronics calculations
  - P/D = 1.35, D = 11 mm
  - Power density of **100 kW/L**
  - $\Delta T/\Delta z$ = **19.2 K/m** -> ΔT = **57.7 K** for 3 m core
  - v = 0.986 m/s
- Did scoping using correlations for free convection and assumed pure conduction within the fuel tube
- Fuel centerline *and* clad temperature too high
  - Over fuel boiling point
  - Clad temp over code for stainless steel 316H
- *Very* conservative approximations
- Would have to **decrease power density by factor of 4** -> **~100 MW_t** for a near full-size core to get acceptable clad/fuel temps
  - Very difficult to remain competitive with LWRs
  - Otherwise, would need to consider advanced materials like SiC
- Need CFD to justify 100 kW/L power density, etc. otherwise, reactor infeasible





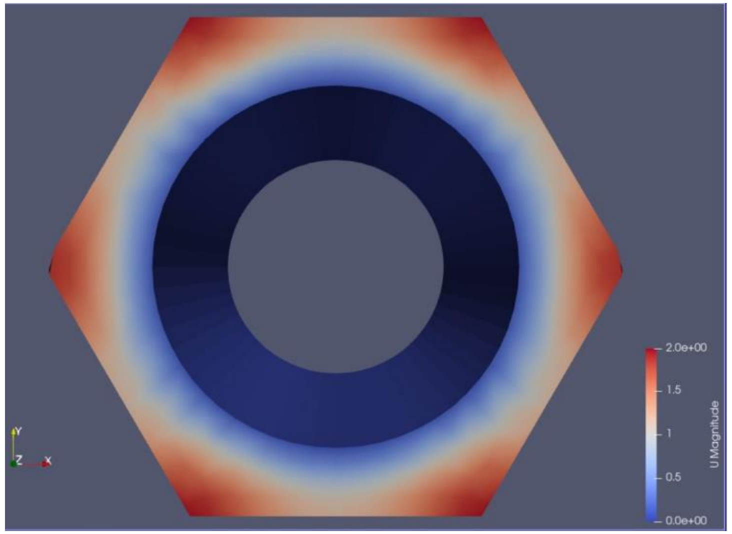
Methodology

- CFD calculations using OpenFOAM
  - Hard to learn but extremely general and powerful
- To avoid the need for a multiregion conjugate heat transfer solver, split into two steps:
  1. Calculate **coolant temperature distribution** in a hexagonal pincell with symmetric boundary conditions **given a fixed heat flux** (calculated from the given power density)
     - Inlet velocity of 1 m/s
     - Inlet temp 800 K
     - Gives clad temp on inner boundary
  2. Calculate **fuel temperature distribution** using clad temp from step 1 and volumetric generation of 100 kW/L
- Meshed with cfmesh
- Domain decomposition and parallel runs on Sawtooth
- Solved using BuoyantSimpleFoam (compressible heat transfer solver)

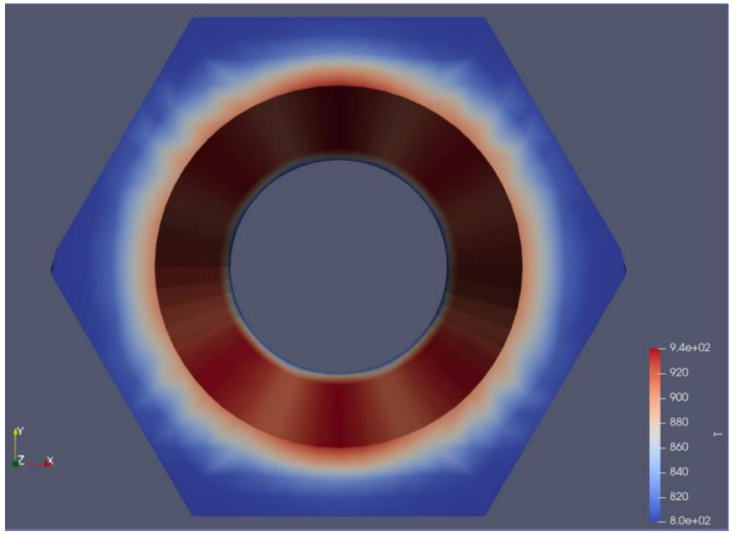


Coolant CFD Calculation

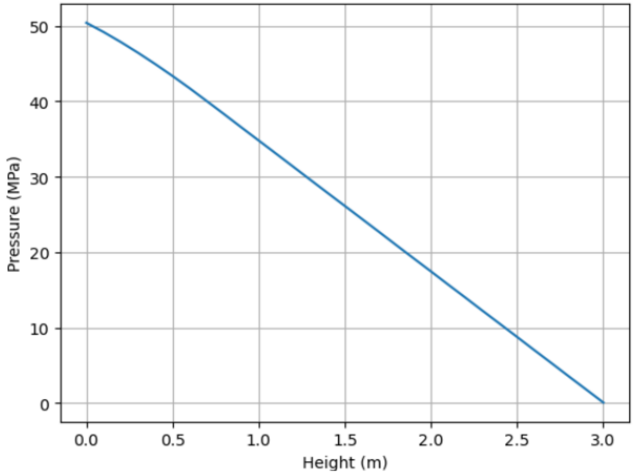
- Profiles show physically reasonable velocity and temperature profiles
- **Radially averaged fields**
  - Outlet temperature **835 K** -> ΔT = **35 K**
  - U_z = 1 m/s (balance between viscous friction and buoyant acceleration)
- **Max clad temp < 930 K (below code for SS 316H)**
- *Large* inlet pressure (~50 MPa)
  - Indicative of *large* pumping requirements?
  - Incorrect boundary conditions or ill-converged simulation?



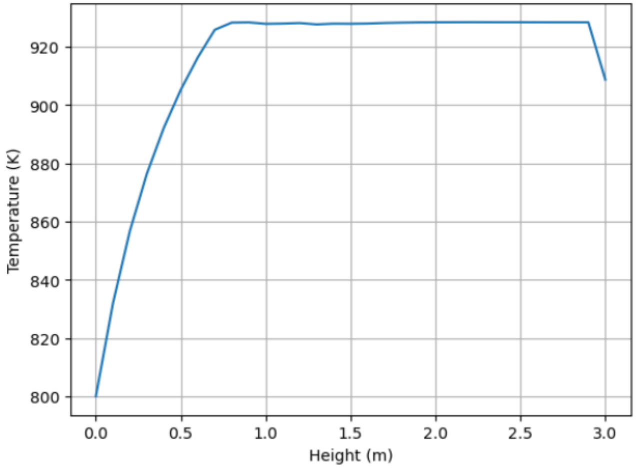
|**u**| profile (m/s)



Temperature Profile (K)



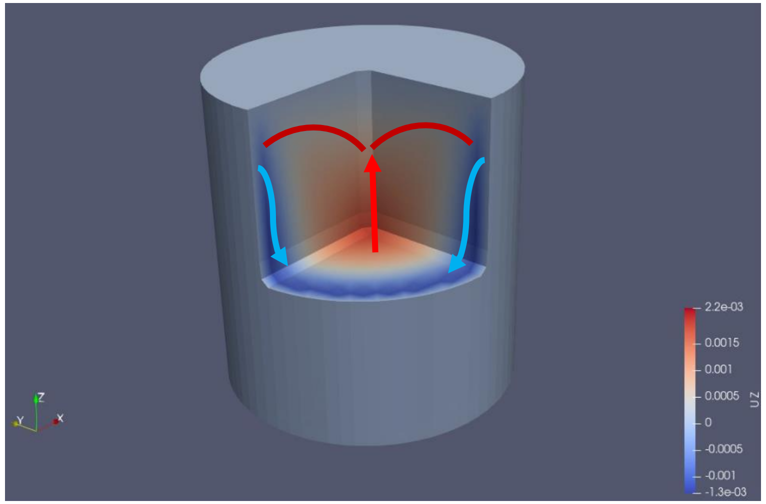
Axial Pressure Profile



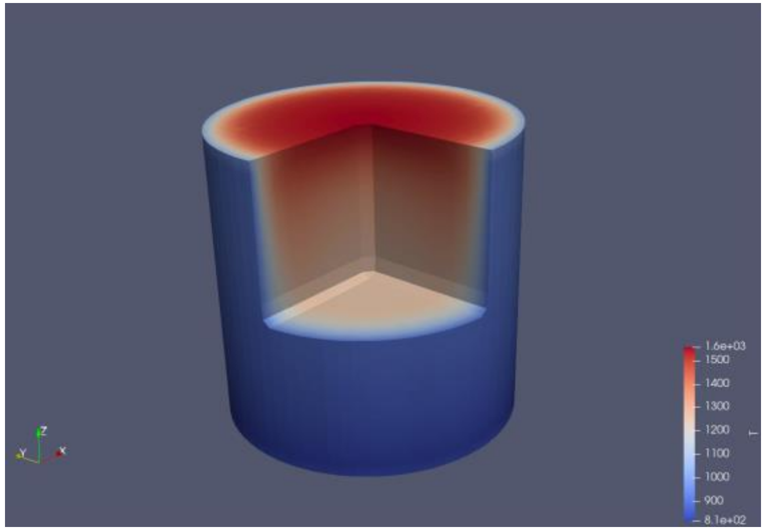
Axial Clad Temperature

Fuel CFD Calculation

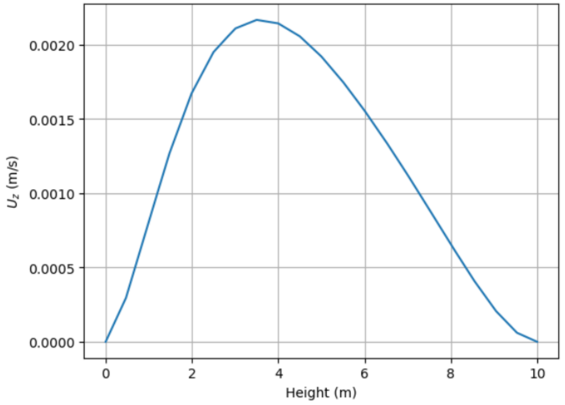
- Couldn't get full length (3m) model to converge
  - Imposed full temp profile on 10mm length model
  - Conservative approximation
- **Max fuel centerline ~1550 K (within 270 K margin of boiling point)**
- Accurate CFD decreased the fuel and clad temps by almost 1000 K
- Justifies 100 kW/L, D=11mm, and P/D=1.35



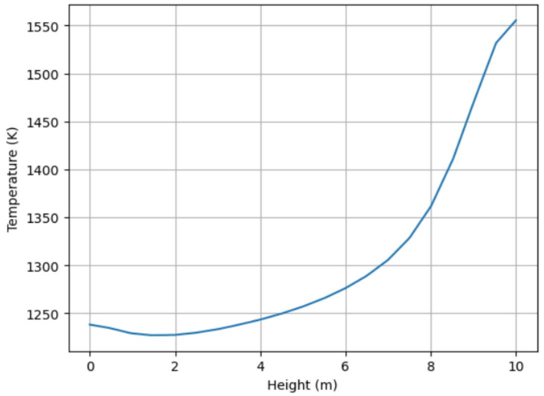
U_z profile (m/s)



Temperature Profile (K)



Axial centerline U_z profile



Axial Centerline Temperature

Safety

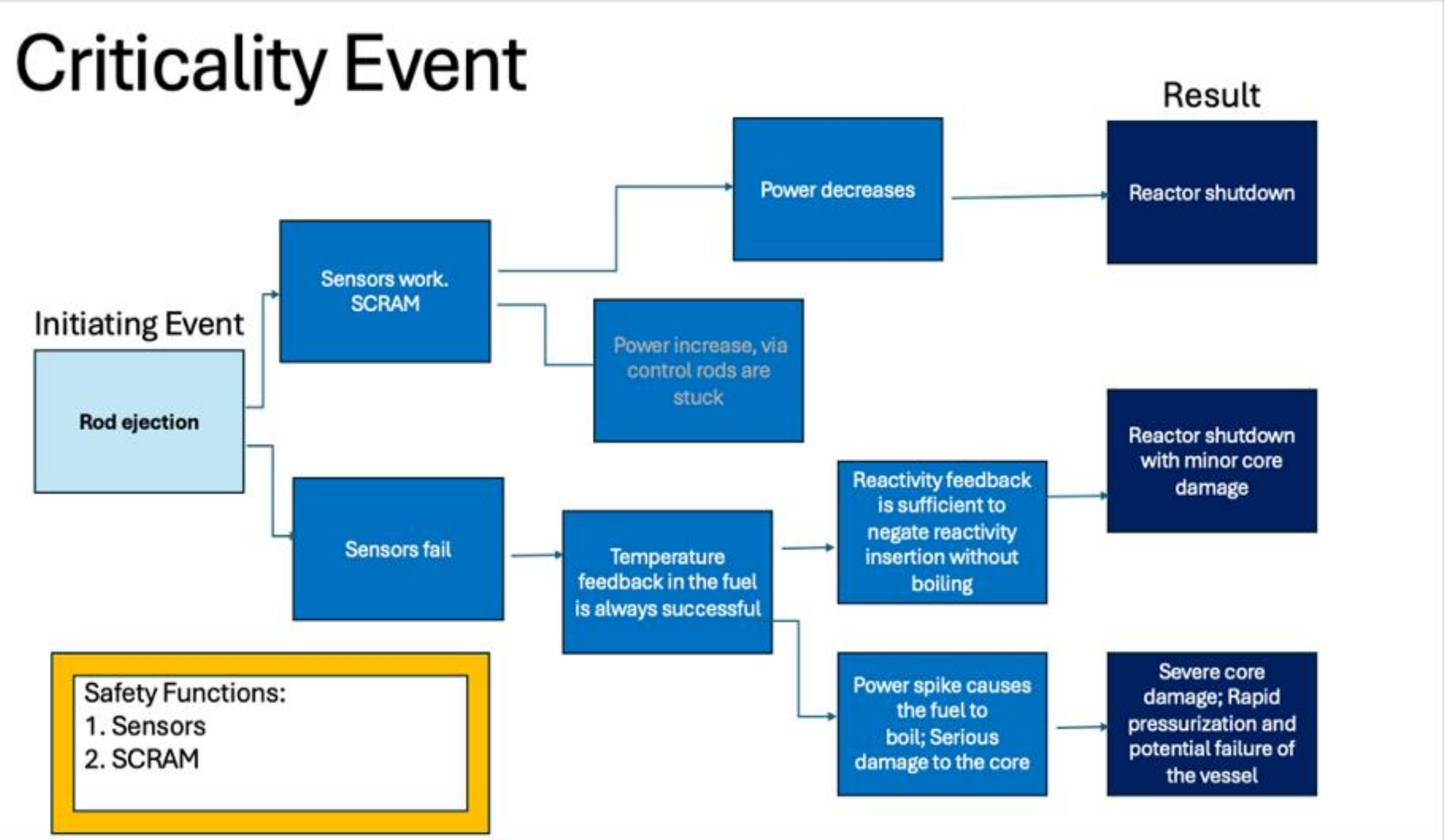
Event Tree Analysis

Event Tree Analysis (ETA) which is a useful schematic for visualizing failures and their consequences in a nuclear accident scenario.

In summary, the main purpose of the Event Tree analysis is to assist with reactor safety by:

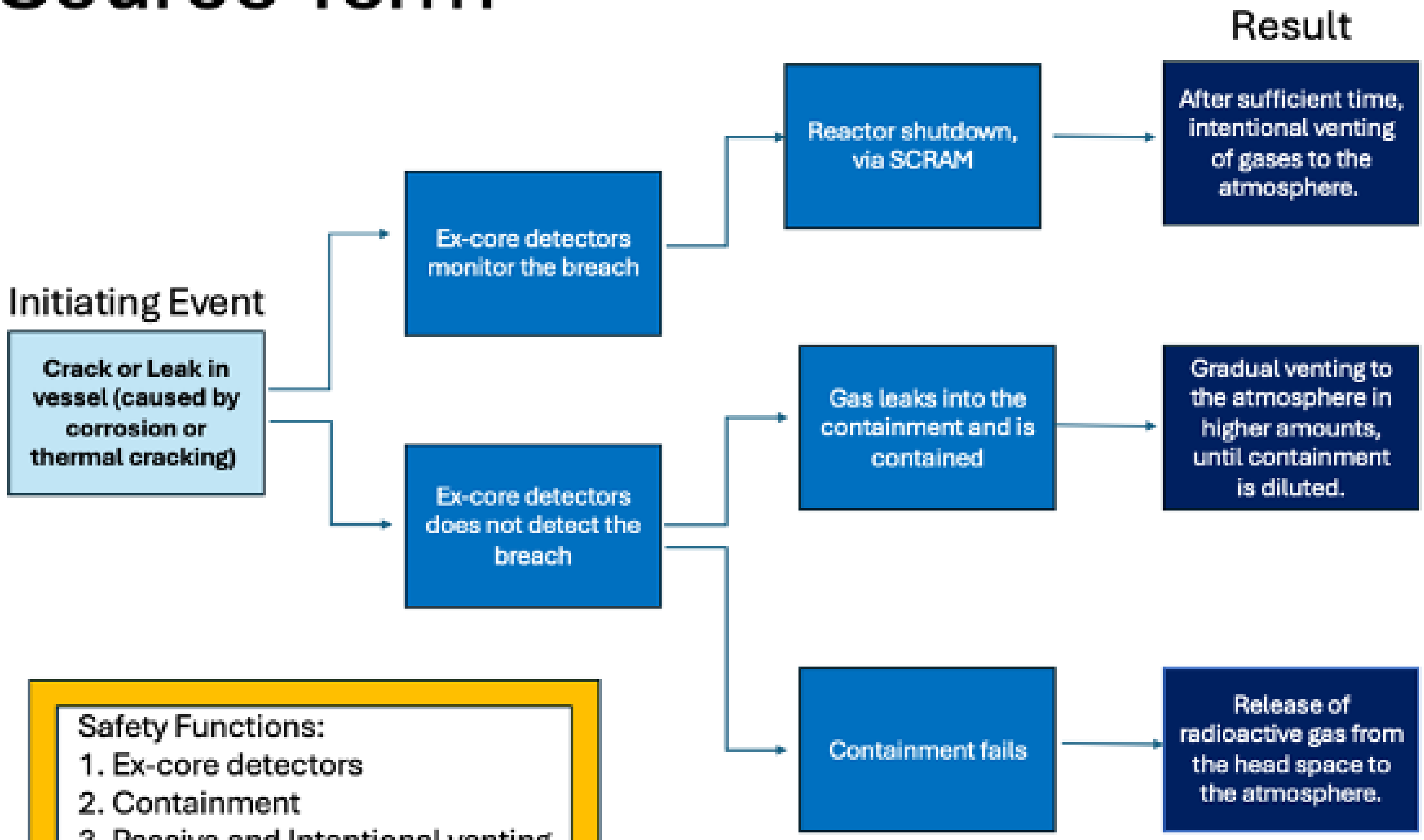
1. Identifying and understanding the possible sequences of events that could lead to common and severe accidents.
2. Assessing the probability of each sequence of events occurring.
3. Evaluating the potential consequences of each event.
4. Identifying vulnerabilities and weaknesses within the safety systems.
5. Facilitating the safety measures and emergency response plans.

Event Tree Analysis



Event Tree Analysis

Source Term



Safety Functions:
1. Ex-core detectors
2. Containment
3. Passive and Intentional venting

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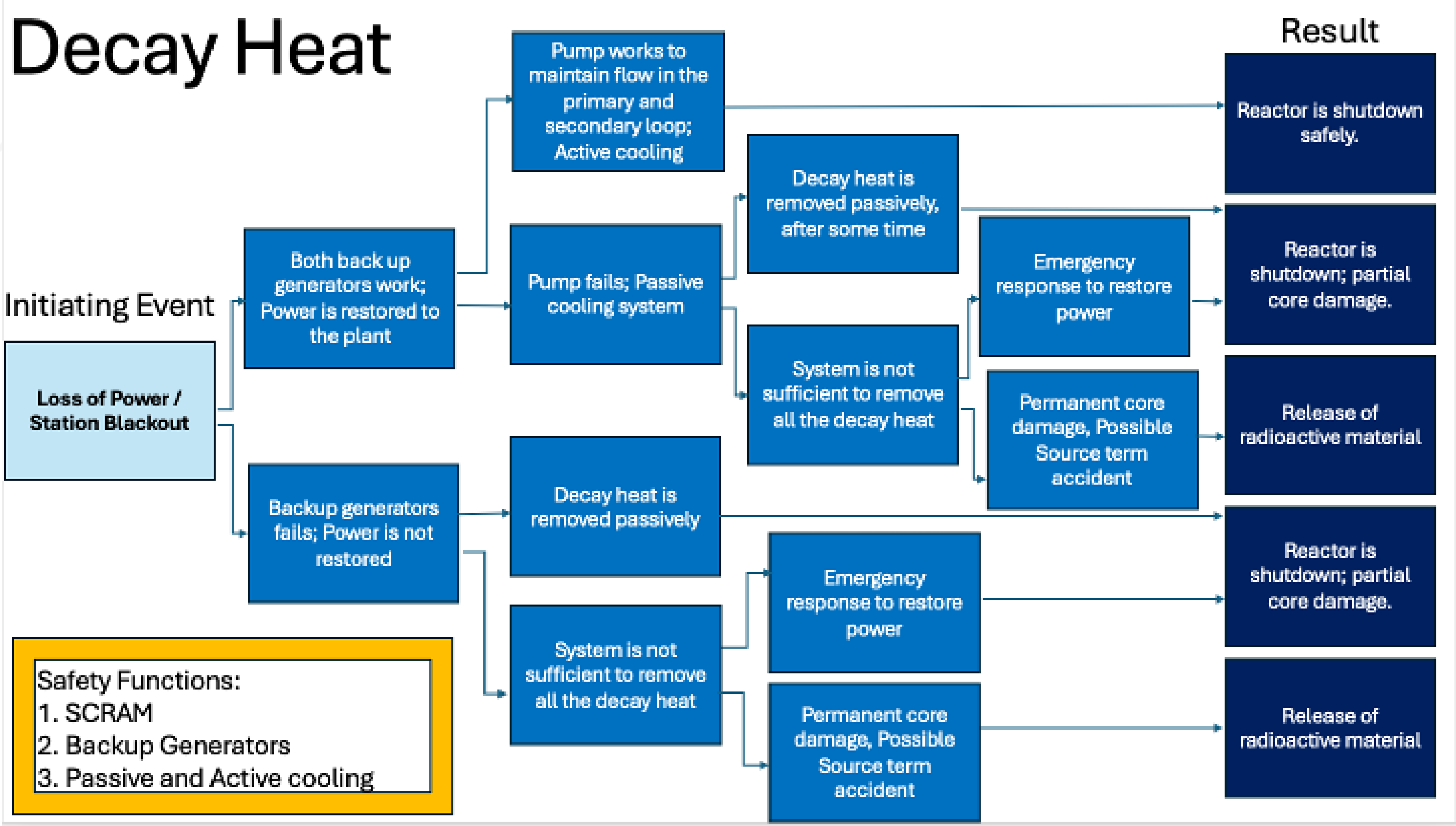
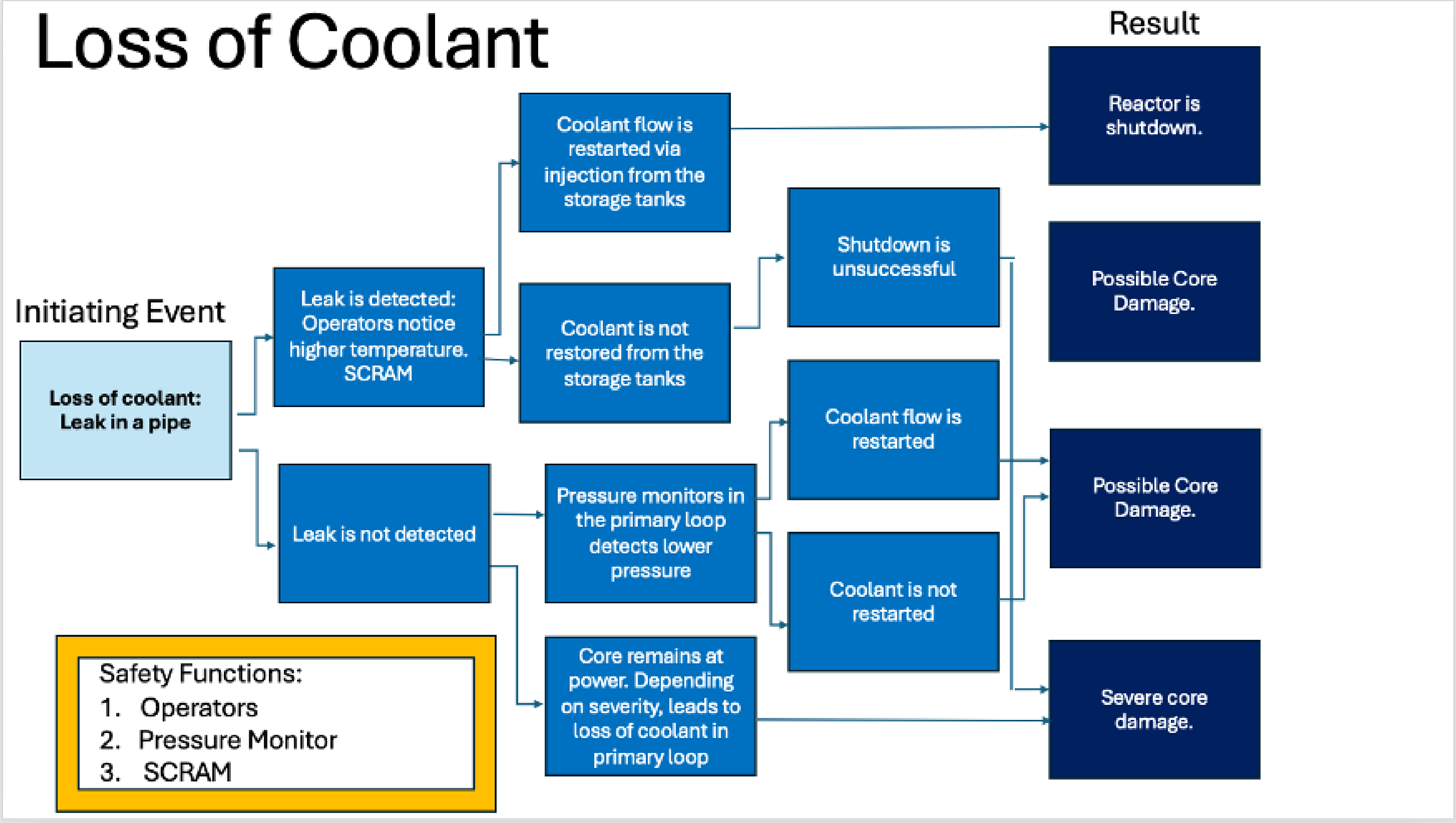
Event Tree Analysis



Event Tree Analysis



Safety Regulations

- For our design the regulations more crucial to pay attention to are those concerning:

Venting of Radioactive gases

- 40 CFR 190.10
- 40 CFR 61.92

§190.10 Standards for normal operations.

Operations covered by this subpart shall be conducted in such a manner as to provide reasonable assurance that:

(a) The annual dose equivalent does not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations.

(b) The total quantity of radioactive materials entering the general environment from the entire uranium fuel cycle, per gigawatt-year of electrical energy produced by the fuel cycle, contains less than 50,000 curies of krypton-85, 5 millicuries of iodine-129, and 0.5 millicuries combined of plutonium-239 and other alpha-emitting transuranic radionuclides with half-lives greater than one year

§ 61.92 Standard.

Emissions of radionuclides to the ambient air from Department of Energy facilities shall not exceed those amounts that would cause any member of the public to receive in any year an effective dose equivalent of 10 mrem/yr.

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Safety Regulations

Radioactivity in the coolant

- 10 CFR 50.36a
- 10 CFR 20.1301

§ 50.36a Technical specifications on effluents from nuclear power reactors.

(a) To keep releases of radioactive materials to unrestricted areas during normal conditions, including expected occurrences, as low as is reasonably achievable, each licensee of a nuclear power reactor and each applicant for a design certification or a manufacturing license will include technical specifications that, in addition to requiring compliance with applicable provisions of § 20.1301 of this chapter, require that:

§ 20.1301 Dose limits for individual members of the public.

(a) Each licensee shall conduct operations so that—

(1) The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contributions from background radiation, from any administration the individual has received, from exposure to individuals administered radioactive material and released under § 35.75, from voluntary participation in medical research programs, and from the licensee's disposal of radioactive material into sanitary sewerage in accordance with § 20.2003, and

(2) The dose in any unrestricted area from external sources, exclusive of the dose contributions from patients administered radioactive material and released in accordance with § 35.75, does not exceed 0.002 rem (0.02 millisievert) in any one hour.

(b) If the licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

(c) Notwithstanding paragraph (a)(1) of this section, a licensee may permit visitors to an individual who cannot be released, under § 35.75, to receive a radiation dose greater than 0.1 rem (1 mSv) if—

(1) The radiation dose received does not exceed 0.5 rem (5 mSv); and

(2) The authorized user, as defined in 10 CFR Part 35, has determined before the visit that it is appropriate.

(d) A licensee or license applicant may apply for prior NRC authorization to operate up to an annual dose limit for an individual member of the public of 0.5 rem (5 mSv). The licensee or license applicant shall include the following information in this application:

(1) Demonstration of the need for and the expected duration of operations in excess of the limit in paragraph (a) of this section;

(2) The licensee's program to assess and control dose within the 0.5 rem (5 mSv) annual limit; and

(3) The procedures to be followed to maintain the dose as low as is reasonably achievable.

(e) In addition to the requirements of this part, a licensee subject to the provisions of EPA's generally applicable environmental radiation standards in 40 CFR part 190 shall comply with those standards.

(f) The Commission may impose additional restrictions on radiation levels in unrestricted areas and on the total quantity of radionuclides that a licensee may release in effluents in order to restrict the collective dose.

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Basic Characteristics for MSR Safety pertaining to Our Design:

1. The primary and secondary systems have pressures lower than 5 bars, therefore there is no danger of accidents due to high pressure which could cause system destruction or salt leakage.
2. The fuel and coolant salts are chemically inert, so therefore they would not ignite if exposed to the air.
3. The boiling point of the fuel salt is about 1670K or much more, so long that it is much higher than the operation temperature 973 K. Therefore, the pressure of primary system cannot increase causing system destruction or damage.
4. The fuel rods on their own are subcritical, only when exposed to a full core geometry will the neutron economy be such that its critical
5. Molten salt reactors have a large negative temperature coefficient of the fuel salt which is much larger than the temperature coefficient of the coolant, which is slightly positive.
6. The delayed-neutron fraction in ^{233}U fission is smaller than that in ^{235}U, and half of the delayed neutrons is generated outside the core. However, it is within the controllable range and due to the longer neutron life, and large negative prompt temperature coefficient of fuel salt.
7. The excess reactivity and required control rod reactivity are sufficiently small, and the reactivity shift by control rods is small.
8. Gaseous fission products such as Kr and Xe are continuously removed from the fuel salt, so that the buildup is minimized.

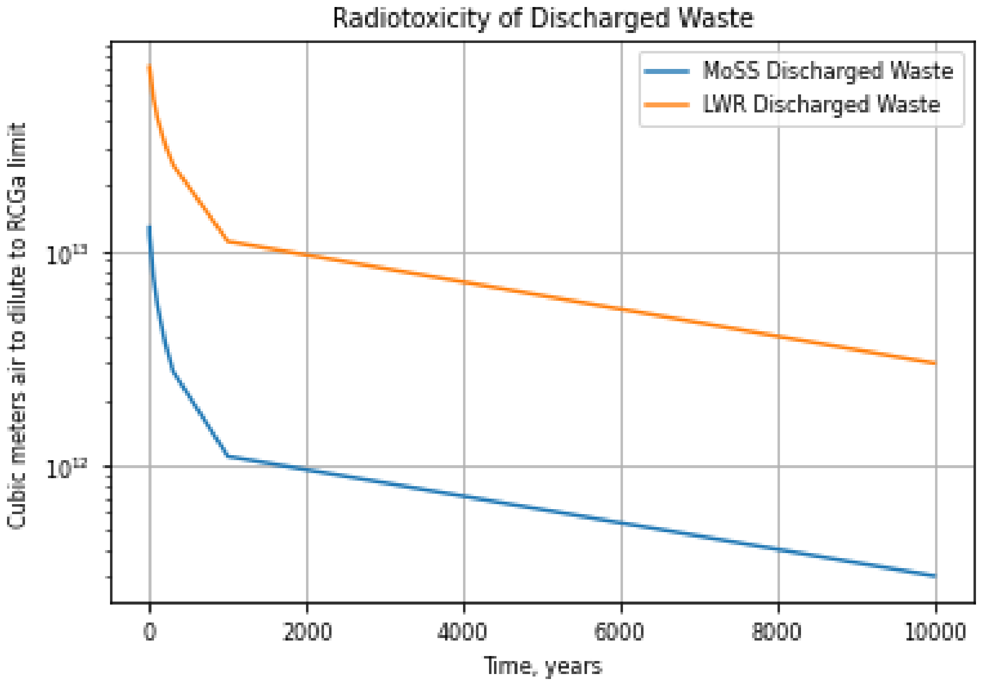
Waste Analysis

Decay Heat

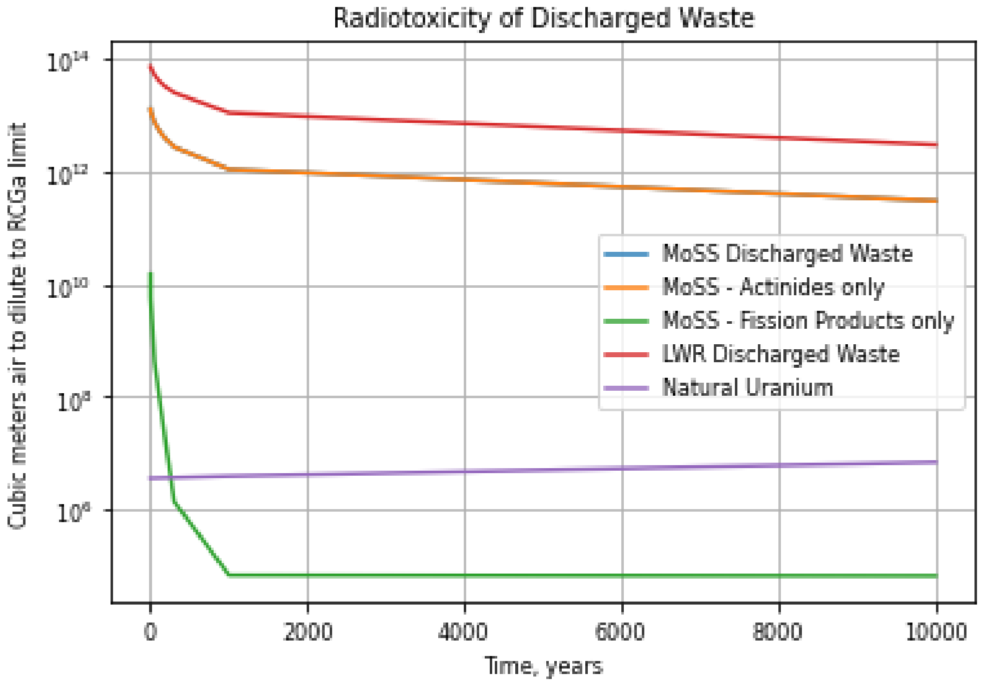
Total Decay Heat (W)	At Shutdown	1 year	50 years	100 years
Total Z1	1394000	38459	10346	7891.3
Total Z2	734110	32270	10422	7641.6
Total Z3	360320	23456	9054.5	6256.6
Total Core	**2488430**	**94185**	**29822.5**	**22059.5**
	0.58% Active Power	**3.8% Shutdown Heat**	**1.2% Shutdown Heat**	**0.9% Shutdown Heat**

Radiotoxicity

Comparison of MoSS to LWR Waste



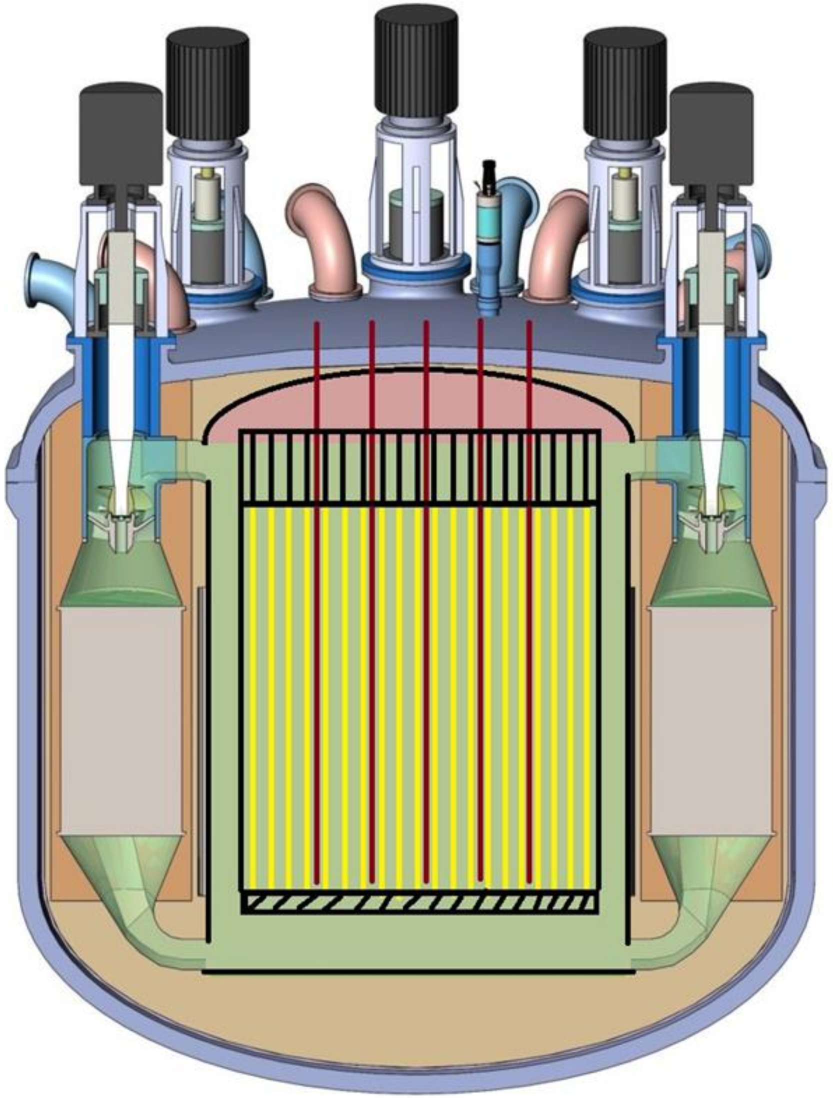
MoSS Components vs LWR vs Nat Uranium



Economics

Approach

- No MSR's in operation or being constructed making economic analysis difficult
- Top-down approach
- Breakdown into four cost components:
  - Capital (Overnight & IDC)
  - Operation & Maintenance
  - Fuel
  - Decommissioning (Decomposition & Waste Management)



Capital Cost

Overnight Capital

- MIT cost estimation for ORNL 1000 MWe MSR

Direct Costs	$/kWe
Reactor Plant Equip.	870
Turbine Plant Equip.	440
Electrical Plant Equip.	226
Misc. Plant Equip.	159
Heat Rejection System	61
Structures & Improvements	659
Total	2455

Interest During Construction (IDC)

- Construction schedule modeled after GE-Hitachi-Toshiba ABWR

Plant	Construction Years	Annual Capital Flow %
ABWR	4	5.4, 20.5, 59.62, 14.6

- Assumptions
  - 8% discount rate
  - 90% capacity factor
  - 38% thermal efficiency
- Further calculations including capital recuperation and present value factors yield a total capital COE of **27.92 $/MWh**

J. C. Palacios, G. Alonso, R. Ramirez, A. Gomez, J. Ortiz, and L. C. Longoria. Levelized costs for nuclear, gas and coal for electricity, under the mexican scenario. Technical report, Instituto Nacional de Investigaciones Nucleares (MX), 2004.

O&M Costs

MOLTEX

- MOLTEX received a cost estimation from Atkins Ltd

Fixed O&M ($/MWh)	Variable O&M ($/MWh)
12.81	2.13

- Assumes the same O&M cost of a UK PWR
  - Large overestimation considering the intrinsic safety mechanisms of MoSS Reactor

GE-Hitachi-Toshiba ABWR

- Similar core size
- Much lower operating pressure
- Likely still an overestimation
- Total O&M COE: **8.19 $/MWh**

T. W. https://www.techniquewebdesign.co.uk. Costs less - Moltex Energy — moltexenergy.com. https://www.moltexenergy.com/costs-less/. [Accessed 20-04-2024].

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Fuel and Decommissioning Costs

Fuel

- Obtaining legacy LWR fuel waste
- Charging reactor companies for fuel fabrication and transportation costs entirely
- Net fuel costs = 0$

Decommissioning

- Decommissioning cost estimations are relatively consistent for various MSR designs
- ORNL 1000 MWe MSR cost estimation broken down in two categories

Component	COE ($/MWh)
Waste Disposal	1.0
Decomposition	0.4

B. Mignacca and G. Locatelli. Economics and finance of molten salt reactors. Progress in Nuclear Energy, 129:103503, 2020. ISSN 0149-1970. doi: https://doi.org/10.1016/j.pnucene.2020.103503. URLhttps://www.sciencedirect.com/science/article/pii/S0149197020302511.

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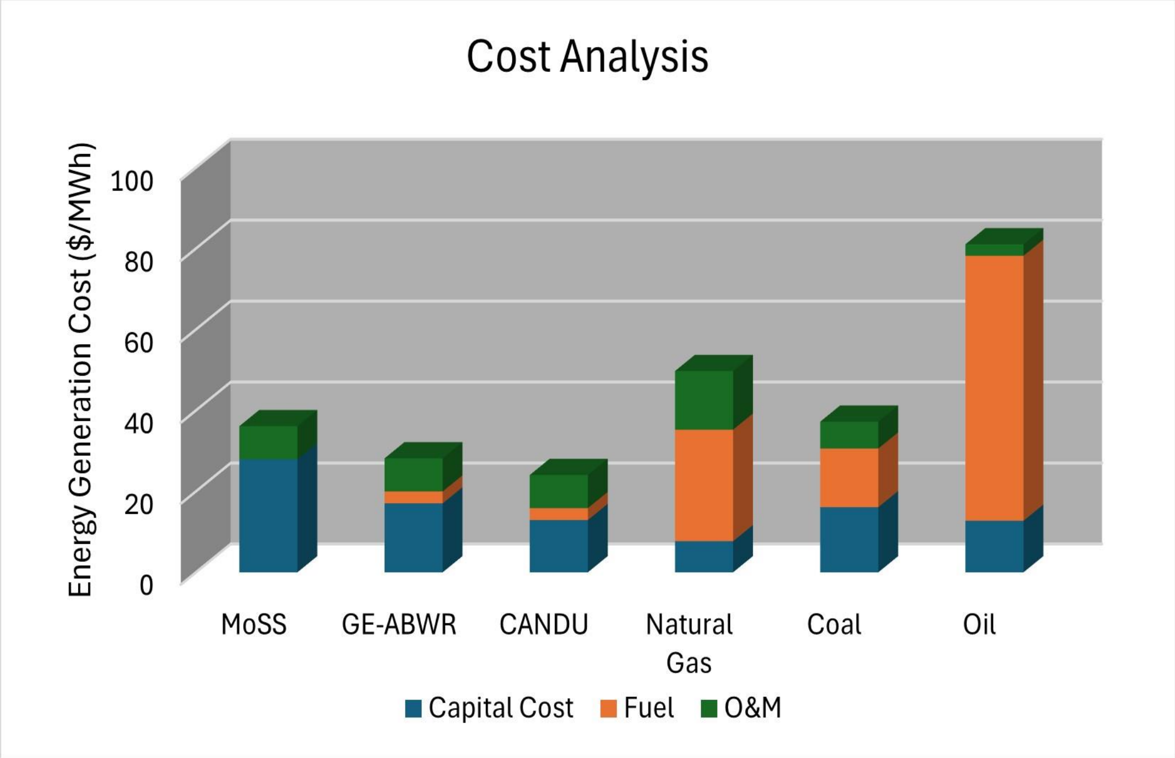
Analysis



Total COE ($/MWh)	Revenue ($/MWh)	Profit ($/MWh)
37.514	126.60	89.086

- Revenue was determined from the average electricity rate in Georgia (**12.26 ¢/kWh**)
- Total cost per MWh is comparable to similar sized LWR's
    - Much smaller total profit considering our low operating power (426 MWth)
    - **120,749,760 $/year profit**

J. C. Palacios, G. Alonso, R. Ramirez, A. Gomez, J. Ortiz, and L. C. Longoria. Levelized costs for nuclear, gas and coal for electricity, under the mexican scenario. Technical report, Instituto Nacional de Investigaciones Nucleares (MX), 2004.

Electricity Rates (Updated April 2024) &x2014; Electric Choice — electricchoice.com. https://www. electricchoice.com/electricity-prices-by-state/. [Accessed 20-04-2024].

Conclusions

Matthew, Nathan, Jaden, Alicia

Summary of Findings

- Achieves criticality utilizing actinides from light water reactor discharged fuel
- Maintains 2-year fuel cycles in a 3-batch core reload pattern
- Decreases radiotoxicity of fuel input by 82% immediately after discharge
- Proved that the fuel salt has a highly negative temperature reactivity coefficient
- Verified that temperatures are kept below constraints for reasonable inlet velocity
- Mitigated corrosion of structural materials using zirconium metal
- Reached profitability with COE comparable to similar sized LWR’s
- Performed event-tree analysis to understand relevant safety systems

Questions?