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Initiating Event Analysis of a Lithium Fluoride Thorium Reactor

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INITIATING EVENT ANALYSIS OF A LITHIUM FLUORIDE THORIUM REACTOR

by

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ABSTRACT

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The primary purpose of this study is to perform an Initiating Event Analysis for a Lithium Fluoride Thorium Reactor (LFTR) as the first step of a Probabilistic Safety Assessment (PSA). The major objective of the research is to compile a list of key initiating events capable of resulting in failure of safety systems and release of radioactive material from the LFTR.

Due to the complex interactions between engineering design, component reliability and human reliability, probabilistic safety assessments are most useful when the scope is limited to a single reactor plant. Thus, this thesis will study the LFTR design proposed by Flibe Energy. An October 2015 Electric Power Research Institute report on the Flibe Energy LFTR asked "what-if?" questions of subject matter experts and compiled a list of key hazards with the most significant consequences to the safety or integrity of the LFTR. The potential exists for unforeseen hazards to pose additional risk for the LFTR, but the scope of this thesis is limited to evaluation of those key hazards already identified by Flibe Energy.

These key hazards are the starting point for the Initiating Event Analysis performed in this thesis. Engineering evaluation and technical study of the plant using a literature review and comparison to reference technology revealed four hazards with high potential to cause reactor core damage. To determine the initiating events resulting in realization of these four hazards, reference was made to previous PSAs and existing NRC and EPRI initiating event lists. Finally, fault tree and event tree analyses were conducted, completing the logical classification of initiating events. Results are qualitative as opposed to quantitative due to the early stages of system design descriptions and lack of operating experience or data for the LFTR.

In summary, this thesis analyzes initiating events using previous research and inductive and deductive reasoning through traditional risk management techniques to arrive at a list of key initiating events that can be used to address vulnerabilities during the design phases of LFTR development.

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NOMENCLATURE

ARE – Aircraft Reactor Experiment

BWR – Boiling Water Reactor

EPRI – Electric Power Research Institute

GFR – Gas-cooled Fast Reactor

GIF – Generation IV Forum

I&C – Instrumentation and Control Circuitry

IE – Initiating Event

LFR – Lead-cooled Fast Reactor

LFTR – Lithium Fluoride Thorium Reactor (A specific application of MSR)

LWR – Light Water Reactor (Generic name encompassing both PWR and BWR)

MSBR – Molten Salt Breeder Reactor (Oak Ridge National Laboratory)

MSR – Molten Salt Reactor

MSRE – Molten Salt Reactor Experiment (Oak Ridge National Laboratory)

NRC – Nuclear Regulatory Commission

ORNL – Oak Ridge National Laboratory

PSA – Probabilistic Safety Assessment

PWR – Pressurized Water Reactor

QRA – Quantitative Risk Analysis

SCWR – Supercritical Water-cooled Reactor

URW – Uncontrolled Rod Withdrawal

VHTR – Very High Temperature Gas Reactor

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

CHAPTER 1

INTRODUCTION

Ever since Enrico Fermi and his fellow engineers brought the Chicago Pile (CP-1) to criticality in December 1942, nuclear fission and its application in electrical power generation has been a source of intrigue, inspiration and controversy. The world's first nuclear reactor, CP-1 consisted of a rudimentary stack of uranium metal and uranium oxide fuel bricks interspersed between graphite blocks designed to absorb neutrons. The experiment was assembled beneath the west stands of Stagg Field at the University of Chicago as part of the Manhattan Project (Koppes n.d.). Called "a crude pile of black bricks and wooden timber" by Fermi (Kelly 2007, 83), the reactor was controlled by withdrawing neutron absorbent rods, allowing the neutrons to cause fission in the uranium fuel, which resulted in the world's first sustained nuclear reaction.

In the decades that followed, nuclear fission reactions would be used in many diverse ways including heat production for power generation; weapons applications; and medical, chemical and metallurgical studies. The first generation of prototype nuclear reactors gave birth to more stable and safer commercial power reactors. For nearly 60 years, nuclear power was dominated by the use of light-water cooled reactors (LWR). Specifically, pressurized water reactors (PWR) and boiling water reactors (BWR) using light water (H_2O) as both the coolant and neutron moderator were the industry standard. This momentum behind PWR and BWR technology led to streamlined licensure and operation at the expense of exploring alternative technologies for nuclear fission.

By the early-2000s, after several iterations of technological advances to PWR and BWR technology, scientists and engineers from around the world convened a forum to discuss the future of nuclear fission and its role in power generation. In response to growing energy demand and in light of continued research demonstrating the harmful effects of fossil fuel use, the turn of the 21st century saw a renewed interest in the development of advanced nuclear reactor technologies as viable and competitive sources of electrical power. Chartered in mid-2001, the Generation IV

International Forum (GIF) represents a collective of 13 countries in which nuclear power plants are seen as vital for meeting future energy demands (World Nuclear Association 2016). After significant deliberation and review of countless proposed reactor designs, the GIF announced the selection of six very promising designs. Selection criteria demanded that the proposed reactor designs be "clean, safe and cost-effective means of meeting increased energy demands on a sustainable basis, while being resistant to diversion of materials for weapons proliferation and secure from terrorist attacks" (World Nuclear Association 2016).

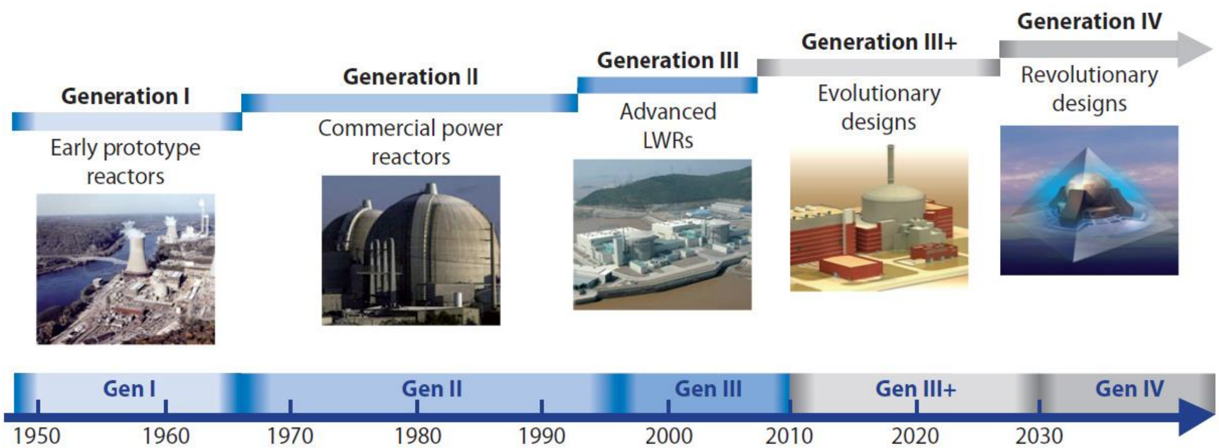

Figure 1. The Evolution of Nuclear Power Plants from Generation I to Generation IV (World Nuclear Association 2016)

Ultimately, the goal of the GIF is to direct international efforts in research and development of these advanced nuclear reactors in order to replace the aging PWR and BWR infrastructure beginning as early as 2020-2030. A brief description of each of the six advanced nuclear reactor technologies selected by the GIF is provided below.

- Gas-cooled Fast Reactors (GFR): The GFR is a helium-cooled reactor reliant on fast-spectrum neutrons for fission of solid uranium fuel. The fuel will be assembled in hexagonal elements consisting of ceramic-clad, mixed-carbide-fueled pins within a ceramic hexagonal tube. Helium gas will be circulated through the core of solid fuel where it is heated to 850°C. At the reactor outlet, the primary helium coolant rejects heat to a secondary helium-nitrogen mixture, which in turn drives a closed cycle gas turbine. The waste heat from the gas turbine heats a steam generator, which drives a steam turbine, resulting in a combined power cycle common in natural gas-fired power plants (The Generation IV International Forum 2017).

- Lead-cooled Fast Reactors (LFR): The LFR is a molten lead or lead-bismuth eutectic-cooled reactor reliant on fast-spectrum neutrons for fission of solid uranium or solid actinides from spent LWR fuels. The molten lead or lead-bismuth eutectic (44.5% lead, 55.5% bismuth) primary coolant rejects heat to a closed cycle carbon dioxide gas turbine through heat exchangers. Waste heat from the turbine drives a steam generator and steam turbine in a combined cycle similar to that described for the GFR. Because of its high boiling point, the primary coolant in the LFR need not be pressurized. This low-pressure reactor obviates the need for high-strength pressure vessels required in legacy LWRs and some other proposed advanced reactors (The Generation IV International Forum 2017).
- Sodium-cooled Fast Reactors (SFR): The SFR is a liquid sodium-cooled reactor reliant on fast-spectrum neutrons for fission of solid uranium-plutonium fuel, oxide or metal fuel, or uranium-plutonium-actinide-zirconium fuel (dependent on the reactor size). Liquid sodium is circulated through the core where temperatures are raised to ~550°C. In the primary heat exchangers, the lead coolant rejects heat to an intermediate sodium loop before the secondary sodium heats a closed gas cycle to drive a turbine power conversion system. Similar to the LFR, the SFR primary coolant remains liquid at low

pressures; therefore, this design does not require any pressure vessels required in legacy LWRs (The Generation IV International Forum 2017).

- Supercritical Water-cooled Reactors (SCWR): The SCWR is a high-temperature, high-pressure light water-cooled reactor that operates above the thermodynamic critical point of water (374°C, 22.1 MPa). Similar to a BWR, the SCWR is a once-through steam cycle in which subcooled liquid water is raised to temperatures and pressures that constitute superheated steam within the core. The superheated steam is used directly to drive a steam turbine power conversion system. Exhausted steam is condensed and returned to the core using a feed pump to recommence the cycle. The SCWR offers significantly improved thermal efficiencies over legacy LWRs due to the high temperatures (500-625°C) but suffers from safety concerns with the associated high pressures (>20 MPa). Still, coal-fired industry has significant operating experience using superheated steam in power generation and many technologies may be adapted for use in the SCWR (The Generation IV International Forum 2017).
- Very High-temperature Gas Reactors (VHTR): The VHTR is a helium-cooled, graphite moderated reactor reliant and thermal-spectrum neutrons to fission various fuel sources. Two types of core are being explored: the prismatic fuel block and pebble bed core, both of which can use open cycle uranium fuel, or closed cycle uranium-plutonium, thorium-uranium or mixed-oxide fuel (MOX). The VHTR is unique among Generation IV designs as it is primarily dedicated to cogeneration of electrical power and hydrogen gas. The hydrogen gas is extracted via thermo-chemical or electro-chemical processes driven by the extremely high temperatures of the helium gas (~1000°C). Of course, the high temperature of the outlet gas yields a high primary system pressure and necessitates pressure vessels to contain the reactor core and primary loops. The power conversion system can be either closed cycle gas turbine or steam turbine depending on

the final outlet temperature of the primary helium (The Generation IV International Forum 2017).

- Molten Salt Reactors (MSR): The MSR is a lithium-fluoride or lithium-beryllium-fluoride salt cooled reactor reliant on fast- or thermal-spectrum neutrons to fission liquid uranium fuel suspended in the coolant. In thermal-spectrum designs, the graphite moderator is positioned in the core to thermalize neutrons to facilitate fission. In all designs, MSRs stand out as unique in their use of liquid fuel suspended in the primary coolant, instead of solid fissile fuel positioned in the reactor core. Heat generated in the molten salt coolant is exchanged to an intermediate salt loop, which then drives a supercritical CO_2 closed Brayton-cycle power conversion system. Because the proposed salts (lithium-fluoride or lithium-beryllium-fluoride) have high boiling points (1676°C) at atmospheric pressures, the MSR is designed to operate at low pressures similar to LFRs and SFRs (The Generation IV International Forum 2017). Additionally, because the fissile fuel material is homogenously distributed in the primary coolant and not concentrated in a solid matrix within the reactor core, the concept of "core meltdown" due to loss of cooling is obsolete. Once circulation through the reactor core stops, fission will not persist because the fuel is suspended within the coolant and not concentrated in the core. This unique design feature is at the heart of the inherent safety of MSRs.

REACTOR	COOLANT	FUEL	TEMPERATURE	PRESSURE
Gas-cooled Fast Reactor	Helium	Solid hexagonal uranium elements	850°C	90 Bar (9MPa) (Stainsby n.d.)
Lead-cooled Fast Reactor	Lead or Lead-Bismuth Eutectic	Solid uranium or actinides	480-570°C	Atmospheric (Alemberti, et al. 2014, 11)
Sodium-cooled Fast Reactor	Sodium	Solid U-Pu, MOX or U-Pu-Actinide	500-550°C	Atmospheric
Supercritical Water Reactor	Light Water (H_2O)	Solid uranium or plutonium	510-625°C	>22.1MPa
Very High Temperature Gas Reactor	Helium	Solid U-Pu, Th-U or actinides	900-1000°C	7 MPa (Oh, et al. 2016)
Molten Salt Reactor	Lithium Fluoride Salts	Liquid U-233 from Th-U fuel cycle	700-800°C	Atmospheric

Table 1. Comparison of Generation IV Advanced Nuclear Reactors
(The Generation IV International Forum 2017)

Of the six technologies selected for future research and development, four have significant operating experience in research applications. Of the four technologies with previous operating experience, one boasts a unique and highly desirable safety feature over all others. The Molten Salt Reactor stands apart as the only GIF proposal that abandons the traditional design of a "solid nuclear fuel core" and instead relies on dissolved fissile material into a molten salt coolant. The safety benefit of this design concept is the complete absence of risk of "nuclear meltdown" in the traditional sense. That is, the most dangerous risk scenario for traditional nuclear reactors exists when cooling of the solid reactor core fails or is compromised. In this case, the solid nuclear fuel may overheat and begin to melt or deform, causing a geometry of fuel and other material whose nuclear fission characteristics are uncontrollable. If this occurs, the heat generated in the reactor core could result in failure of other structural materials and a release of radioactive fission products to the environment and public exposure to radiation. The risk of solid fuel meltdown is the basis for most public concern and was the mode of failure in Chernobyl's Reactor Four in 1986 and

Fukushima Daiichi in 2011. This basic description of a “nuclear meltdown” becomes obsolete in the Molten Salt Reactor because the nuclear fuel is not concentrated into solid elements in a reactor core but is evenly disbursed in the circulating coolant. The reactor core is simply a vessel whose geometry and materials enable fission of the uranium fuel suspended in the coolant. Once the salt leaves the core, the nuclear reaction stops and heat is rejected to intermediate salt loops and then to CO_2 which drives a gas turbine. In the event that the fuel salt overheats, a frozen plug of salt in the bottom of the reactor will melt away, draining the fuel salt into a subcritical collection tank where nuclear fission is impossible.

The unique quality of liquid nuclear fuel makes the LFTR both inherently safe and revolutionary in its method of employing nuclear fission. For this simple reason, Molten Salt Reactors and specifically the Lithium Fluoride Thorium Reactor were selected as the subject of this study. Flibe Energy’s LFTR is not, however, the first example of proposed MSR technology in the United States.

1.1 EARLY MOLTEN SALT REACTOR EXPERIENCE

The initial development and operation of molten salt reactors was performed by researchers at Oak Ridge National Laboratory following World War II. The Molten Salt Reactor Experiment (MSRE) and the Aircraft Reactor Experiment (ARE) represent the only two molten salt reactors ever built and operated in the United States.

In 1946 the United States Air Force initiated a program to develop a nuclear-powered airplane under contract with Fairchild Engineering and the Airplane Corporation. In the years that followed, heightened tensions of the Cold War drove the US Atomic Energy Council to establish the Aircraft Nuclear Propulsion (ANP) program at Oak Ridge National Laboratory (ORNL) in Tennessee. Two proposals were put forth, the first calling for air through the jet engine to directly cool fuel elements from the reactor, while the second called for an indirect cycle in which molten salt was heated in the reactor and then cooled by the flow of air to the jet engines.

The indirect cycle using molten salt was researched by ORNL and resulted in the Aircraft Reactor Experiment (ARE), which took approximately 12 years to develop and was operational for only nine days. The reactor shown in Figure 2 operated at a modest 2.5 MW of thermal output at temperatures of ~1580°F (Rosenthal 2009, 26). Although the operation demonstrated the feasibility of nuclear powered aircraft, the program was halted in 1961 with the election of President John F. Kennedy. Still, the lessons learned in molten salt reactors and the developments in materials and shield design would be used in the laboratory's next undertaking: the Molten Salt Reactor Experiment (MSRE).

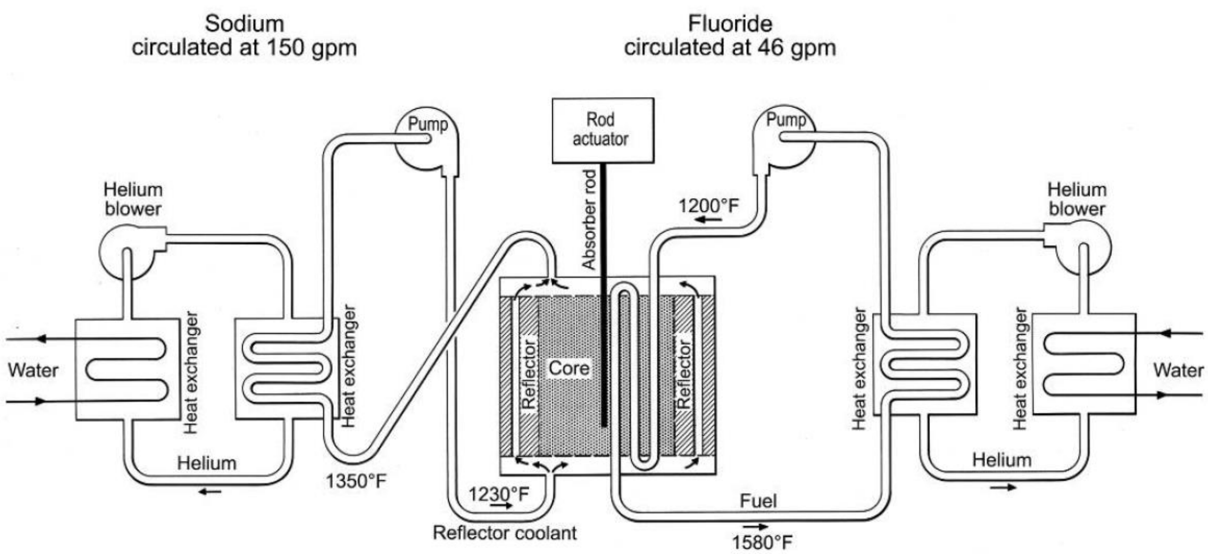

Figure 2. Oak Ridge National Laboratory's Aircraft Reactor Experiment (Operated in 1954) (Rosenthal 2009, 27).

The Molten Salt Reactor Experiment (MSRE) was funded by the Atomic Energy Council following successful demonstration of the technology in the ARE. Originally, two distinct designs were proposed that took the form of a single-fluid and a two-fluid reactor. In both variants, Uranium-235 (^{235}U) and Uranium-233 (^{233}U) were used as fuel dissolved in lithium-fluoride and beryllium-fluoride salts, and a solid graphite matrix was constructed in the reactor core to act as a neutron moderator.

In the single-fluid variant, ^{235}U served as fuel mixed into a single coolant salt. ^{232}Th was also added to the coolant salt because of its large cross-section for neutron absorption and its ability to decay into ^{233}U, which is another fissile nuclear fuel. The ability of ^{232}Th to absorb neutrons and decay into fissile Uranium makes Thorium a "fertile" material. The single-fluid variant contained fluoride salts, ^{233}U and ^{232}Th all in the same volume of fluid, which circulated through the reactor core.

In the two-fluid variant, ^{235}U is dissolved into fluoride salts and circulates through the core. This is known as the "fuel salt" and contains the fissile Uranium needed for fission. A second fluid, known as the "blanket salt" surrounds the reactor core and is separated from the fuel salt by a mechanical barrier, usually made of graphite (Rosenthal 2009, 29). The blanket salt contains fertile ^{232}Th that absorbs neutrons that have escaped the core and then decays into ^{233}U. A separate chemical processing plant extracts the fissile ^{233}U from the blanket salt and injects it into the fuel salt, where it will enter the core and fission to create heat. Further detail on the $^{233}U/^{232}Th$ fuel cycle is provided in Chapter 2, which describes the Flibe Energy LFTR in detail as a two-fluid molten salt reactor.

The MSRE was a single-fluid molten salt reactor containing lithium-, beryllium-, and zirconium-fluoride salts with dissolved ^{235}U and ^{232}Th. As the fuel salt passed through the graphite-moderated reactor core, neutrons from decaying fission products were slowed, or "moderated" to energy levels that allowed absorption by the nuclear fuel and resulted in fissions. The kinetic energy of the fission products created heat within the fuel salt. The heat was then transferred to an intermediate fluoride salt and ultimately rejected to an air radiator that was cooled by blower fans. Sump-type salt pumps were designed as the high point of the reactor, with access that allowed sampling of the fluoride fuel salts and also allowed adding of more nuclear fuel. Both ^{233}U and

Plutonium were used later to demonstrate the flexibility of the MSRE to utilize different fissile materials for fuel (Rosenthal 2009, 32).

The MSRE first went critical on June 1, 1965 using ^{235}U, and was later brought critical on October 2, 1968 using ^{233}U. The MSRE operated until December 1969 but was shut down due to budget constraints. The Atomic Energy Council had decided to redirect funds to the sodium-cooled fast-spectrum breeder reactor and in 1973 the molten salt reactor program was dismantled (Rosenthal 2009, 33). Nonetheless, significant achievements were realized during the MSRE, demonstrating not only the feasibility but also the inherent safety of this novel technology. Much advancement would be required to elevate the MSRE to an industrial scale, and government funding proved inadequate to support such advancements. Thus it was almost 50 years before universities, private investors and engineers began pursuing the revival of research on molten salt reactors. Flibe Energy's LFTR stands among only a handful of MSRs under development in the United States today and is a direct representation of the Generation IV International Forum's vision for the future of advanced nuclear reactors.

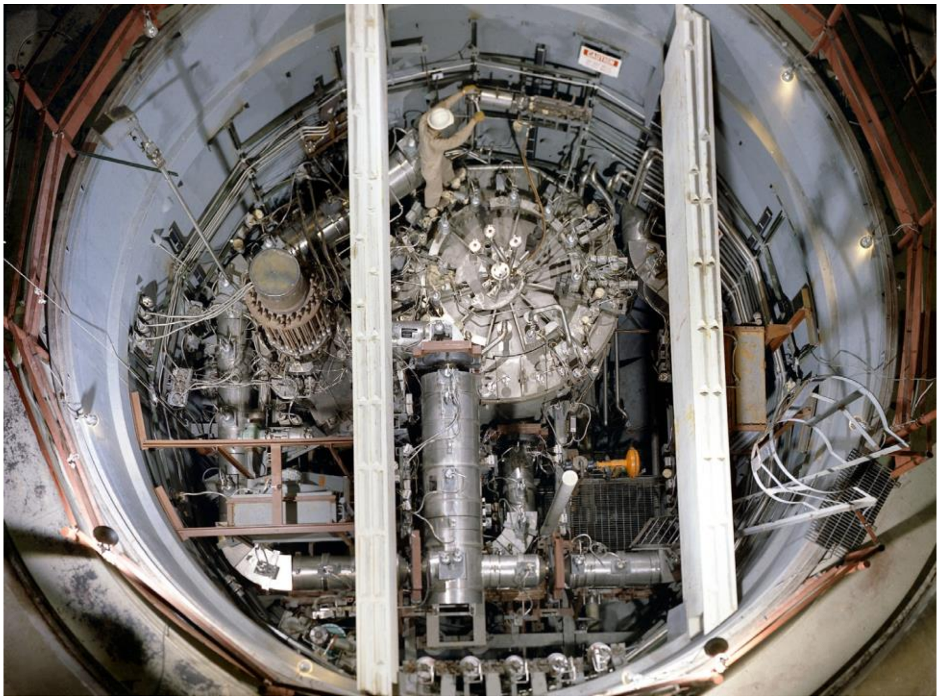

Figure 3. Oak Ridge National Laboratory's Molten Salt Reactor Experiment (Operated from 1965-1969) (Rosenthal 2009, p. 33)

1.2 PROBABILISTIC SAFETY ASSESSMENTS

In the 1970s, following two decades of successful operation of Generation I nuclear reactors, engineers and licensing authorities became increasingly interested in developing a method to capture the true magnitude of risk associated with operation of commercial nuclear power plants. Two key founders of the quantitative risk assessment were B. John Garrick and Stan Kaplan, engineers who worked together at the Atomic Energy Council and later formalized their quantitative approach in an article titled "On the Quantitative Definition of Risk" (1981).

In their work, Kaplan and Garrick define the "triplet definition of risk" where the engineer must answer the following three questions:

1. What can happen?
2. How likely is it that such an event will happen?
3. If it happens, what are the consequences?

Answering these questions will result in a set of scenarios and their associated outcomes. Consider Table 2 where a list of scenarios, the likelihood or probability of occurrence and the consequence for each is captured.

Scenario	**Likelihood**	**Consequences**
s_1	p_1	x_1
s_2	p_2	x_2
…	…	…
s_n	p_n	x_n

Table 2. Scenario List for Triplet Definition of Risk

The i^{th} line of Table 2 can be thought of as a triplet:

$$<s_i , p_i , x_i>$$

where s_i is a scenario identification or description

p_i is the probability or likelihood of that scenario (deterministic or assumed); and

x_i is the consequence or evaluation measure (i.e. measure of damage) (Kaplan and Garrick 1981, 13)

Garrick and Kaplan's early work and continued research led to great breakthroughs in the field of Quantitative Risk Assessment (QRA). In particular, the application of this approach to the nuclear power industry became known as Probabilistic Safety Assessment (PSA) and is used extensively to this day as a tool for design risk mitigation and licensure of commercial nuclear power plants.

In 1975, the first use of the Probabilistic Safety Assessment was demonstrated when the U.S. Atomic Energy Commission published the *Reactor Safety Study* under the direction of N.C. Rasmussen of M.I.T. (Garrick 2008, 248). This work took over three years to complete and included failure data from three decades of nuclear plant operations. Using these statistics, engineers were able to assign likelihoods of failure to different plant components, and quantify the consequences of these failures.

In his work *Quantifying and Controlling Catastrophic Risk,* Garrick went on to refine his approach to PSAs and listed the following six steps (Garrick 2008, 249) as a thorough methodology for capturing the "triplet" mentioned above:

1. Define the system being analyzed in terms of what constitutes normal operation to serve as a baseline reference point.
2. Identify and characterize the sources of danger, that is, the hazards (i.e. stored energy, toxic substances, hazardous materials etc.).
3. Develop "what can go wrong" scenarios to establish levels of damage and consequences while identifying points of vulnerability.
4. Quantify the likelihoods of the different scenarios and their attendant levels of damage based on the totality of relevant evidence available.
5. Assemble the scenarios according to damage levels and cast the results into the appropriate risk curves and risk priorities.
6. Interpret the results to guide the risk management process.

Unfortunately, for advanced nuclear reactors in the design stage it is often difficult or impossible to quantify levels of damage as required in Step 3 or assign likelihoods of occurrence required by Step 4. In an international effort to guide PSA efforts for advanced nuclear reactors, one committee recognized that "the technical challenges of the PSA for new reactors, which are in the last phases of design and commissioning stage, include a lack of design detail, a lack of empirical data, and the possibility of failure scenarios that differ in character from those treated in PSAs for current reactors" (Nuclear Energy Agency 2013, 5). Another engineer notes that "epistemic problems such as uncertainties due to lack of design information, unknown phenomena, plant-specific hazards, data etc., are larger than that from existing reactors, and will impose a significant challenge to decision makers" (Alrammah 2014).

In his work, Garrick agrees that quantitative risk assessments must be performed individually for different proposed reactor plants due to the inherent changes in risk probabilities based on design differences (Garrick 2008, 252). In observance of these limitations, analysis will be conducted on the proposed Flibe Energy LFTR based on the availability of design descriptions and existence of "what-if" analysis results for the Flibe Energy design.

Steps 1 and 2 of Garrick's methodology were thoroughly addressed in the "Technology Assessment of a Molten Salt Reactor Design" (2015). The end result is a comprehensive list of important hazards that pose the most significant consequences for safety or integrity of the LFTR system. Step 3 of Garrick's methodology then requires the engineer to determine "what can go wrong." In this step, an initiating event analysis must be conducted to determine *how* the identified hazards may be realized. This initiating event analysis represents the first step to a Level 1 Probabilistic Safety Assessment. Figure 4 below illustrates the development of probabilistic safety assessments, from Level 1 to Level 3.

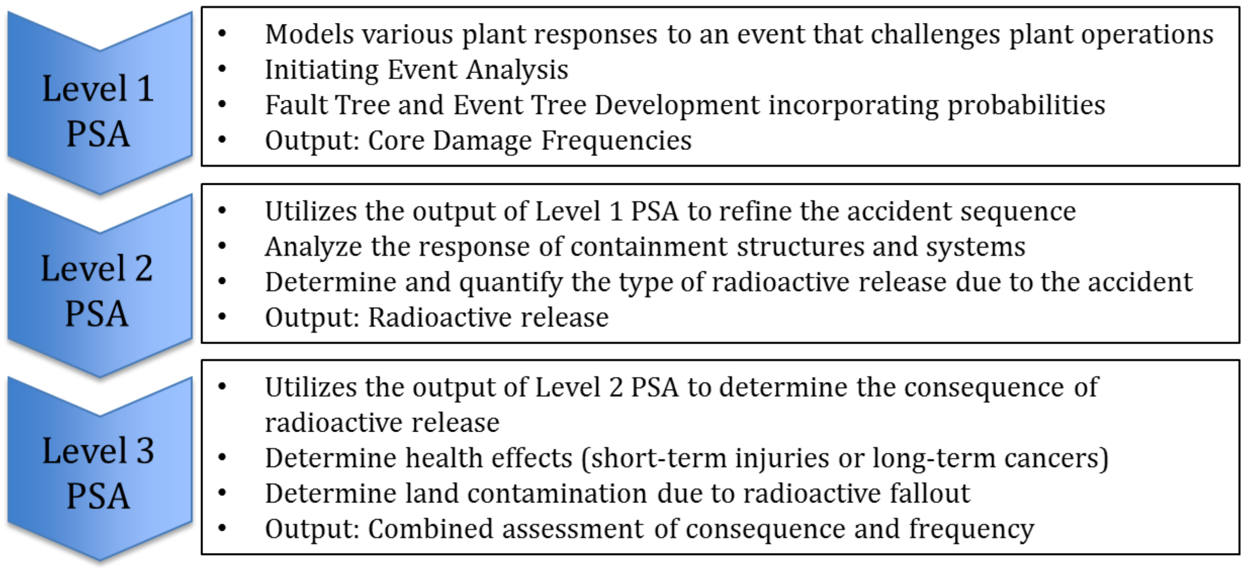

Figure 4. Development of Probabilistic Safety Assessments

This thesis falls short of satisfying the requirements of a Level 1 PSA because of the inability to apply probabilities and core damage frequencies due to a lack of design detail and operating experience. Still, the fault tree analysis and event tree analysis will prove useful to decision-makers and engineers in identifying vulnerabilities to the current LFTR design.

Starting with the list of hazards identified by Flibe Energy and the EPRI, the objective of this thesis is to conduct an Initiating Event Analysis. Using International Atomic Energy Agency guidance, this process will involve a review of previous NRC and EPRI initiating events, reference to previous PSAs, performance of event tree analysis (inductive reasoning) and performance of fault tree analysis (deductive reasoning) using master logic diagrams. The goal is to develop a list of initiating events that may lead to a violation of the safety or integrity of the Flibe Energy LFTR as described in the "Technology Assessment of a Molten Salt Reactor Design" (2015). The author recognizes that many unforeseen or undeveloped risks may exist in addition to those identified by the EPRI and Flibe Energy. Later efforts to perform probabilistic safety assessments may incorporate more specific design information, and may determine additional hazards not discovered by elicitation of expert judgment by the EPRI. However, for the purpose of scoping this thesis, evaluation is limited to the list of primary hazards in Table 4-4 of the "Technology Assessment of a Molten Salt Reactor" (2015).

CHAPTER 2

LITERATURE REVIEW

To provide a foundation of technological understanding, a description of the design and operation of the two-fluid Flibe Energy LFTRs follows, including a breakdown of major system components and engineered safety features. The majority of the system design description is gathered from the "Technology Assessment of a Molten Salt Reactor Design" (2015) with supplemental information included from Oak Ridge National Laboratory MSRE and MSBR technical documentation.

In the two-fluid LFTRs, lithium-beryllium-fluoride with uranium-tetrafluoride fuel ($2LiF_2$-BeF_2-UF_4) is the primary fuel salt that will be circulated through the reactor. The blanket salt is comprised of lithium-beryllium-fluoride with thorium-tetrafluoride ($2LiF_2$-BeF_2-ThF_4). The fuel salt and blanket salt are kept physically separated by the reactor vessel, which is constructed to provide separate plenums for each salt. As fission occurs in the reactor core, some neutrons released during fission leak into the blanket salt and are absorbed by fertile ^{232}Th. This neutron absorption begins the thorium fuel cycle, shown below, in which fertile thorium is converted into fissile uranium.

$$^{232}_{90}\mathrm{Th} + {}^{1}_{0}\mathrm{n} \rightarrow {}^{233}_{90}\mathrm{Th} \rightarrow^{\beta-} {}^{233}_{91}\mathrm{Pa} \rightarrow^{\beta-} {}^{233}_{92}\mathrm{U}$$

Using a chemical processing plant, ^{233}U is then removed from the blanket salt and returned to the fuel salt to maintain the inventory of fissile fuel. Within the reactor core, a solid graphite moderator aids in slowing or "thermalizing" fission neutrons. Once in the thermal spectrum, the neutrons can be absorbed by the ^{233}U causing fission and heat generation. Heat is then transferred to the fuel salt itself, which rejects heat to the intermediate loop and ultimately drives the supercritical CO_2 power conversion system to generate electricity. An external cooling system is used to maintain temperatures of the power conversion system, and fission product gases caused by fission of ^{233}U must be removed from the primary fuel salt. From this basic description, the reader sees that there are essentially seven major subsystems:

1. Reactor Core and Vessel
2. Primary Fuel Salt loop
3. Intermediate Coolant Salt loop
4. Chemical Processing Plant
5. Off-gas Handling Plant
6. Power Conversion System (Supercritical CO_2 Closed Brayton-cycle)
7. External cooling system

Because the power conversion system and external cooling system are already used in coal- and natural gas-fired power plants, the technology is well established and not included in the initiating event analysis. A more detailed description of the design and role of each new subsystem is provided in the following sections.

2.1 REACTOR CORE AND VESSEL

The reactor core and vessel of the Flibe Energy LFTR serve several functions crucial to successful operation and safety of the reactor. The reactor core contains a matrix of solid graphite material whose large macroscopic cross-section for scattering makes it a perfect for thermalizing neutrons. The remainder of the reactor vessel will be constructed of Hastelloy-N and serves the structural purpose of separating the fuel salt and blanket salt, and directing the hot fuel salt exiting the core to the primary fuel salt loop (Electric Power Research Institute 2015, 3-8). In the two-fluid MSR design, the fuel salt and blanket salt must be kept separate by designing the reactor vessel with two plenums that are physically separated to direct fuel salt through the core and maintain blanket salt surrounding the core.

Active and passive control rod systems are designed to be inserted or withdrawn from the reactor core to maintain a critical nuclear reaction. Common with traditional PWRs, active control rods would be made of neutron-absorbing material and controlled by a human operator. In order to maintain a critical reaction, the operator could insert the rods to absorb neutrons, slowing or stopping the nuclear reaction as desired. Another design option for the LFTR active control rod system is a pneumatically actuated “liquid control rod” that utilizes a column of blanket salt whose height is adjusted by varying the pressure of helium over the fluid. Theoretically, this liquid control rod would fail open during a loss of electrical power, with gas pressure being vented allowing the neutron-absorbent blanket salt to fill a central channel and shut down the reactor (Electric Power Research Institute 2015, 3-9). Additionally, novel in the LFTR is the concept of passive control rods. Due to the neutron-absorbing properties of the blanket salt, it has been identified that a loss of blanket salt would cause an increase in reactor power. To compensate for this increase in reactor power, passive control rods are designed to “float” on the blanket salt, remaining outside the reactor core during normal operations. Upon a blanket salt leak, these floating control rods would lose buoyancy and lower into the reactor core, slowing the nuclear reaction or shutting down the reactor until the casualty has been corrected. (Electric Power Research Institute 2015, 3-9).

During the thorium fuel cycle following neutron-absorption in the blanket salt, ^{233}Th β-minus decays into ^{233}U, which generates heat. A small heat exchanger is being designed to accommodate cooling of the blanket salt. Natural circulation drives the blanket salt through the heat exchanger to maintain proper temperatures surrounding the core.

2.2 PRIMARY FUEL SALT LOOP

The Primary Fuel Salt loop serves to direct hot fuel salt from the reactor core to the primary heat exchangers, where heat is rejected to the intermediate loop coolant salt and then recirculated to the core. A primary fuel salt pump provides the pressure differential to overcome losses in the primary heat exchanger and the reactor core.

Additionally, the primary fuel salt loop contains the fuel salt drain tank and cooling system. At the lowest point in the primary fuel salt loop, a freeze plug is maintained solid by an active cooling system. In the event of a casualty in which the fuel salt overheats, coolant flow stops or the active cooling capacity of the freeze plug is exceeded, the freeze plug fails open and gravity drains the primary fuel salt into a subcritical fuel salt drain tank (Electric Power Research Institute 2015, 3-9). Because the drain tank does not contain the required geometry of graphite to thermalize neutrons and cause absorption by ^{233}U, the nuclear fission reaction will immediately cease, causing the fuel salt to solidify in a stable condition until corrective actions and cleanup can occur.

2.3 INTERMEDIATE COOLANT SALT LOOP

The Intermediate Coolant Salt loop serves to keep the primary fuel salt physically separate from the power conversion system. This design serves a crucial role in plant integrity as the power conversion system operates at extremely high pressures (~200 Bar) (Electric Power Research Institute 2015, 3-10). Due to the high boiling point of the primary fuel salt, the reactor vessel and primary piping are not designed to accommodate high pressure.

In the absence of an intermediate loop, a rupture in the gas heat exchanger could translate pressure from the CO_2 gas back to the primary loop, causing a rupture and release of radioactivity from the primary loop. To mitigate this risk, the intermediate loop stands between the high-pressure power conversion system and the low-pressure primary loop. Pressure relief valves designed into the intermediate loop would relieve pressure caused by a failure of the gas heat exchanger. The subsequent loss of intermediate salt would cause a loss of cooling within the primary, initiating the melting of the freeze plug and resulting in the complete draining of the primary loop into the subcritical drain tank (Electric Power Research Institute 2015, 3-10).

Included in the Intermediate Loop are another coolant salt pump and the salt side of the gas heat exchanger for transferring thermal energy to the supercritical CO_2 Closed Brayton-cycle power conversion system.

2.4 CHEMICAL PROCESSING PLANT

The function of the chemical processing plant is to remove radioisotopes from the blanket salt that are generated from neutron-absorption of the fertile ^{232}Th. These isotopes are primarily protactinium-233 (^{233}Pa) and uranium-233 (^{233}U). Ultimately, the ^{233}U will be returned to the primary fuel salt loop to serve as fuel. A secondary function of the chemical processing plant is to remove fission products from the primary loop and prepare them for storage or shipment off-site.

The chemical processing plant is extremely complicated and must handle both gaseous and liquid metal radioactive byproducts of fission and absorption. One major safety concern is the production of gaseous fluorine and hydrogen, both of which are highly chemically reactive. Flibe Energy intends for the Chemical Processing Plant to operate primarily with human supervision but with limited human actuation (Electric Power Research Institute 2015, 3-11). Due to the high operating temperatures and high radioactivity of fluids in the system, continued research and development is needed before the chemical processing plant is ready for use in the LFTR.

These safety concerns and the lack of proven design data will be addressed in greater detail in the initiating event analysis within this thesis.

2.5 OFF-GAS HANDLING SYSTEM

Following fission of ^{233}U, Xenon and Krypton gases build up in the primary loop and must be removed to prevent gas pockets from interrupting the hydraulic performance of the fuel salt in the reactor core. Fortunately, most isotopes of Xenon and Krypton formed from fission are short-lived and decay into stable elements within approximately 30 days (Electric Power Research Institute 2015, 3-12).

The off-gas handling system serves to redirect these fission product gases to the fuel salt drain tank, where most of the radioactive decay will occur transforming Xenon and Krypton into the stable non-gaseous daughters Cesium, Rubidium, Strontium and Barium. Gaseous Krypton and Xenon are then passed through a charcoal filter cooled by water. This gas stream is cryogenically frozen and the Xenon bottled for resale. Krypton gas still contains radioactive Krypton-85 (half-life of 10 years) and must be stored until complete decay. Helium gas from this process is redirected to the chemical processing plant for use cleaning the fuel and blanket salts (Electric Power Research Institute 2015, 3-12).

The mechanical requirements to accomplish off-gas handling are relatively simple, and the radioisotopes are well understood as they are common between the LFTR and LWRs.

Figure 5 represents a simplified reactor schematic including all of the major subsystems described in Chapter 2.

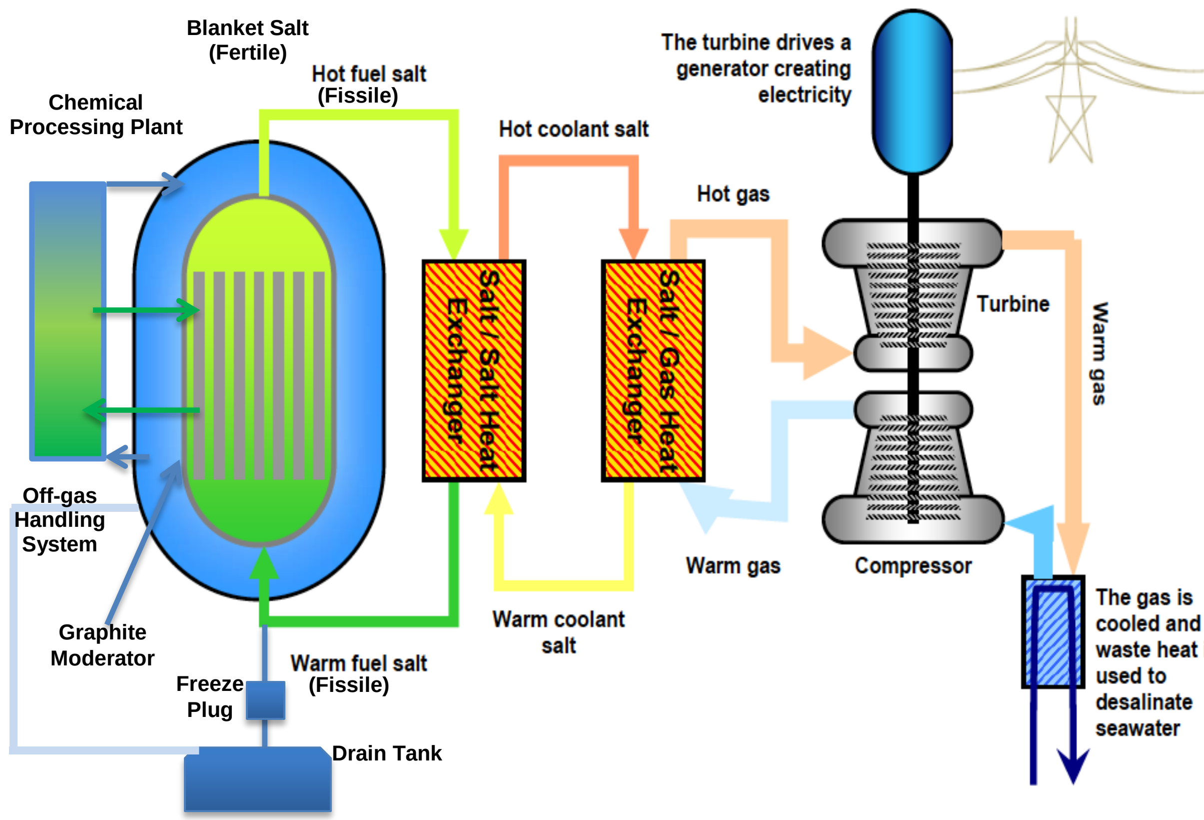

Figure 5. Liquid Fluoride Thorium Reactor
Modified from "Introduction to Flibe Energy" (Sorenson and Dorius 2011)

CHAPTER 3

APPROACH AND METHODOLOGY

Several resources exist that guide the conduct of Probabilistic Safety Assessments. Primarily, the IAEA Technical Document 719 titled *Defining Initiating Events for Purposes of Probabilistic Safety Assessments* (1993) provides guidance on how to develop a complete list of initiating events (IEs).

An initiating event is defined as "an occurrence that creates a disturbance in the plant and has potential to lead to core damage, depending on the success or failure of the various mitigating systems in the plant" (International Atomic Energy Agency 1993, 7). In traditional nuclear reactors, core damage refers to the release of nuclear fuel and fission products from the fuel elements into the primary coolant. Damage to the reactor core could ultimately lead to the release of fuel or fission products to the surrounding environment and result in public exposure. Since the nuclear fuel is already homogenously distributed in the primary coolant of the LFTR, the definition of core damage must be slightly altered for application to molten salt reactors. For the purpose of this thesis, core damage for the LFTR is defined as the release of long-lived radioisotopes from the primary plant boundary. This could include the release of fuel salt or fission product gases from the primary boundary.

This change to the definition of core damage focuses the scope of this thesis to investigate only those initiating events with the potential to release long-lived radioisotopes from the primary plant boundary to the surrounding environment. The research questions to be addressed are

1. Which hazard scenarios from Table 4-4 of the "Technology Assessment of a Molten Salt Reactor Design" (2015) would result in the release of long-lived radioisotopes from the primary plant boundary?
2. Which initiating events would cause the realization of the hazard scenarios identified above?

Initiating events are generally broken down into three categories: loss-of-coolant accidents (LOCA), transient IEs and special or "common cause" IEs.

The LOCA refers to any mechanical failure resulting in loss of the primary coolant and is extremely concerning in PWR and BWR applications because it results in a rapid loss of cooling capability for the solid nuclear fuel (International Atomic Energy Agency 1993, 19). In the LFTR, where nuclear fuel and fission products are already suspended within the coolant by design, a LOCA itself would constitute the release of long-lived radioisotopes from the primary plant boundary. Therefore, any transient or special IE identified that leads to a LOCA will constitute core damage as defined above.

Transient initiating events refer to those that result in a disturbance during normal plant operation but do not result in a loss of coolant. Still, transient IEs require either automatic or manual plant shutdown to prevent equipment damage or the release of radioactivity (International Atomic Energy Agency 1993, 20).

Special initiating events are those that, in addition to requiring plant shutdown, also disable one or more safety systems intended to mitigate the risk of radioactive release (International Atomic Energy Agency 1993, 20).

Determining a comprehensive list of transient and special initiating events must be done using several methods. Due to the lack of operating experience with MSRs and due to the limitations inherent to early design phase reactors, the following methods will be used to determine transient and special initiating events for the Flibe Energy LFTR:

1. Engineering evaluation and technical study of the plant
2. Review of EPRI Lists of initiating events (EPRI-NP-2230, NUREG/CR-3862, 6928, 5750, 1829)
3. Reference to previous Probabilistic Safety Assessments
4. Logical Classification

a. Fault Tree Analysis (deductive reasoning)

b. Event Tree Analysis (inductive reasoning)

CHAPTER 4

ANALYSIS

The first step in proceeding with the Initiating Event Analysis is to perform and engineering evaluation of the Flibe LFTR as described by the EPRI (2015) and attempt to determine which hazards may result in the release of long-lived radioactivity. A review of EPRI and NUREG Initiating Event Lists and reference to previous PSAs will also be conducted to determine applicability of previously identified initiating events to the LFTR design.

4.1 ENGINEERING EVALUATION AND TECHNICAL STUDY OF THE PLANT

First, consider the hazards that were identified in the "Technology Assessment of a Molten Salt Reactor Design" (2015).

LFTR System or Component	Hazard Scenario
Reactor Vessel and Containment Cell	Unintentional control rod withdrawal
	Loss of blanket salt
	Premature criticality during filling
	Inflow of contaminants or unexpected isotopic ratio in the fuel salt
	Breakage of one or more graphite tubes
	Inadvertent release of fission product gas from reactor cell or containment
Fuel Salt Processing	Hydrogen reacts with fluorine in the chemical processing system
	Excess pressure in the helium bubbler
Primary Heat Exchanger	Minor failure in the primary heat exchanger
	Major failure in the primary heat exchanger
	Sealed housing for the electric drive motors for pumps fail
Blanket Salt Processing	Inadequate removal of Pa or U from the blanket salt
	Electrolytic cell is improperly operated
Off-gas Handling System	Potassium hydroxide is released
Drain Tank	Improper or inadequate cooling of the drained fuel salt
	Partially thawed piece of salt plug or solid mass obstructs piping to drain tank

Table 3. Important Hazards to safety and integrity of the LFTR (Electric Power Research Institute 2015, 4-17)

Through careful consideration of the discussions in Appendix A of the "Technology Assessment of a Molten Salt Reactor Design" (2015) the following hazards were selected for further study:

- Unintentional control rod withdrawal
- Breakage of one or more graphite tubes
- Improper or inadequate cooling of the drained fuel salt
- Partially thawed piece of salt plug or solid mass obstructs piping to the drain tank

The selection process and justification for inclusion of these hazards is discussed further in Appendix A to this study. Before conducting an initiating event analysis for these four hazards, it is necessary to consider the mode of failure that is possible as a result of the realization of these casualties. Below is a discussion of the potential for core damage and release of long-lived radioactivity that may result from unintentional control rod withdrawal, breakage of graphite tubes, inadequate cooling of the drain tank or obstruction of the drain piping.

4.1.1 UNINTENTIONAL CONTROL ROD WITHDRAWAL

Unintentional or unexpected withdrawal of the control rods from any reactor represents one of the most concerning reactivity addition casualties because of the potential to cause rapid and uncontrollable increase in reactivity, which in turn causes temperature increase, potential structural failure of core materials, and expansion of the fuel salt that may approach design limits.

Perhaps one of the most severe reactor accidents caused by unintentional control rod withdrawal occurred at the U.S. Army Stationary Low-power Reactor Number 1 (SL-1) operated at the National Reactor Test Station in Idaho in January 1961. At 9:01 pm on the evening of January 3rd, firefighters and medics responded to radiation alarms and fire alarms at the SL-1 site where three men had been conducting routine maintenance in preparation for reactor operations in the coming days. An explosion had occurred at SL-1 due to the inadvertent withdrawal of a central control rod beyond the allowable limit. Two of the three technicians were pronounced dead at the scene and one perished during resuscitation efforts. In addition to the loss of life, the SL-1 reactor site itself was completely destroyed by the explosion. The reactor vessel had jumped almost nine feet in the air, shearing connecting piping due to the blast (Thatcher n.d., 11), and several of the reactor control rods had been ejected from the core resulting in overheating and failure of the fuel cladding. After an exhaustive investigation, it was determined that inadvertent control rod withdrawal had caused the reactor explosion and the deaths of three military technicians who were on site. Part of the required maintenance called for the technicians to manually raise the control rods from within the core with the reactor shutdown (Thatcher n.d., 2). Accidentally raising the center control rod to a height of over 20 inches, one technician unknowingly added enough positive reactivity to cause the reactor to experience "prompt criticality," a condition in which neutrons are generated so rapidly from fission that the reaction becomes uncontrollable. This condition resulted in a vaporization of fuel materials and a steam explosion, which ejected the center control rod and killed the technician instantly. This accident constituted the first nuclear accident-related fatality and indeed the first "reactor accident" in the United States.

In the aftermath of the SL-1 accident, significant improvements were made to design and safety requirements in modern nuclear reactors. Among them, more controlled maintenance evolutions and procedures were developed. More importantly, reactor designs were improved such that withdrawal of a single control rod could not add enough positive reactivity to cause such a significant power excursion. It is with the SL-1 accident in mind that the Flibe LFTR liquid control rod design must be critiqued and considered.

In the Flibe LFTR, there is discussion of use of a liquid control rod containing blanket salt (neutron-absorbent) that will "fail open" upon loss of power (Electric Power Research Institute 2015, 3-9). However, another potentially damaging scenario has not been considered – rapid overpressurization of the blanket salt control rod that causes an almost immediate ejection of the liquid control rod. Research indicates that there is little to no experience using this concept of liquid control rods in modern nuclear power plants. The closest comparison is hydraulically-operated solid control rods, which use water to actuate pistons to raise and lower solid control rods within a reactor core. The concern with reliable operation of hydraulically-operated control rods is so important that specific safety mechanisms have been designed to prevent overpressurization of actuating fluid which could result in ejection of control rods (Carruth 1989). In the Flibe Energy LFTR System Design Description, limited design detail is offered for the proposed liquid control rod system. As such, assumptions are made about the basic engineering design required to accomplish such a system. It is assumed that a blanket-gas will be kept pressurized over the blanket salt control rod, maintaining a column of blanket salt of specific height in the center of the reactor core to act as a neutron absorber to control neutron flux. A single valve or series of valves subsequently referred to as the “blanket-gas control valve” will be used to govern the pressure of the blanket-gas, which in turn governs the liquid control rod height. Some form of overpressure protection system will be included to prevent rapid pressurization of the blanket-gas. The next iteration of Flibe Energy LFTR must include further design detail for the liquid control rod system to allow more thorough evaluation of the risks. Though a novel concept with potential for success, the risk of control rod ejection through overpressurization of a liquid control rod system has not been thoroughly addressed for the Flibe LFTR.

There are several mechanisms of failure that may result from uncontrolled rod withdrawal. As demonstrated in the SL-1 accident, prompt criticality and fuel vaporization represent the most extreme mode of failure (Thatcher n.d.). However, the expected thermal expansion of the fuel salt may also represent a hazard to plant integrity. In pressurized water reactors, thermal expansion occurs in the light water coolant circulated through the core during normal operations and power transients. This thermal expansion is accommodated by an expansion volume called the pressurizer. Any condition in which the pressurizer is not available to accommodate thermal expansion of the water is referred to as “solid plant operations” and is known to be very dangerous because the potential exists to rapidly overpressurize the primary system and cause brittle fracture of the reactor vessel or primary plant piping (International Atomic Energy Agency 2010, 16). As of now, the Flibe Energy LFTR has no design feature to accommodate thermal expansion of the fuel salt during normal or casualty modes of operation. Indeed, during elicitation of expert judgment in May 2015, the EPRI and Flibe Energy acknowledged that design features such as surge capacitors must be added to the LFTR to accommodate this thermal expansion of fuel salt (Electric Power Research Institute 2015, A-5 and A-10).

Consider the following discussion of the potential changes in fuel salt volume based on thermal expansion. Under normal operating circumstances, fuel salt will enter the reactor at 500°C with a density of 2005.1 kg/m^3 and exit at 653°C with a density of 1952.1 kg/m^3 (Electric Power Research Institute 2015, 3-15). Though Flibe Energy has not confirmed the total fuel salt volume of the LFTR, an approximation can be made based on similar proposed molten salt reactor designs. In Japan, the superFUJI MSR is a 2,272MWt/1,000MWe plant with a 62.0 m^3 inventory of primary fuel salt and the FUJI-Pu MSR is a 250MWt/100MWe plant with a 12.0 m^3 inventory of primary fuel salt (Yoshioka, et al. 2016, 24-27). By scaling the FUJI MSR designs, the 600MWt/250MWe Flibe LFTR may be expected to contain between 16.0-28.8 m^3 of primary fuel salt for an average of 22.4 m^3 of primary fuel salt (~50 tons). Given the expected change in density across the reactor core, it is seen that there is a 2.7% increase in volume of the fuel salt under normal steady state conditions alone. This equates to a 0.6 m^3 change in volume during normal reactor operations. Under transient or casualty conditions, this thermal expansion could reasonably exceed 1.0 m^3, necessitating some form of expansion volume or surge capacitor to prevent rapid overpressurization of the reactor vessel and primary loops. The absence of an expansion volume effectively constitutes “solid plant operations” and places the LFTR at risk for brittle fracture or pressurized thermal shock (Boyd 2008, 463).

Overall, the hazard of unintentional control rod withdrawal in the LFTR presents significant safety and integrity concerns. The liquid control rod concept is in the early design phase and suffers from a lack of detail and the potential for control rod ejection. The mechanism of failure due to unintentional control rod withdrawal is prompt criticality as a worst case scenario, and at the very least the potential for thermal expansion of fuel salt resulting in fracture of the reactor vessel or primary plant boundary due to the absence of any sufficient expansion volume.

4.1.2 BREAKAGE OF ONE OR MORE GRAPHITE TUBES

During design and operation of the MSRE at ORNL, breakage of one or more graphite tubes was recognized as "the scenario that could represent the largest reactivity addition" to a molten salt reactor (Electric Power Research Institute 2015, A-12). Several mechanisms exist whereby breakage of the graphite fuel tubes or moderator elements may result in higher reactivities within the core. The most credible and severe accident involving the graphite moderator is a net fuel addition to the core region due to the expulsion of graphite and replacement with fuel salt (Kasten 1967, 18). This would result in net positive reactivity addition and a power excursion due to higher fuel concentration in the core region. Still, some studies for the MSRE postulated that the net addition of reactivity due to replacing graphite with fuel salt would be negligible and does not present a safety hazard (Beall, et al. 1964, 219). Other mechanisms for reactivity increase due to loss of graphite or permeation by fuel salt include (Beall, et al. 1964, 219-221):

- Bowing of the graphite moderator and fuel channels due to irradiation, resulting in higher localized fuel concentrations within the core region,
- Graphite shrinkage that causes decreased moderator volume and larger fuel channels,
- Fuel salt permeating the pores in the graphite resulting in increased amount of uranium in the core and higher levels of afterheat in the graphite following shutdown due to the sustained reaction occurring in the moderator,

- Sorption of uranium onto the graphite surface due to irradiation and fission at elevated temperatures causing higher reactivity levels during normal operation, higher graphite temperatures and higher levels of afterheat.

Though initial safety assessments of the MSRE indicated little impact on safety due to loss of graphite in the core, this potential casualty must be further studied as a possible cause for core damage. One hazard scenario not discussed in the MSRE Safety Analysis is the potential for localized regions of high neutron flux due to failure of the graphite moderator. For example, in the event of breakage of one or more graphite tubes, fuel salt channels may be obstructed which may allow for regions of high temperature, high neutron flux and low fuel salt flow due to off-design clearances within fuel salt channels. These localized regions of higher neutron flux would not likely cause catastrophic power transients, which also means that automatic protective action such as a reactor scram would not likely occur. However, localized regions of the core may experience temperatures in excess of design allowance, which could result in further structural damage or at worst a failure of core integrity due to overheating. This casualty would be similar to a coolant channel blockage in LWRs where material obstruction causes insufficient flow and higher temperatures to occur within a localized region of the reactor core (Salama and El-Morshedy 2011). Coolant channel blockage in LWRs can lead to a failure of fuel cladding and the introduction of nuclear fuel and fission products to the primary coolant which constitutes core damage under the traditional definition.

Another mode of failure may result from transport of the broken graphite moderator to the primary fuel salt loops or to the drain tank piping. Section 4.1.4 includes further discussion of the impact of the graphite moderator's presence in the primary loops or drain tank piping.

Of course, determination of the feasibility of core damage in the LFTR due to breakage of the graphite moderator is dependent on design-specific neutronic calculations for this casualty which are not yet available. One recommendation is to conduct "hot channel analysis" and to investigate the impact of fuel salt channel blockage on core integrity in a graphite-moderated MSR such as the LFTR. For the purpose of this study, the author assumes that breakage of one or more graphite tubes has the potential to result in core damage due to localized overheating or gross addition of positive reactivity described by Beall et al (1964).

4.1.3 IMPROPER OR INADEQUATE COOLING OF THE DRAINED FUEL SALT

In the Flibe Energy LFTR, the passively cooled subcritical drain tank stands as one of the most crucial components to demonstrating the "walk-away safety" advertised by designers. For most credible casualties resulting in overheating of the reactor fuel salt, the safety mechanism is a freeze plug which melts and enables draining the contents of the reactor to a subcritical drain tank where the nuclear reaction is impossible. Here, the drained fuel salt cools and solidifies until corrective action and cleanup can be performed (Electric Power Research Institute 2015, 3-9). Additionally, engineers have developed two potential flowpaths into the drain tank under casualty conditions. The first is a dedicated piping system from the freeze plug to the drain tank, designed to accommodate transport of the fuel salt during an accident or during normal shutdown operations. The second is a catch pan beneath the reactor vessel designed to direct molten fuel salt into the drain tank in the event of gross reactor vessel damage (Electric Power Research Institute 2015, 3-25). In all cases, the drain tank is designed to utilize passive cooling where heat is rejected to the outside environment and no electrical power or active cooling mechanisms are required for safe operation.

If the drain tank is expected to be the cornerstone of passive safety under all casualty conditions, which result in fuel salt exiting the reactor core, serious consideration must be given to the integrity of the tank and the design-specific features, which guarantee proper performance. Currently, the LFTR design team proposes a single drain tank "with sufficient volume to receive the entire inventory of fuel salt from the primary loop" and must "incorporate sufficient passive cooling capability to accommodate the thermal load of a fuel salt inventory that contains a fresh, equilibrium inventory of fission products" (Electric Power Research Institute 2015, 3-22). The specific mechanism for passive cooling is not proposed, though Appendix B presents the options of water, liquid metals, fused salts, organics or gases (Electric Power Research Institute 2015, B-2). Though Flibe Energy dismissed the impact of this hazard as "very low due to passive heat removal system," a review of the literature suggests that proper design of the fuel salt drain tank is crucial to its integrity.

During design and operation of the MSRE at Oak Ridge National Laboratory in the 1960s, engineers performed analyses to determine the feasibility of criticality being achieved in the fuel salt drain tanks. In its original configuration, the MSRE was fueled by ^{235}U and included four salt drain tanks: two for fuel salt, one for coolant salt and one for flush salt. Designers acknowledged the remote possibility that concentration of ^{235}U may increase as the fuel salt freezes and could result in criticality being achieved within the drain tank (Robertson 1965, 220). More specifically, it was demonstrated empirically that equilibrium cooling of fuel salt mixtures resulted in segregation of UF_4 from the fuel salt carrier and a subsequent concentration by a factor of three in the last phases to freeze (Beall, et al. 1964, 230). Although the ^{235}U-loaded MSRE required concentrations of four-times normal or higher to achieve criticality in the drain tank, this risk was obviated completely by splitting the contents of the fuel salt into two separate drain tanks in the MSRE. Moreover, bayonet heat exchanger thimbles carrying liquid water would operate by natural circulation, with liquid water turning to steam while removing heat from the fuel salt (Beall, et al. 1964, 30). This system was meant to preclude the possibility of criticality in the fuel salt drain tanks. However, with the introduction of ^{233}U to the MSRE in 1968, scientists and engineers were forced to revisit this possibility.

It was determined that the nuclear reactivity of ^{233}U in the fuel salt drain tanks is higher than that of ^{235}U, which required further analysis on the credibility of criticality and structural failure within the drain tanks. Engineers determined that the most reactive situation would occur if the entire contents of the fuel salt were present in a single drain tank with thimbles full of water, resulting in higher concentrations of uranium *and* the presence of a moderator in the form of liquid water. According to calculations, fuel salt at 1200°F with no water in the thimbles would keep the drain tank subcritical with a multiplication factor (k_{eff}) of 0.85. However, under the most reactive conditions with fuel salt at room temperature and water in the thimbles, criticality in the fuel salt drain tank could be achieved with k_{eff} = 1.0. Splitting the uranium inventory into two drain tanks resulted in a subcritical configuration with a maximum k_{eff} =0.88 (Haubenreich, et al. 1968, 68).

Though the temperature rise may not be significant (Haubenreich, et al. 1968, 69), the concern is valid that criticality may be achieved in the drain tanks under the right conditions. Elements key to successful operation of the drain tank will be detailed neutronic calculations on the feasibility of criticality, estimated temperature changes under conditions of criticality, and selection of appropriate passive cooling systems. Currently, the Flibe Energy LFTR is considering use of a convective air cooling system similar to the direct reactor auxiliary cooling system (DRACS) loop which is also used in the fluoride-cooled high temperature reactor (FHR) (Electric Power Research Institute 2015, 4-15). Use of air convection for passive cooling removes the possibility of water providing neutron moderation in the drain tank. Additionally, the LFTR design must consider the impact of the graphite moderator entering the drain tank following breakage of one or more graphite tubes. This addition of moderator to the drain tank contents may also enable criticality and a temperature rise in the drain tanks. Because there are credible scenarios under which criticality and temperature rise may occur in the drain tanks, inadequate cooling of the drain tanks represents a real hazard, which may lead to the release of radioactivity from the reactor core to the surrounding environment.

4.1.4 FAILED FREEZE VALVE OR OBSTRUCTION OF THE PIPING TO THE DRAIN TANK

In conjunction with the drain tanks, the freeze valve and associated drain piping represent some of the most important passive safety features of the LFTR design. During many design-based casualties, safety can be shown for the LFTR when the freeze valve melts, allowing the fuel salt inventory to separate from the graphite moderator and drain to subcritical tanks (Electric Power Research Institute 2015, 3-8). Unfortunately, the EPRI evaluation of the Flibe Energy LFTR does not thoroughly address the hazards associated with obstruction of drain piping and focuses only on the occurrence of this casualty during reactor fill. Potential hazard scenarios that obstruct this crucial flowpath must be considered in detail for their impact on overall safety and integrity of the LFTR, especially during reactor operations and casualty scenarios.

Depending on the type of obstruction, this hazard may result in different types of failure. Consider an obstruction in the form of partially thawed freeze plug. This may result in the fuel salt remaining in the reactor core, where moderation occurs and the nuclear reaction can be sustained. Thermal expansion of the fuel salt could result in rapid pressurization of the primary system as described in Section 4.1.1. In the Molten Salt Breeder Reactor design from Oak Ridge National Laboratory, an allowance was made for overfilling or thermal expansion, which caused fuel salt to overflow through a standpipe into the fuel salt drain tanks (Robertson, Smith, et al. 1968, 47). However, the LFTR still has insufficient design features allowing for thermal expansion of the fuel salt, relying only on the off-gas handling lines as overflow for expanding fuel salt (Electric Power Research Institute 2015, A-10 and A-36).

Additionally, criticality may be maintained in the core region due to the inability to drain the fuel salt to a subcritical configuration. Criticality of course means a rise in fuel salt temperatures with the inability to drain the core to the drain tanks. In one study, thermal calculations were conducted to estimate the temperature rise due only to decay heat, not including fission, during a casualty where fuel salt was not drained. It was determined that fuel salt would reach ~1200°C within 8 minutes and potentially cause core damage if the fuel salt could not be drained (Brovchenko, et al. 2013, 338) as seen in Figure 5.

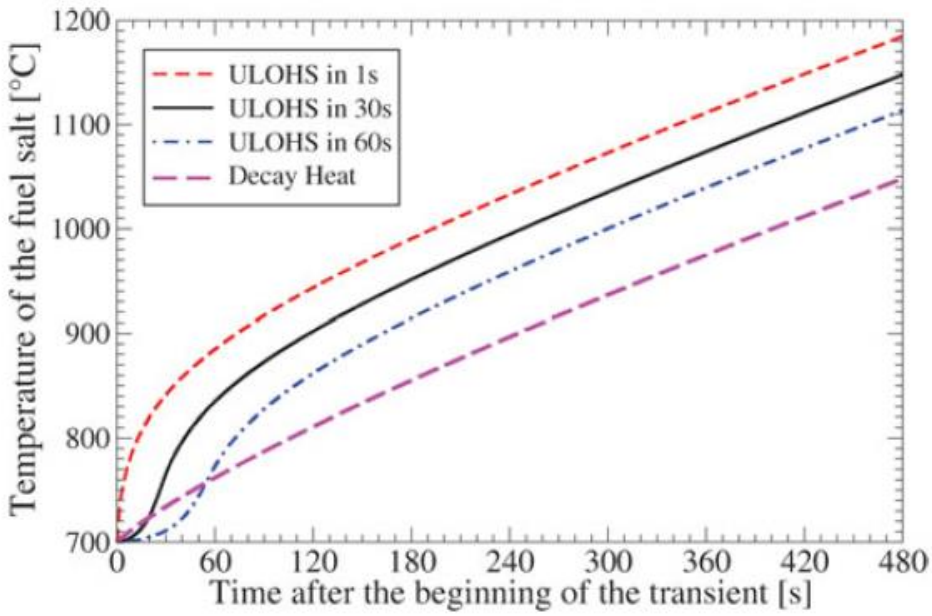

Figure 6. Temperature of the Fuel Salt during an Unprotected-loss-of-heat-sink (Brovchenko, et al. 2013, 338)

Finally, research indicates that freeze valve failure could also occur if a partial thaw results in piping rupture due to thermal expansion of trapped fuel salt (Beall, et al. 1964, 231). Though rupture of the drain tank piping would ultimately drain the fuel salt into the LFTR catch pan and direct the salt to drain tanks, this casualty still constitutes core damage due to the release of radioactive material from the primary plant boundary.

Another mode of failure would be graphite obstruction of the drain line. The Flibe Energy LFTR design team did briefly address this concern in Table A-33 by asking "What if a piece of graphite enters the drain tank in the event of an emergency drain?" (Electric Power Research Institute 2015, A-36). In addition to the effects described above, namely thermal expansion of the undrained fuel salt and supercriticality in the core region, graphite blockage of the drain piping presents its own unique challenges. Unlike a partially thawed freeze plug, graphite blockage represents a moderator material past which the fuel salt would flow. This could result in fissions taking place in the drain piping, which is not rated to accommodate heat increase in the fuel salt due to fissions outside the reactor core. Further, the graphite moderator may also make its way into the drain tanks introducing a neutron-moderating medium to what is intended to be a subcritical tank. The Flibe Energy LFTR design description briefly addresses these concerns, citing that graphite floats in the proposed fuel salt material (Electric Power Research Institute 2015, A-36) and thermal/hydraulic transport phenomena must be better understood to prevent this type of drain piping blockage.

Although most experts agree on the inherent safety of the molten salt reactor due to the ability to rapidly drain the fuel contents to a dedicated subcritical tank, the possibility of obstructing the drain tank piping must be considered for its potential impact on LFTR integrity and safety. Obstruction of the drain tank piping may result in thermal expansion or supercritical core conditions, rupture of the drain piping, and the possibility of transporting the graphite moderator to the drain tanks where criticality outside the core region must be prevented. For these reasons, blockage of the drain piping is considered a credible hazard with the potential to result in core damage.

4.2 REFERENCE TO EPRI AND NRC INITIATING EVENT LISTS AND PREVIOUS PSAs

In this section EPRI and Nuclear Regulatory Commission initiating event lists are considered for their applicability to the LFTR. Additionally, a review of previous PSAs for Generation IV Advanced Reactors is conducted to determine the applicability of initiating events to the LFTR. Unfortunately, due to the lack of operating experience of molten salt reactors, the EPRI and NRC have only published data for initiating events at PWR and BWR power plants. Still, some of the initiating events developed in these reports may impact the safety and integrity of the LFTR and will be considered for their applicability in this analysis.

Using hundreds of reactor-years of operating experience, the EPRI and NRC have developed several transient and special initiating events lists that include probabilities and frequencies of occurrence based on data from operating experience. Understanding that PWR and BWR technology is vastly different from that proposed in the MSR, many of the initiating events compiled in these lists do not apply and will not be considered. However, there are some fundamental similarities between LWRs and MSRs in how the LFTR is expected to operate. Of the world's 441 operating or operational nuclear power plants, 282 are PWRs, 78 are BWRs and the remaining 81 are heavy-water, gas-cooled, graphite-moderated or fast-breeder reactors (Nuclear Power Reactors 2017). To limit the scope of review of EPRI and NRC initiating event lists, this study considers initiating event lists from only PWRs as they are more prevalent and have significantly more operating experience than other types of reactors.

4.2.1 EPRI AND NRC INITIATING EVENT LISTS FOR PRESSURIZED WATER REACTORS

Table 4 represents a list of generic EPRI Initiating Events adjacent to the results of the Oconee Nuclear Station Probabilistic Risk Assessment (International Atomic Energy Agency 1993, 28-37).

EPRI List of Initiating Events for PWR	Oconee PRA List of Special and Transient IE
1. Loss of reactor coolant flow (One loop)	1. Rod drop
2. Uncontrolled rod withdrawal	2. Inadvertent rod withdrawal
3. Problems with control drive mechanism/rod drop	3. Rod ejection
4. Leakage from control rods	4. Inadvertent boration or dilution
5. Leakage in primary system	5. Reactor trip
6. Low pressurizer pressure	6. Cold water addition
7. Pressurizer leakage	7. Reactor coolant pump trip
8. High pressurizer pressure	8. Reactor coolant pump seizure
9. Inadvertent safety injection signal	9. Flow channel blockage
10. Containment pressure problems	10. Loss of main feedwater
11. Chemical and volume control system malfunction – boron dilution	11. Excess feedwater
12. Pressure, temperature, power imbalance – rod-position error	12. Loss of condenser vacuum

Table 4. EPRI and Oconee Nuclear Station List of IEs for PWR
(International Atomic Energy Agency 1993, 28-37)

EPRI List of Initiating Events for PWR	Oconee PRA List of Special and Transient IE
13. Startup of inactive coolant pump	13. Inadequate main feedwater
14. Total loss of RCS flow	14. Feedwater or condensate line breaks
15. Loss or reduction in feedwater	15. Steam line breaks
16. Loss of reactor coolant flow (One loop)	16. Rod drop
17. Uncontrolled rod withdrawal	17. Inadvertent rod withdrawal
18. Problems with control drive mechanism/rod drop	18. Rod ejection
19. Leakage from control rods	19. Inadvertent boration or dilution
20. Leakage in primary system	20. Reactor trip
21. Low pressurizer pressure	21. Cold water addition
22. Pressurizer leakage	22. Reactor coolant pump trip
23. High pressurizer pressure	23. Reactor coolant pump seizure
24. Inadvertent safety injection signal	24. Flow channel blockage
25. Containment pressure problems	25. Loss of main feedwater
26. Chemical and volume control system malfunction – boron dilution	26. Excess feedwater
27. Pressure, temperature, power imbalance – rod-position error	27. Loss of condenser vacuum
28. Startup of inactive coolant pump	28. Inadequate main feedwater
29. Total loss of RCS flow	29. Feedwater or condensate line breaks
30. Loss or reduction in feedwater (one loop)	30. Steam line breaks
31. Total loss of feedwater (all loops)	31. Turbine and control valve malfunctions
32. Total or partial closure of main steam isolation valve (one loop)	32. Turbine-bypass valve inadvertent opening
33. Closure of all main steam isolation valves (all loops)	33. Turbine malfunction
34. Increase in feedwater flow (one loop)	34. Loss of condenser circulating water
35. Increase of feedwater flow (all loops)	35. Small reactor coolant pipe breaks
36. Feedwater flow instability (operator error)	36. Large reactor coolant pipe breaks
37. Feedwater flow instability (mechanical)	37. Inadvertent pilot-operated relief valve or safety-valve opening
38. Loss of condensate pumps (one loop)	38. Reactor coolant pump seal failure
39. Loss of condensate pumps (all loops)	39. Control rod drive seal failure
40. Loss of condenser vacuum	40. Interfacing system loss of coolant
41. Steam-generator leakage	41. Reactor vessel rupture
42. Condenser leakage	42. Steam generator tube leakage/rupture

Table 4. Continued

EPRI List of Initiating Events for PWR	Oconee PRA List of Special and Transient IE
43. Miscellaneous leakage in secondary system	43. Charging exceeds letdown
44. Sudden opening of steam relief valves	44. Letdown exceeds charging
45. Loss of circulating water	45. Inadvertent high pressure injection
46. Loss of component cooling	46. Failure on or off of pressurizer heaters
47. Loss of service-water system	47. Failure on or off of pressurizer spray
48. Turbine trip, throttle valve closure, EHC problems	48. Loss of off-site power
49. Generator trip or generator-caused faults	49. Loss of power to necessary systems
50. Loss of all off-site power	50. Loss of power to control systems
51. Pressurizer spray failure	51. Loss of service water
52. Loss of power to necessary plant systems	52. Loss of component cooling
53. Spurious trips – cause unknown	53. Loss of instrument air
54. Automatic trip – no transient condition	54. Integrated control system power
55. Manual trip – no transient condition	55. Fires affecting necessary systems
56. Fires within the plant	56. Internal flooding affecting necessary systems
	57. Generator faults
	58. Grid disturbances
	59. Administrative shutdowns
	60. Main steam isolation valve closures
	61. Anticipated transient without scram

Table 4. Continued

At the onset, many of the above initiating events can be discarded as not applicable to molten salt reactor technology. These include pressurizer, feedwater, main steam isolation valve, steam generator and condenser casualties. Still, many of these systems have parallels in the LFTR design and thus the initiating event may be modified such that it is applicable to the MSR design. For example, while there is no feedwater system in the LFTR, the equivalent system is the gaseous CO_2 to be heated for use in the closed Brayton-cycle power conversion system. In PWR applications, loss of feedwater represents a loss of heat sink to the reactor coolant system just as a loss of CO_2

flow would represent a loss of heat sink for the LFTR coolant salt. Using this approach, initiating events in Table 4 were screened and altered to ensure applicability to molten salt reactor technology. The following initiating events were derived from Table 4 giving special consideration to specific LFTR subsystems and components:

- Loss of reactor coolant flow (loss of fuel salt flow)
- Inadvertent rod withdrawal or rod ejection (liquid control rod system failure)
- Inadvertent control rod injection (rod drop)
- Chemical and volume control system malfunction (Off-gas handling and chemical processing systems)
- Startup of inactive fuel salt pump or coolant salt pump
- Increase or decrease in coolant salt flow
- Increase or decrease in CO_2 flow in power conversion system (compressor failure)
- Sudden opening of coolant salt relief valves or CO_2 relief valves
- Loss of off-site power
- Loss of power to necessary plant systems
- Loss of component cooling systems
- Loss of instrument air
- Loss of integrated control system power
- Flow channel blockage
- Cold fuel salt addition
- Automatic or manual reactor trips with no transient condition
- Fires or internal flooding affecting plant systems

Further review of several NRC Reports resulted in a compilation of the following applicable initiating events for consideration in this study. Table 5 consists of initiating events gathered from resources including the NUREG-1150, NUREG/CR-3862, NUREG/CR-6928 and NUREG/CR-5750.

Initiating Event	Reference
Loss of on-site and off-site AC power and failure of auxiliary cooling systems, high pressure injection system or reactor coolant pump seal failure	(U.S. Nuclear Regulatory Commission NUREG-1150 1990, 3-1)
Steam generator tube rupture followed by depressurization of the reactor coolant system	
Seismic events	
Fire within the plant	
Loss of vital AC or DC electric bus	(U.S. Nuclear Regulatory Commission NUREG/CR-6928 2007, Appendix D)
Loss of component cooling water system	
Loss of condenser heat sink at PWRs	
High flux due to rod withdrawal at startup	(Mackowiak, Gentillon and Smith 1985, Table 5-7)
Trip of one or more feedwater pumps	
Pressurizer spray fails open/closed	
Reactivity control imbalance	(U.S. Nuclear Regulatory Commission 1999, Table 2-1)
Reactor coolant system high pressure	
Reactor coolant system low pressure	

Table 5. Select U.S. Nuclear Regulatory Commission Initiating Events

The following initiating events were derived from Table 5 giving special consideration to specific LFTR subsystems and components:

- Loss of on-site and off-site AC power resulting in failure of auxiliary cooling systems
- Coolant salt heat exchanger rupture followed by pressurization of the fuel salt loop
- Seismic events
- Loss of vital AC or DC electric bus
- Reactivity control imbalance (chemical processing plant malfunction or fuel addition malfunction)
- Reactor coolant system high pressure

4.2.2 REFERENCE TO PREVIOUS PSAs FOR GENERATION IV NUCLEAR REACTORS

Finally, a review of existing PSAs was conducted to ensure completeness of the proposed initiating event lists. A review of the literature reveals that there are very few mature probabilistic safety assessments for molten salt reactors. Table 6 includes those initiating events gleaned from existing Generation IV Reactor PSAs that are applicable to the Flibe Energy LFTR. Special care was given not to duplicate initiating events already developed from Tables 4 and 5 above.

Initiating Event	**Reference**
Secondary shutdown system mistakenly inserted	(Zuo, et al. 2017, 678)
Core geometry failure	
Core coolant flow channel or area is blocked	
Secondary coolant flow channel blockage	
Air cooling tower ventilation doors get stuck	(Zhang 2016, 395-396)
Loss of off-site power without a scram	
Radioactive gas waste disposal system leakage or breakage	
Radioactive liquid waste disposal system leakage or breakage	

Table 6. Initiating Event List compiled from Previous Generation IV PSAs

The initiating events derived from Tables 4 and 5 and those identified in Table 6 are evaluated by logical classification in Chapter 5 to determine their impact on the Flibe Energy LFTR.

CHAPTER 5

RESULTS AND DISCUSSION – LOGICAL CLASSIFICATION

In Chapter 4, technical evaluation of the LFTR plant revealed four principle hazards whose realization could lead to core damage and the release of radioactive isotopes from the reactor plant boundary. Review of existing initiating event lists and reference to previous PSAs provides a starting point for logical classification of specific initiating events impacting the LFTR.

5.1 FAULT TREE ANALYSIS

Fault Tree Analysis is defined in the Fault Tree Handbook used by the Nuclear Regulatory Commission as follows:

> "Fault tree analysis is a deductive failure analysis which focuses on one particular undesired event and which provides a method for determining causes of this event. The undesired event constitutes the top event in a fault tree diagram constructed for the system, and generally consists of a complete, or catastrophic failure" (Vesely, et al. 1981, III-3).

Four fault trees were constructed, each hosting one of the four principle hazards as the undesired event at the top of the fault tree. Events were assigned alpha-numeric codes in each fault tree which aided in developing Boolean expressions for minimal cut set determination. The key for fault tree coding is as follows:

TX_Y: Top event

EX_Y: Intermediate event

CX_Y: Conditioning event

BX_Y: Basic event (initiating event)

where X represents the top event and Y is an index for the quantity of events attributed to each top event. The X-variable takes on values as follows:

1. Uncontrolled rod withdrawal
2. Breakage of one or more graphite tubes

3. Improper or inadequate cooling of the drain tanks
4. Obstruction of the drain tank piping.

Boolean expressions were formulated utilizing the rules described in the *Fault Tree Handbook* (Vesely, et al. 1981, IX-7) and using the laws of absorption, distribution and idempotence. From the Boolean expressions, minimal cut sets were determined that represent the “smallest combination of component failures which, if they occur, will cause the top event to occur” (Vesely, et al. VII-15). A qualitative analysis was conducted to evaluate vulnerabilities and areas for improvement in the LFTR design. Appendix C contains the derivation of minimal cut sets for each hazard.

5.1.1 FAULT TREE ANALYSIS FOR UNCONTROLLED ROD WITHDRAWAL

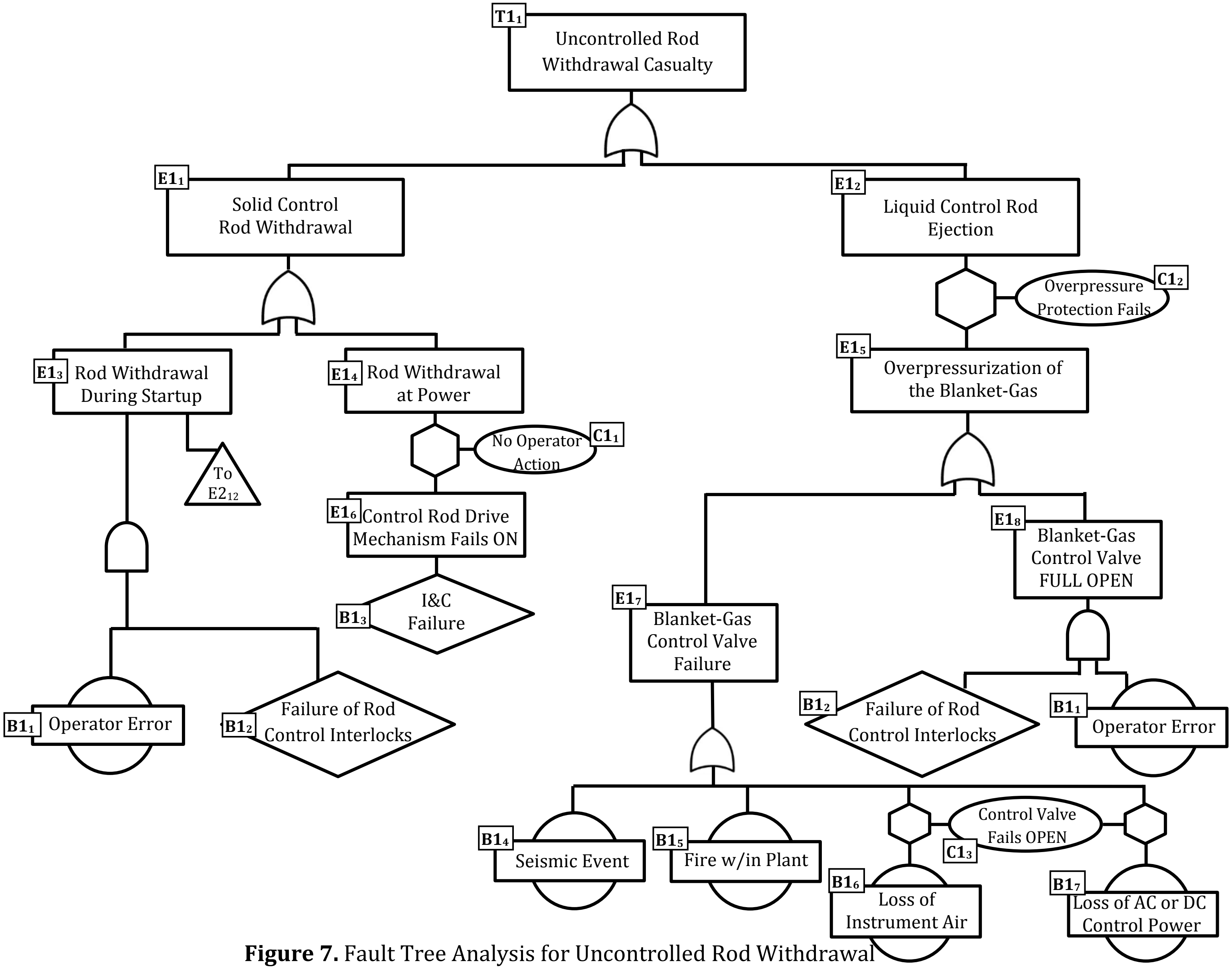

Figure 7. Fault Tree Analysis for Uncontrolled Rod Withdrawal

$T1_1$ – Uncontrolled Rod Withdrawal Casualty
$E1_1$ – Continuous Solid control rod withdrawal during power or startup operations
$E1_2$ – Liquid control rod ejection during any operating condition
$E1_3$ – Continuous Solid control rod withdrawal during startup
$E1_4$ – Continuous Solid control rod withdrawal during power operations
$E1_5$ – Overpressurization of the blanket gas used to maintain liquid control rod level
$E1_6$ – Control rod drive mechanism fails on due to instrumentation and control failure
$E1_7$ – Blanket-gas control valve mechanically fails FULL OPEN
$E1_8$ – Blanket-gas control valve is manually FULL OPEN
$C1_1$ – No operator action is taken to counter an unexpected control rod withdrawal casualty
$C1_2$ – Overpressure protection fails in the blanket-gas control system
$C1_3$ – Blanket-gas control valve will FAIL OPEN on loss of instrument air or AC or DC control power (based on engineering design)
$B1_1$ – Operator error or procedural noncompliance causes a continuous rod withdrawal casualty
$B1_2$ – Rod control interlocks fail to operate to prevent rod withdrawal
$B1_3$ – Instrumentation and control circuitry failure causes unexpected rod withdrawal
$B1_4$ – Seismic event causes mechanical failure of the blanket-gas control valve
$B1_5$ – Fire within the plant causes mechanical failure of the blanket-gas control valve
$B1_6$ – Loss of instrument air to the blanket-gas control valve causes it to FAIL OPEN
$B1_7$ – Loss of AC or DC control power causes the blanket-gas control valve to FAIL OPEN

Minimal Cut Set:

$$T1_1 = (B1_1 \bullet B1_2)+(B1_3 \bullet C1_1)+(B1_4 \bullet C1_2)+(B1_5 \bullet C1_2)+(B1_2 \bullet C1_2)+(B1_1 \bullet C1_2)+ (B1_6 \bullet C1_2 \bullet C1_3)+(B1_7 \bullet C1_2 \bullet C1_3)$$

Results:
Zero single-component minimal cut sets
Six double-component minimal cut sets
Two triple-component minimal cut sets

Figure 7 and Boolean reduction indicate that there are no single-component minimal cut sets for an uncontrolled rod withdrawal casualty, indicating that based on the availability of design data, it is expected that multiple initiating event conditions must be present to enable the uncontrolled withdrawal of either solid control rods or liquid control rods.

Further evaluation of the double-component minimal cut sets reveals that four of the six require a conditioning event, either $C1_1$ or $C1_2$ to cause the top event. These conditioning events are identified as human error (i.e. the operator does not take required actions during a casualty scenario) or material failure (overpressure protection of liquid control rod blanket-gas fails). The first condition is not further evaluated in this study as it implies human reliability analysis, which is beyond the scope of this thesis. The second conditioning event, however, reveals an important vulnerability in the liquid control rod design: although the design "fails open" and allows the liquid control rod to shutdown the reactor on loss of electrical power, there is insufficient design detail to prove that the risk of liquid control rod ejection is properly mitigated. The triple-component minimal cut sets introduce yet another conditioning event, namely the condition that a loss of power or loss of instrument air would cause the blanket-gas control valve to fail OPEN. This of course is a design feature and can be engineered to have a probability of identically zero. If the control mechanism for the blanket-gas is engineered to fail SHUT, the risk is categorically prevented. However, until further design fidelity is provided for the liquid control rod mechanism, this condition is considered to impact the likelihood of uncontrolled rod withdrawal in the Flibe Energy LFTR.

The list of initiating events found to cause uncontrolled rod withdrawal includes:

1. Operator error
2. Failure of rod control interlocks
3. Instrumentation and control failure
4. Seismic event
5. Fire within the plant
6. Loss of instrument air
7. Loss of AC or DC control power

Note that failure of rod control interlocks and instrumentation and control failure appear as "undeveloped events" in the diamond shapes because detailed design specifications are not yet available for the Flibe Energy LFTR. In designing this crucial reactor safety logic and circuitry, engineers should ensure that rod control interlocks and I&C circuitry address the risks identified above, where failure of either may cause uncontrolled rod withdrawal.

5.1.2 FAULT TREE ANALYSIS FOR BREAKAGE OF ONE OR MORE GRAPHITE TUBES

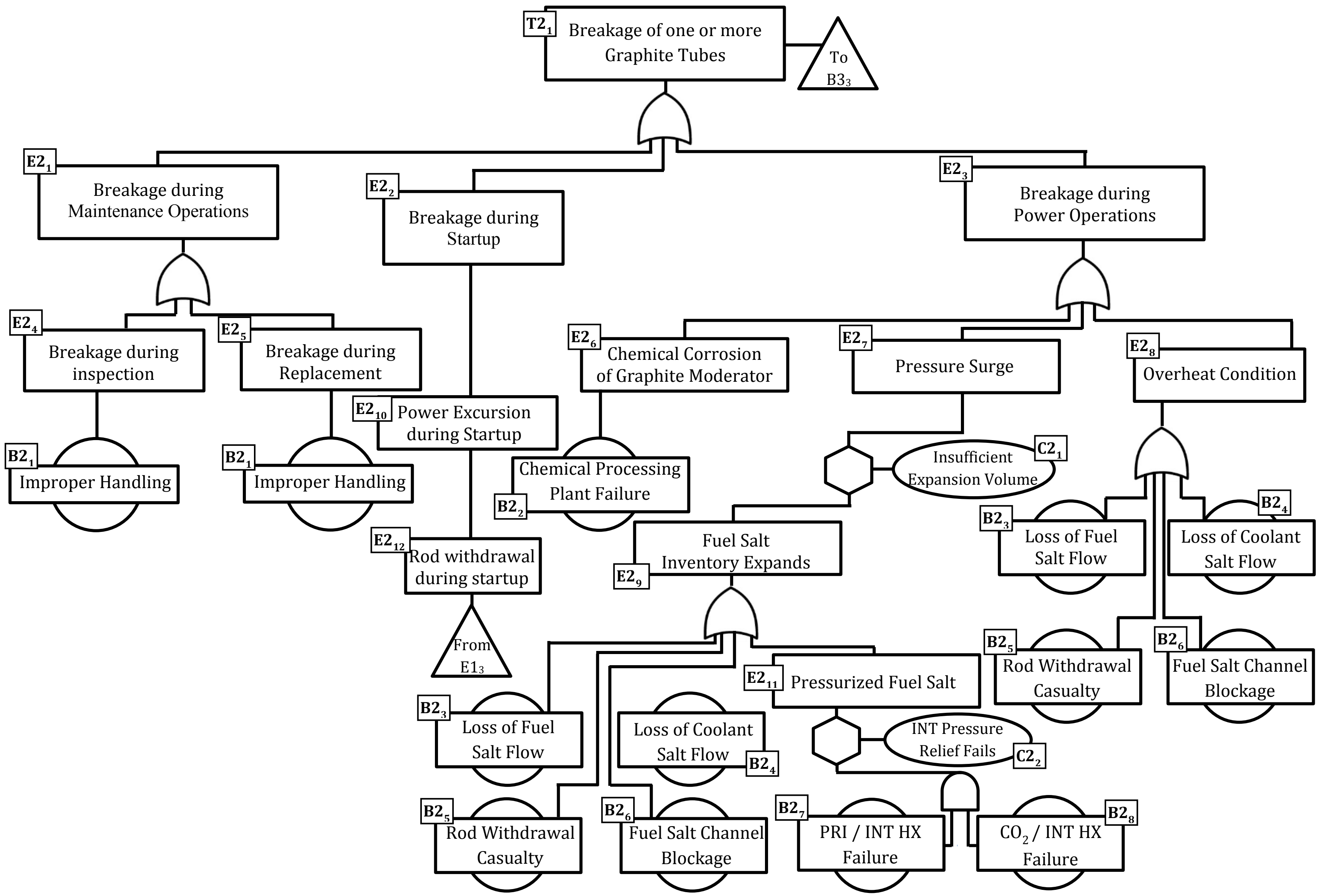

Figure 8. Fault Tree Analysis for Breakage of one or more Graphite Tubes

$T2_1$ – Breakage of one or more Graphite Tubes
$E2_1$ – Breakage of Graphite Tubes during Maintenance Operations
$E2_2$ – Breakage of Graphite Tubes during Reactor Startup
$E2_3$ – Breakage of Graphite Tubes during Power Operations
$E2_4$ – Breakage during Routine Inspection
$E2_5$ – Breakage during Replacement of Graphite Tubes
$E2_6$ – Chemical Corrosion of the Graphite Moderator during Power Operations
$E2_7$ – Pressure surge causing breakage of graphite during Power Operations
$E2_8$ – Overheat condition causing failure of the graphite moderator during Power Operations
$E2_9$ – Fuel salt inventory undergoes thermal expansion during Power Operations
$E2_{10}$ – Power excursion causing rapid fuel expansion during Reactor Startup
$E2_{11}$ – Pressurized Fuel Salt in the Primary Loops
$E2_{12}$ – Rod Withdrawal Casualty during Reactor Startup
$C2_1$ – Insufficient Expansion Volume to accommodate fuel salt inventory expansion
$C2_2$ – Intermediate Loop Pressure Relief Valves fail to function
$B2_1$ – Improper handling of Graphite Tubes during Inspection or Replacement
$B2_2$ – Chemical Processing Plant Failure leading to corrosion of the Graphite
$B2_3$ – Loss of Fuel Salt Flow
$B2_4$ – Loss of Coolant Salt Flow
$B2_5$ – Continuous or Rapid Rod Withdrawal Casualty (Liquid or Solid Control Rod Configuration)
$B2_6$ – Fuel Salt Channel Blockage
$B2_7$ – Primary to Intermediate Loop Heat Exchanger Failure
$B2_8$ – Intermediate Loop to CO_2 Heat Exchanger Failure

Minimal Cut Set:

$$T2_1 = B2_1 + B2_2 + B2_3 + B2_4 + B2_5 + B2_6 + (B1_1\bullet C1_1) + (B2_3\bullet C2_1) + (B2_4\bullet C2_1) + (B2_5\bullet C2_1) + (B2_6\bullet C2_1) + (B2_7\bullet B2_8\bullet C2_1\bullet C2_2)$$

Results:
Six single-component minimal cut sets
Five double-component minimal cut sets
Zero triple-component minimal cut sets
One quadruple-component minimal cut set

Figure 8 and Boolean reduction indicate that there are several single-component minimal cut sets identified above which may cause breakage of one or more graphite tubes. The first single-component failure involves improper handling of graphite during maintenance, inspection or replacement. This risk is primarily driven by human error; therefore, further human reliability analysis is not included. It is evident that proper handling of graphite must be a priority to minimize the possibility of introducing loose graphite to the primary fuel salt loops.

The chemical processing plant is another vulnerability to the integrity of graphite components within the reactor core. The chemical processing plant described in the Flibe Energy LFTR System Design Description provides great detail on fuel processing, but offers little insight on potential corrosion of Hastelloy-N and graphite structural materials exposed to molten salts. A review of the literature indicates that the threat of corrosion and structural failure due to exposure to high-temperature, high-neutron flux conditions present in molten salt reactor cores is very real (Lane 1958, 623). Because there are many modes of failure of the chemical processing plant that may make the LFTR susceptible to breakage of graphite tubes, this basic initiating event was found to be of the first-order pending further design detail and studies demonstrating compatibility of graphite with the proposed fuel and blanket salts.

The remaining single-component minimal cut sets include loss of fuel salt or coolant salt flow, flow channel blockage and rod withdrawal casualty. All of these initiating events share the same mechanism of graphite failure – overheating of core material. These same initiating events reappear as double-component minimal cut sets because, in addition to the potential for overheat in the core these will cause a pressure surge due to thermal expansion of the fuel salt unless there is sufficient expansion volume engineered into the LFTR design. These basic initiating events leading to breakage of graphite tubes can be mitigated by design of a surge capacitor or expansion volume in the core.

Of course, the least likely hazard is a multi-dimensional failure in that the extreme pressures from the closed Brayton-cycle supercritical CO_2 power conversion system are translated back through the intermediate coolant salt to the primary fuel salt by failure of heat exchangers and inoperable pressure reliefs. Accordingly, this hazard takes the form of a quadruple-component failure. Although unlikely, the consequences of exposing the fuel salt loop and reactor vessel to the high CO_2 pressures would be catastrophic, and several redundancies are recommended in the engineering design to preclude the possibility of the power conversion system pressures from reaching the reactor vessel. These redundancies could include but are not limited to pressure relief systems in the intermediate coolant salt and primary fuel salt loops, high-pressure rated intermediate and primary loops, and hardened heat exchangers.

The list of initiating events found to lead to breakage of one or more graphite tubes includes:

1. Improper handling of Graphite Tubes during Inspection or Replacement
2. Chemical Processing Plant Failure leading to corrosion of the Graphite
3. Loss of Fuel Salt Flow
4. Loss of Coolant Salt Flow
5. Continuous or Rapid Rod Withdrawal Casualty (Liquid or Solid Control Rod Configuration)
6. Fuel Salt Channel Blockage
7. Primary to Intermediate Loop Heat Exchanger Failure
8. Intermediate Loop to CO_2 Heat Exchanger Failure

5.1.3 FAULT TREE ANALYSIS FOR IMPROPER OR INADEQUATE COOLING OF DRAIN TANKS

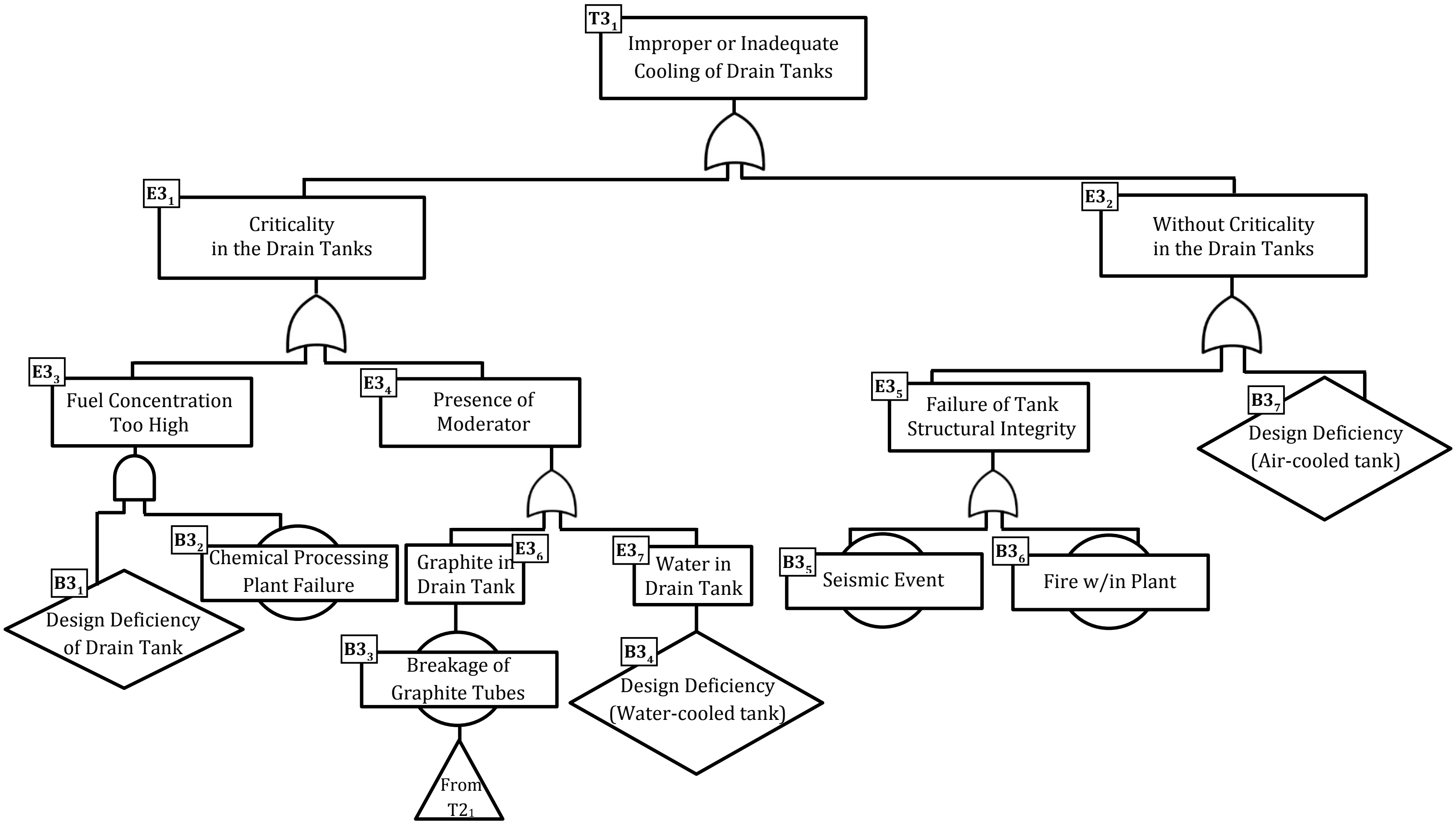

Figure 9. Fault Tree Analysis for Improper or Inadequate cooling of the Drain Tanks

$T3_1$ – Improper or Inadequate Cooling of the Drain Tanks
$E3_1$ – Criticality causes heat increase in the Drain Tanks following transfer of fuel salt
$E3_2$ – No criticality occurs in the Drain Tanks following transfer of fuel salt
$E3_3$ – Fuel concentration is too high due to freezing and contraction of fuel salt
$E3_4$ – Neutron Moderator is present in the Drain Tanks to facilitate Criticality
$E3_5$ – Failure of structural integrity of the Drain Tanks following transfer of the fuel salt
$E3_6$ – Graphite is transferred to the Drain Tanks during drainage of the fuel salt
$E3_7$ – Water is present in the Drain Tanks following transfer of the fuel salt
$B3_1$ – Design Deficiency in Drain Tank geometry enables critical fuel concentrations to exist as the fuel freezes
$B3_2$ – Chemical Processing Plant Failure leading to excessive fuel concentrations in the drained fuel salt
$B3_3$ – Breakage of Graphite Tubes causes transfer of neutron moderating graphite into the Drain Tanks
$B3_4$ – Design Deficiency in selection of Drain Tank cooling mechanism could cause water to be present following fuel salt transfer
$B3_5$ – Seismic event before or during fuel salt transfer to the Drain Tanks
$B3_6$ – Fire within the plant before or during fuel salt transfer to the Drain Tanks
$B3_7$ – Design Deficiency in the selection of Drain Tank cooling mechanism causes the air-cooled system to fail

Minimal Cut Set:

$$T3_1 = B2_1 + B2_2 + B2_3 + B2_4 + B2_5 + B2_6 + B3_4 + B3_5 + B3_6 + B3_7 + (B1_1 \bullet C1_1) + (B3_1 \bullet B3_2) + (B2_3 \bullet C2_1) + (B2_4 \bullet C2_1) + (B2_5 \bullet C2_1) + (B2_6 \bullet C2_1) + (B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2)$$

Results:
Ten single-component minimal cut sets
Six double-component minimal cut sets
Zero triple-component minimal cut sets
One quadruple-component minimal cut set

Improper or inadequate cooling of the drain tanks is a fault tree impacted by several variables as a result of the lack of design detail provided for the Flibe Energy LFTR. The reference technologies of the MSRE and MSBR at Oak Ridge National Laboratory recognized the possibility of criticality in the drain tanks and made some design changes to prevent this hazard. The LFTR, however, does not address this possibility in sufficient detail. As such, there are many single-component minimal cut sets that may lead to criticality or inadequate cooling until further design detail is available.

Figure 9 and Boolean reduction indicate that the first six basic initiating events and four multi-component minimal cut sets listed in the $T3_1$ minimal cut set are shared with $T2_1$. Because breakage of graphite tubes may directly contribute to criticality inside the drain tanks, all minimal cut sets leading to breakage of graphite tubes are included in the basic initiating events for inadequate cooling of the drain tanks.

Unique to the $T3_1$ minimal cut set are seismic events and fires impacting structural integrity of the drain tanks or the passive cooling system; chemical processing plant failures resulting in high fuel concentration; or design deficiencies in the drain tanks or selected passive cooling mechanism that compromise the cooling capacity. A common theme across many of the multi-dimensional failures is the potential for criticality and rising temperatures of the fuel salt being stored in the drain tanks. Haubenreich et al. recognized the credible hazard that criticality may exist outside the core region depending on fuel enrichment, fuel concentration, and the presence of a neutron moderator in the drain tank (1968, 68). In the absence of any other design information such as number or configuration of drain tanks, and without sufficient discussion of the proposed passive cooling design (whether water-based or air-based) several minimal cut sets exist that begin with "design deficiency" and lead to inadequate cooling of the tanks.

It follows that several of the initiating events for this primary hazard are undeveloped events. Specifically, there are three undeveloped events considered only as "design deficiencies." These include a design deficiency that would allow high enough fuel concentrations to cause criticality while the salt freezes (Haubenreich, et al. 1968, 68); a design deficiency in which water-cooled drain tanks add moderator to enable nuclear fission; and a design deficiency in an air-cooled system that simply doesn't have the required passive cooling capacity to prevent structural damage and release of fuel salt from the boundary. Until further design detail is offered by Flibe Energy and proof-of-concept work is complete, these undeveloped events will stand as initiating events for inadequate drain tank cooling.

The list of initiating events found to lead to improper or inadequate cooling of the drain tanks includes:

1. Design Deficiency in Drain Tank geometry
2. Chemical Processing Plant Failure
3. Breakage of Graphite Tubes (moderator in the drain tanks)
4. Design Deficiency in the selection of Drain Tank cooling mechanism (water-cooled)
5. Seismic event
6. Fire within the plant
7. Design Deficiency in the selection of Drain Tank cooling mechanism (air-cooled)

5.1.4 FAULT TREE ANALYSIS FOR OBSTRUCTION OF THE DRAIN PIPING

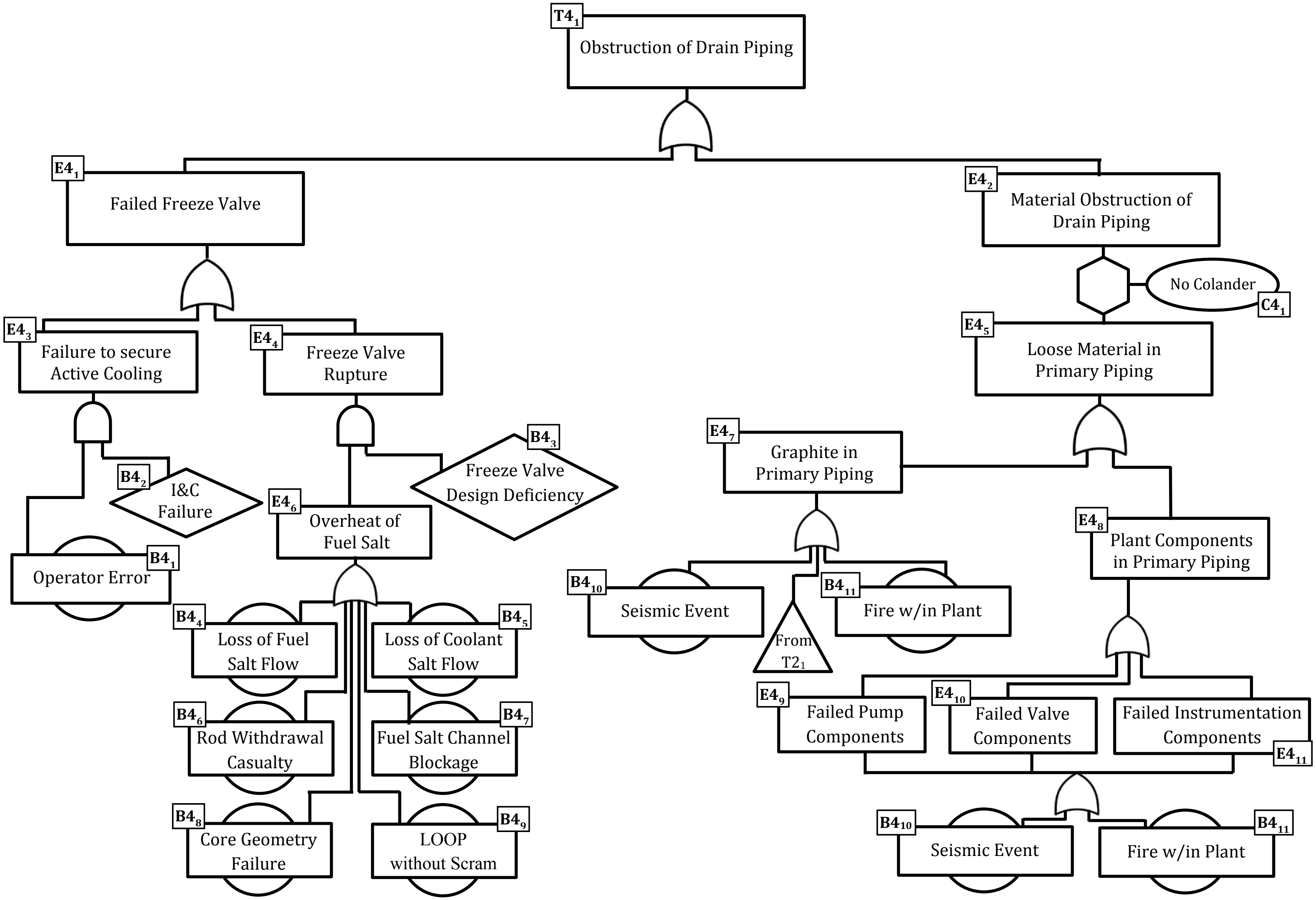

Figure 10. Fault Tree Analysis for Obstruction of the Drain Piping

$T4_1$ – Obstruction of the Drain Piping preventing transfer of the fuel salt from the core
$E4_1$ – Failed freeze valve during a casualty scenario requiring drainage of the fuel salt
$E4_2$ – Material obstruction of the drain piping during a casualty scenario requiring drainage of the fuel salt
$E4_3$ – Failure to secure passive cooling to the freeze valve
$E4_4$ – Freeze valve ruptures due to expansion of the frozen salt plug
$E4_5$ – Loose material in the primary loops
$E4_6$ – Overheat of fuel salt occurs during operations requiring drainage of the fuel salt
$E4_7$ – Graphite moderator is present in the primary loops
$E4_8$ – Failed plant components are present in the primary loops
$E4_9$ – Failed pump components are present in the primary loops (impeller vanes, fasteners etc.)
$E4_{10}$ – Failed valve components are present in the primary loops (valve discs or stems etc.)
$E4_{11}$ – Failed instrumentation components are present in the primary loops
$C4_1$ – No colander is factored into the design to prevent loose plant material from entering the drain piping
$B4_1$ – Operator fails to properly secure active cooling to the freeze valve during casualty scenario
$B4_2$ – Instrumentation and control failure prevents securing active cooling to the freeze valve
$B4_3$ – Freeze valve design demonstrates vulnerability to failure during thermal expansion of the freeze plug
$B4_4$ – Loss of fuel salt flow causing rise in temperatures and fuel salt volume
$B4_5$ – Loss of coolant salt flow causing rise in temperatures and fuel salt volume
$B4_6$ – Rod withdrawal casualty causing rise in temperatures and fuel salt volume
$B4_7$ – Fuel salt channel blockage causing rise in temperatures and fuel salt volume
$B4_8$ – Core geometry failure causing rise in temperatures and fuel salt volume
$B4_9$ – Loss of off-site power without Scram causing rise in temperatures and fuel salt volume
$B4_{10}$ – Seismic Event causing failure of plant components in the reactor vessel and primary loops
$B4_{11}$ – Fire within the plant causing failure of plant components in the reactor vessel and primary loop

Minimal Cut Set

$$\begin{aligned} T4_1 = {} & (B4_1 \bullet B4_2) + (B4_4 \bullet B4_3) + (B4_5 \bullet B4_3) + (B4_6 \bullet B4_3) + (B4_7 \bullet B4_3) + \\ & (B4_8 \bullet B4_3) + (B4_9 \bullet B4_3) + (B4_{10} \bullet C4_1) + (B4_{11} \bullet C4_1) + (B2_1 \bullet C4_1) + \\ & (B2_3 \bullet C4_1) + (B2_4 \bullet C4_1) + (B2_5 \bullet C4_1) + (B2_6 \bullet C4_1) + (B2_2 \bullet C4_1) + \\ & (B1_1 \bullet C1_1 \bullet C4_1) + (B2_3 \bullet C2_1 \bullet C4_1) + (B2_4 \bullet C2_1 \bullet C4_1) + \\ & (B2_5 \bullet C2_1 \bullet C4_1) + (B2_6 \bullet C2_1 \bullet C4_1) + (B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 \bullet C4_1) \end{aligned}$$

Results:
Zero single-component minimal cut sets
Fifteen double-component minimal cut sets
Five triple-component minimal cut sets
Zero quadruple-component minimal cut sets
One quintuple-component minimal cut set

Figure 10 and Boolean reduction indicate that there are no single-component minimal cut sets for an obstruction of the drain piping. Within the double-component minimal cut sets there are two distinct subsets: those attributed to inadequate design and those attributed to material deficiencies. The risk of inadequate design resulting in failure of the freeze plug is driven primarily by the mechanical considerations. As discussed by Beall et al. there must be careful consideration during design and testing of the freeze valve to ensure proper melting of the freeze plug, preventing thermal expansion and rupture of the freeze plug piping (1964, 231). Additionally, the fault tree reveals that a partially thawed freeze plug would constitute an obstruction of the drain piping and can only be mitigated by proper design of the valve itself and the active cooling system keeping the plug frozen.

The remaining double-component minimal cut sets are predicated on material failure and the conditioning event that no perforated colander or straining device is present to preclude clogging of the drain piping. Material failure includes anything from the graphite tubing to failed valve components, check valve or globe valve discs, failed pump components such as vanes or fasteners, or primary plant instrumentation components that are resident in the fuel salt loop. These failures could be attributed to chemical corrosion, seismic events, fires or any other basic initiating event resulting in high temperatures or pressures in the fuel salt loop.

The final triple- and quintuple-component minimal cut sets are very unlikely because of the order of magnitude. They assume material failure of plant materials due to uncontrolled rod withdrawal or breakage of graphite tubes, which will lead to obstruction of the drain piping.

The reader will observe that a common factor for 14 of the 21 minimal cut sets is the conditioning event $C4_1$, the lack of a colander present upstream of the drain piping. Without a colander or strainer device, there is significantly increased risk that an obstruction may be transported directly into the drain piping and prevent proper draining of the fuel salt.

Finally, there are two undeveloped initiating events due to a lack of design detail in the LFTR System Design Descriptions. The first is an instrumentation and control failure that precludes securing the active cooling system. Because the operator must physically secure the active cooling system in all but the most severe overheat casualties to ensure the core drains, any fault that precludes securing cooling would pose great risk to the inherent safety of the LFTR. Secondly, the freeze valve mechanical design must be confirmed and tested to demonstrate repeated success in thawing the freeze plug without compromising the integrity of surrounding piping due to salt expansion. Because these two hazards cannot be further assessed, they remain undeveloped events but are considered basic initiating events because of their potential for damage.

The list of initiating events found to lead to obstruction of the drain piping includes:

1. Operator error (failing to secure active cooling)
2. Instrumentation and control failure prevents securing active cooling to the freeze valve
3. Freeze valve design deficiency (thermal expansion damages freeze valve piping)
4. Loss of fuel salt flow
5. Loss of coolant salt flow
6. Rod withdrawal casualty
7. Fuel salt channel blockage
8. Core geometry failure
9. Loss of off-site power
10. Seismic Event causing failure of plant components in the reactor vessel and primary loops
11. Fire within the plant causing failure of plant components in the reactor vessel and primary loop
12. Breakage of one or more graphite tubes

5.2 EVENT TREE ANALYSIS

Event Tree Analysis is similar to Fault Tree Analysis in that it explores the sequence of events leading from initiating event to system or subsystem failure. However, whereas Fault Tree Analysis uses "top-down" or deductive reasoning, Event Tree Analysis uses "bottom-up" or inductive reasoning. Event trees were constructed for each of the significant initiating events determined in previous sections. In some cases, initiating events that could be categorized into more broad groups were used to illustrate that many event sequences are common to several basic initiating events.

Traditional event tree analysis calls for the assignment of probabilities to each branch in the event tree. However, due to the lack of operating experience and limited design data for lithium fluoride thorium reactors, probabilities were omitted in favor of a qualitative binary analysis. Each branch of the tree was assigned a value of "Success" or "Failure" based on the *possibility* of failure, not probability. As discussed during the engineering evaluation and technical study of the LFTR, many hazard scenarios exist in which the top events are *possible*, and event trees constructed below reinforce the fault tree analysis results indicating which basic initiating events may possibly lead to each top event. Based on the findings of the event tree analysis, recommendations are made to improve the engineering design to properly mitigate against the failure modes identified below.

5.2.1 EVENT TREE ANALYSIS FOR UNCONTROLLED ROD WITHDRAWAL

Initiating events for uncontrolled rod withdrawal casualty were grouped into the following broad categories for Event Tree Analysis:

Basic Initiating Event	Category
$B1_1$ – Operator Error	Human Reliability Analysis
$B1_2$ – Rod Control Interlock failure	Rod Control Circuitry and Protective Logic
$B1_3$ – Instrumentation and Control error	
$B1_4$ – Seismic Event	Catastrophic Mechanical Failure
$B1_5$ – Fire within the Plant	
$B1_6$ – Loss of Instrument Air	Blanket-Gas Control Valve Design
$B1_7$ – Loss of AC or DC Control Power	

Table 7. Uncontrolled Rod Withdrawal Initiating Event Categories

As previously mentioned, human reliability analysis is not performed within this study and event trees were not constructed for operator error as an initiating event. Similarly, rod control circuitry and protective logic remain undeveloped events due to the premature design phase of the LFTR and event trees cannot reasonably be constructed due to a lack of design detail. As such, catastrophic mechanical failure of the liquid control rod blanket-gas valves and poor blanket-gas control valve engineering design were the only initiating event categories analyzed for the LFTR.

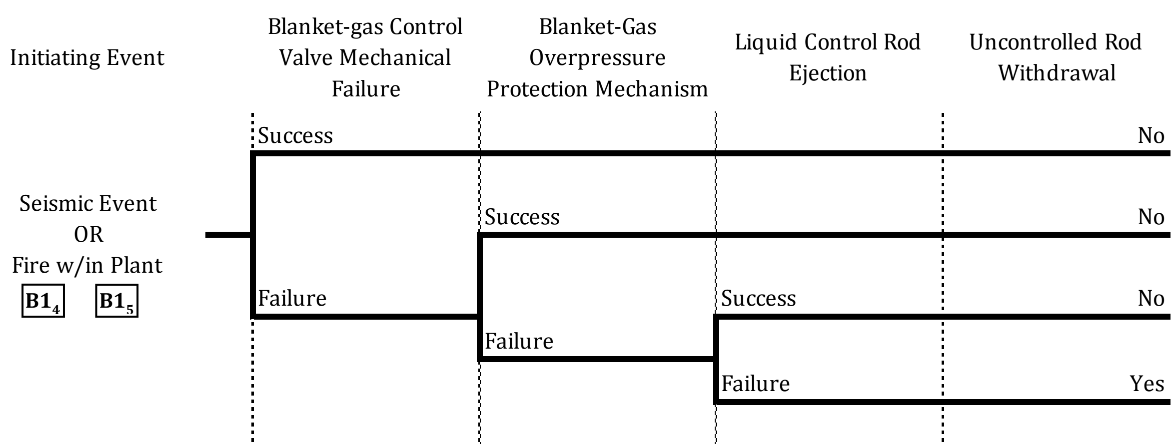

Figure 11. Event Tree for URW – Mechanical Failure of Blanket-gas Control Valve

Seen in Figure 11, catastrophic mechanical failure of the blanket-gas control valve can be shown to lead to uncontrolled rod withdrawal. If the control valve fails, high pressure air will be applied to the top of the blanket salt liquid control rod, causing rapid reduction of the salt column, in effect ejecting the control rod. The only engineered system to combat this casualty would be some form of overpressure protection for the blanket-gas, whose failure also constitutes a step towards uncontrolled rod withdrawal.

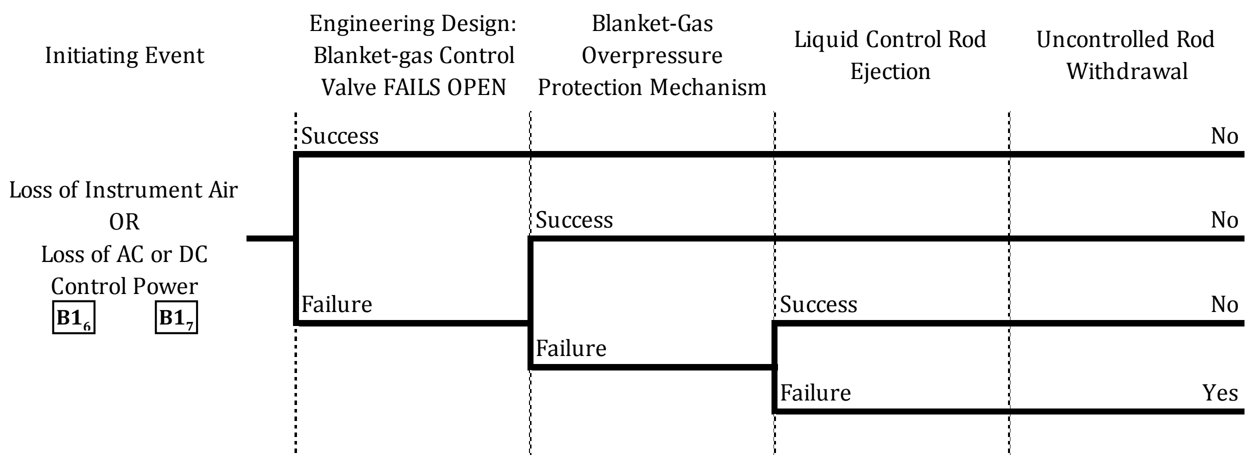

Figure 12. Event Tree for URW – Engineering Design Deficiency in Blanket-gas Control Valve

Figure 12 demonstrates the same event sequence leading to failure as that shown in Figure 11, namely that the application of high pressure air will eject the liquid control rod in the absence of any overpressure protection mechanism. The one major difference is in the mode of failure that leads to applying high pressure air to the blanket-gas. Figure 11 assumes a seismic event or fire that leads to the catastrophic failure of the control valves. In Figure 12, a certain engineering design is assumed in which the control valves fail OPEN, such as in the use of a solenoid-operated valve or air-operated valve that uses electrical current or air to *close* the valve. In this design selection, loss of power or instrument air would lead to the uncontrolled application of high pressure blanket-gas causing rapid expulsion of the liquid control rod. Of course, mitigation of this risk is simple and requires only that the blanket-gas control valves be designed to fail SHUT, preventing the rapid pressurization of the liquid control rod during loss of power or loss of instrument air casualties.

5.2.2 EVENT TREE ANALYSIS FOR BREAKAGE OF ONE OR MORE GRAPHITE TUBES

Initiating events for the breakage of one or more graphite tubes were grouped into the following broad categories for Event Tree Analysis:

Basic Initiating Event	Category
$B2_1$ – Improper Handling of Graphite Tubes during Inspection or Replacement	Improper Graphite Handling
$B2_2$ – Chemical Processing Plant Failure leading to corrosion of Graphite	Chemical Processing Plant Failure
$B2_3$ – Loss of fuel salt flow	Loss of Heat Sink
$B2_4$ – Loss of coolant salt flow	
$B2_5$ – Continuous rod withdrawal casualty	Excess Reactivity
$B2_6$ – Fuel salt channel blockage	
$B2_7$ – Primary to Intermediate Loop Heat Exchanger Failure	Heat Exchanger Failure
$B2_8$ –Intermediate Loop to CO_2 Heat Exchanger Failure	

Table 8. Breakage of one or more Graphite Tubes Initiating Event Categories

All basic initiating events identified in Table 8 were evaluated using event trees with the exception of Improper handling of the graphite tubes during maintenance or inspection. This initiating event is trivial and would lead directly to the breakage of the graphite moderator. Additionally, this initiating event is mitigated by procedural and administrative requirements as well as operator control over the evolution. Because these elements are not clearly defined and do involve human reliability analysis, the improper handling of graphite is recognized as a basic initiating event but is not further analyzed for its corresponding event sequence.

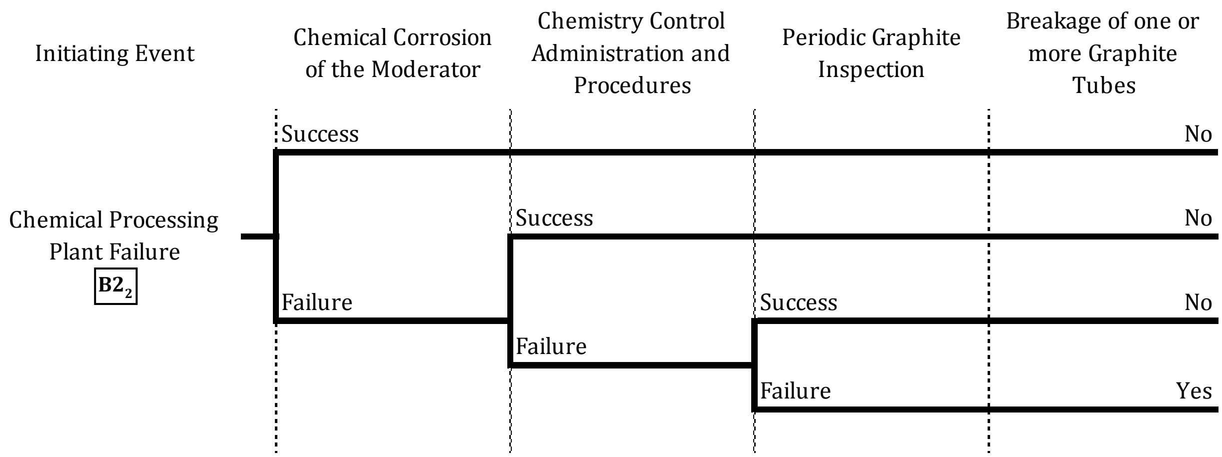

Figure 13. Event Tree for Breakage of Graphite Tubes – Chemical Processing Plant Failure

The potential exists for failure of the graphite tubes due to exposure to chemical corrosion and high-temperature, high-neutron flux conditions within the core (Lane 1958, 623). Figure 13 depicts the event sequence from an initial failure or malfunction in the chemical processing plant, leading to chemical corrosion of the graphite. If chemistry control procedures and periodic graphite inspections fail, the potential exists for breakage of one or more graphite tubes.

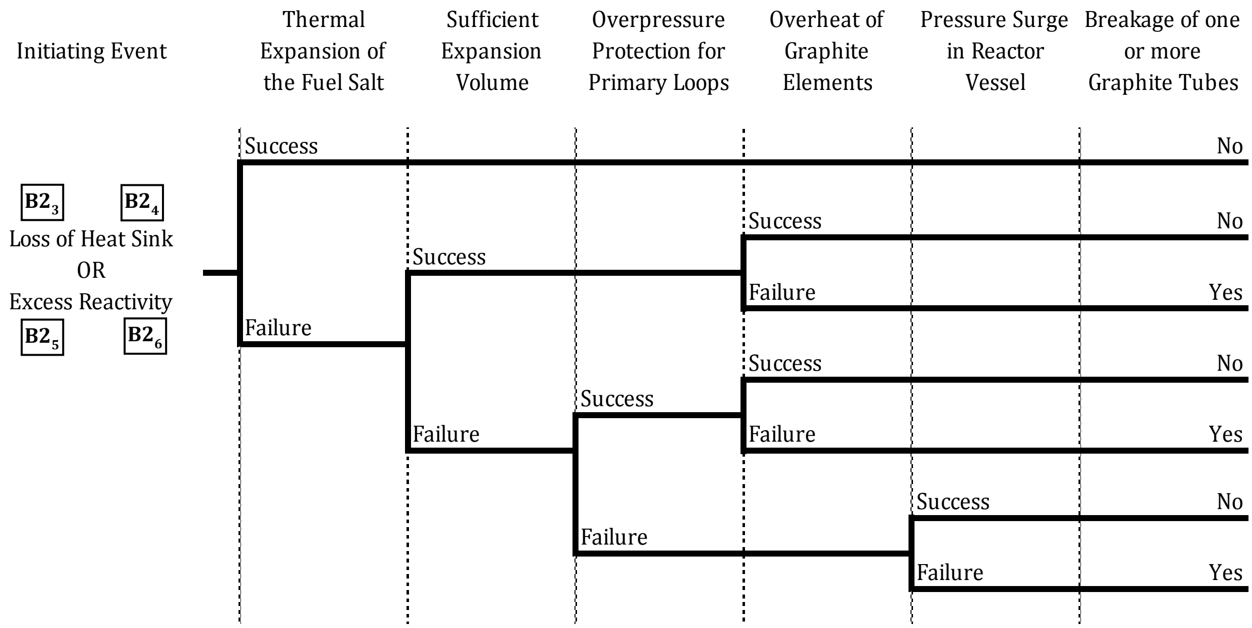

Figure 14. Event Tree for Breakage of Graphite Tubes – Loss of Heat Sink or Excess Reactivity

Basic initiating events $B2_3$ through $B2_6$ were categorized and combined into one event tree shown in Figure 14. The loss of heat sink or excess reactivity casualties represented by the four basic initiating events from Table 8 all immediately lead to thermal expansion of the fuel salt. In a reactor fundamentally designed to operate at low pressures, the thermal expansion of the fuel salt represents a risk for pressure rise and fracture of primary plant components. Several protective mechanisms exist to mitigate this risk, including an expansion volume to accommodate expanding fuel salt and overpressure protection in the form of pressure relief valves. If these mitigations fail and high pressure or temperature conditions permeate the reactor core, breakage of one or more graphite tubes is possible.

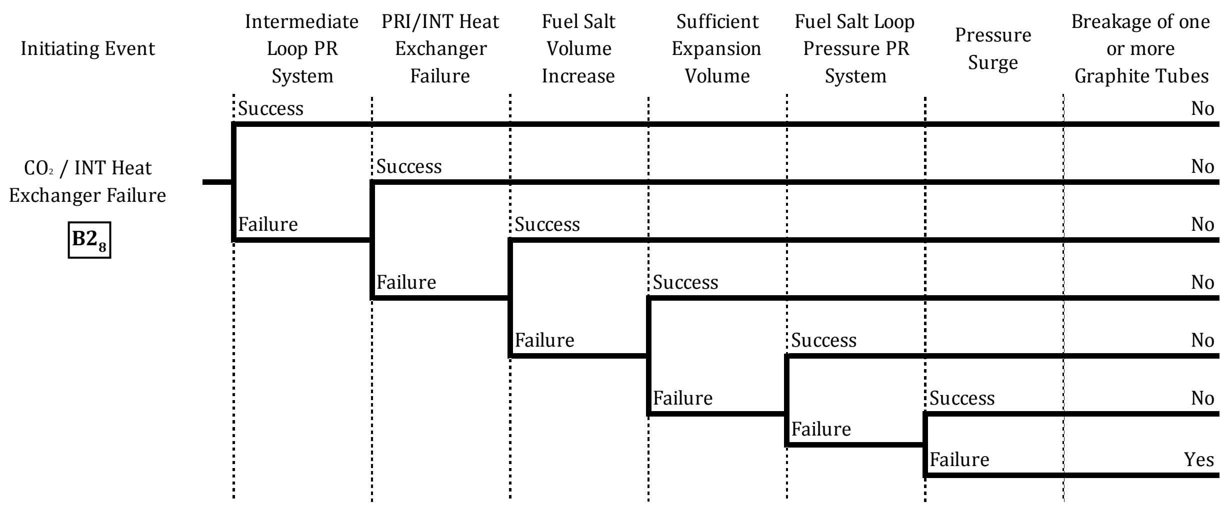

Figure 15. Event Tree for Breakage of Graphite Tubes – Heat Exchanger Failure

Finally, the failure of the CO_2 / Intermediate Coolant Salt Heat Exchanger may also lead to breakage of one or more graphite tubes due to the translation of extremely high pressures of the closed Brayton-cycle power conversion system back to the fuel salt loop. As discussed in the associated fault tree analysis, successful translation of high pressures back to the fuel salt loop requires several coincident failures and is therefore very unlikely. However, as seen in Figure 15, if the CO_2 / Intermediate Heat Exchanger fails and overpressure protection of the coolant salt loop is inoperable, the high pressure could rupture the Primary / Intermediate heat exchanger. In the absence of an expansion volume or primary pressure reliefs, this high pressure would cause a pressure surge and possible breakage of one or more graphite tubes in the reactor core.

5.2.3 EVENT TREE ANALYSIS FOR IMPROPER OR INADEQUATE COOLING OF DRAIN TANKS

Initiating events for the improper or inadequate cooling of the drain tanks were grouped into the following broad categories for Event Tree Analysis:

Basic Initiating Event	Category
$B3_1$ – Design Deficiency – Drain Tank Geometry	Drain Tank Design Deficiency
$B3_4$ – Design Deficiency – Water-cooled Mechanism	
$B3_7$ – Design Deficiency – Air-cooled mechanism	
$B3_2$ – Chemical Processing Plant Failure results in increased fuel concentrations	Chemical Processing Plant Failure
$B3_3$ – Breakage of one or more graphite tubes	Moderator in Drain Tanks
$B3_5$ – Seismic Event	Catastrophic Mechanical Failure
$B3_6$ – Fire within the plant	

Table 9. Improper or inadequate cooling of the drain tanks Initiating Event Categories

Event tree analysis for the improper or inadequate cooling of drain tanks was challenging because of the limited discussion in the Flibe LFTR Technology Assessment on design of the tanks or the cooling circuit. A review of the literature indicates credible hazards exist that are not yet accounted for by the LFTR design. Basic initiating events $B3_1$, $B3_4$ and $B3_7$ are all predicated with "Design deficiency" because, in the absence of more detailed plans, these inadequacies could lead directly to improper cooling of the drain tanks. These design deficiencies are not evaluated in event tree format but are recognized as vulnerabilities, and will be included in the final discussion and recommendations to improve upon the existing LFTR design.

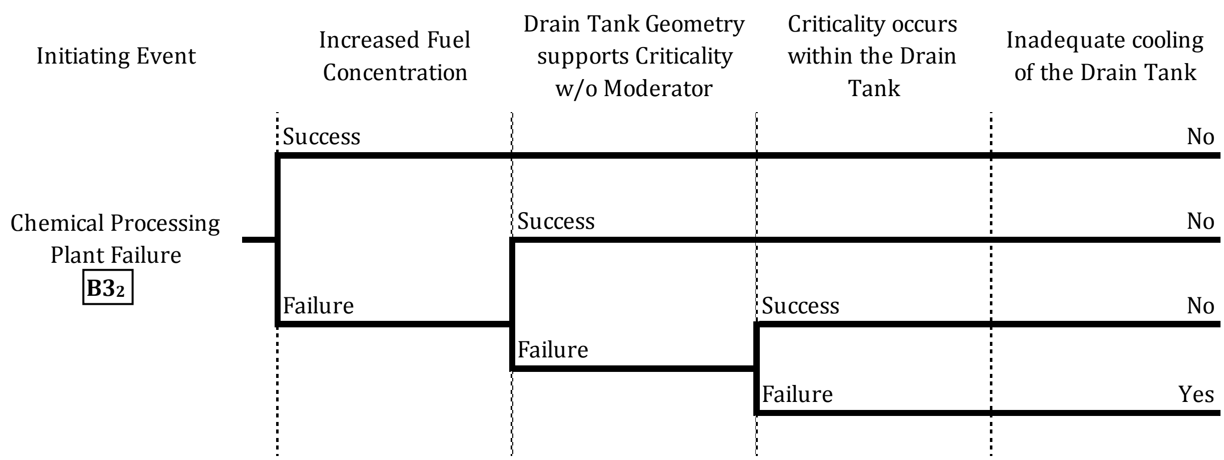

Figure 16. Event Tree for Improper or Inadequate cooling of the Drain Tanks – Chemical Processing Plant Failure

The event sequence in Figure 16 depicts how failure of the chemical processing plant could serve as a basic initiating event for criticality in the drain tanks. Based on the research conducted by Beall et al. for the MSRE, normal fuel concentrations may increase by a factor of three as the fuel salt freezes and contracts while in the drain tanks (1964, 230). This was determined to be sufficient to cause $k_{eff} = 1.0$ and allow criticality in the drain tanks even without a moderator such as graphite or water present. Any further increase in fuel concentration, such as is possible during a chemical processing plant malfunction, would only worsen the potential for criticality to occur within the drain tanks. Other factors, such as drain tank geometry and neutron moderator are appropriately included in the event tree.

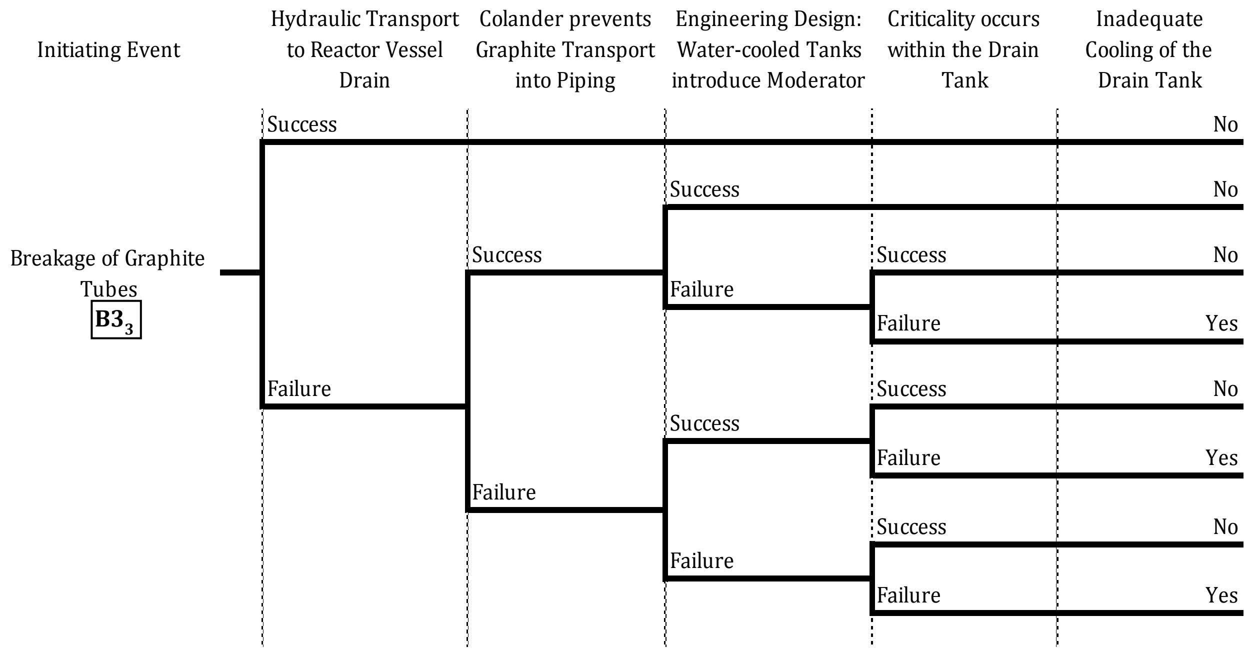

Figure 17. Event Tree for Improper or Inadequate cooling of the Drain Tanks – Breakage of Graphite Tubes

Though previously identified and evaluated as a top event, the breakage of graphite tubes also represents a basic initiating event for improper or inadequate cooling of the drain tanks because it represents the potential for moderator to be hydraulically transported into the drain tanks. In the fault tree analysis, minimal cut set determination further reduced breakage of graphite tubes into its basic initiating events. This process was not repeated to conduct event tree analysis; instead, breakage of graphite tubes was treated as its own initiating event to focus on engineering design recommendations that would preclude the introduction of moderator to the drain tanks. As seen in Figure 17, breakage of graphite could lead to transport to the drain piping. At this stage of the event tree, an opportunity exists to introduce a new design detail to prevent transport of loose material into the drain piping and tanks. In pressurized water reactors, perforated colanders are frequently used to encourage thorough mixing of coolant and prevent transport of debris into piping subsystems (Rhodes and McGregor 2008; U.S. Nuclear Regulatory Commission 2004, 363).

Graphite could be prevented from transporting into the drain tanks by a similar device not yet described in the Flibe Energy LFTR. Finally, a water-cooled circuit would inherently introduce more neutron moderator to the tanks and is not advised.

5.2.4 EVENT TREE ANALYSIS FOR OBSTRUCTION OF THE DRAIN PIPING

Initiating events for obstruction of the drain piping were grouped into the following broad categories for Event Tree Analysis:

Basic Initiating Event	Category
$B4_1$ – Operator error (failure to secure freeze plug active cooling)	Human reliability analysis
$B4_2$ – Instrumentation and Control failure to secure freeze plug active cooling	Reactor Protection Circuitry and Logic
$B4_3$ – Freeze valve design vulnerable to thermal expansion failure	Freeze Valve design deficiency
$B4_4$ – Loss of fuel salt flow	Loss of Heat Sink
$B4_5$ – Loss of coolant salt flow	
$B4_9$ – Loss of off-site power without scram	
$B4_6$ – Continuous rod withdrawal casualty	Excess Reactivity
$B4_7$ – Fuel salt channel blockage	
$B4_8$ – Core geometry failure	
$B4_{10}$ – Seismic Event	Catastrophic Mechanical Failure
$B4_{10}$ – Fire within the plant	

Table 10. Obstruction of Drain Piping Initiating Event Categories

Initiating events involving human reliability analysis and undeveloped engineering design ($B4_1$ through $B4_3$) were omitted from event tree analysis. Consequently, basic initiating events $B4_4$ through $B4_{11}$ are evaluated in Figures 18 and 19 below to determine the event sequence from initiating event to subsystem failure.

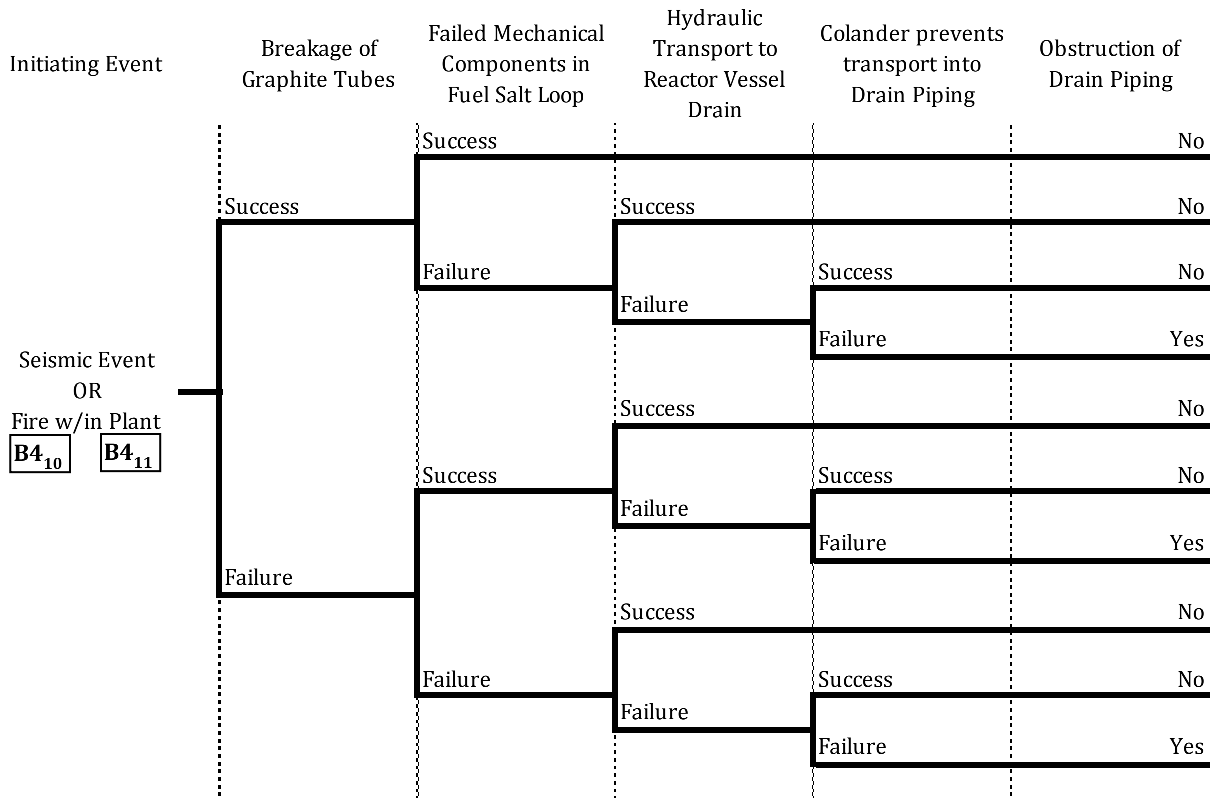

Figure 18. Event Tree for Obstruction of the Drain Piping – Catastrophic Mechanical Failure

The event sequence identified above for seismic events or fires within the plant confirms the associated fault tree analysis, indicating a clear path from catastrophic mechanical failure of graphite or other plant materials to obstruction of drain piping. As identified in the engineering evaluation, obstruction of the piping could have important consequences due to the inability to separate fuel from moderator under casualty scenarios. Important opportunities for prevention of this hazard are material selection and engineering design to mitigate the possibility of mechanical failure in the first place; empirical testing demonstrating the credibility of hydraulic transport; and the inclusion of a perforated colander to prevent any debris from entering undesired piping systems.

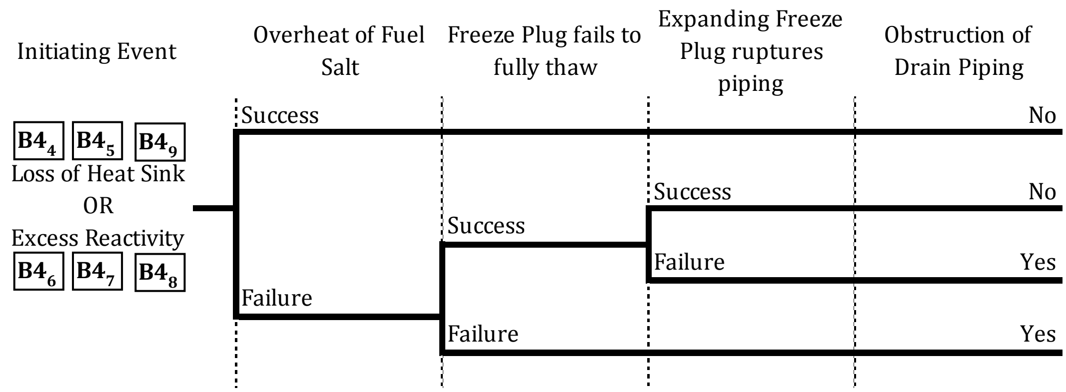

Figure 19. Event Tree for Obstruction of the Drain Piping – Loss of Heat Sink or Excess Reactivity

The loss of heat sink and excess reactivity initiating event categories have similar event sequences in the LFTR because both cause an increase in core temperatures and, therefore, thermal expansion of fuel salt. Under overheat conditions, the freeze plug is designed to melt and open the flowpath to the drain tanks. Failure of the freeze plug to melt quickly and fully allows for continued increase in core temperatures and the possibility of core damage. For this reason, a partially thawed freeze valve causes obstruction of the drain piping. Additionally, freeze valve failure due to thermal expansion and rupture of the valve body also constitutes core damage. Though the molten salt would ultimately flow from the ruptured pipe to the catch pan into drain tanks, the rupture itself represents the release of radioactivity from within the primary plant boundary. Both partial thawing of the freeze plug and freeze valve rupture are preventable hazards through proper engineering design and empirical data collection demonstrating the probability of occurrence of each intermediate event.

CHAPTER 6

LIMITATIONS

It is important to identify the limitations of application of this research before proceeding to the conclusion and recommendations. Because the Flibe Energy LFTR is in its early design stages, system design descriptions lack the level of detail required for component-level failure analysis. Piping schematics, detailed redundant engineered safety mechanisms and instrumentation functionality, among other things, are notably absent from the Flibe Energy LFTR system design descriptions. Observing these limitations on expected plant operation at the component level, this thesis evaluates the principle of LFTR operation. Even in the absence of component-specific data, the fundamental normal plant operations were subject to initiating event analysis to provide coarse recommendations for design improvement. Certainly, on further development of reactor schematics, the Level 1 PSA must again be analyzed to provide the fine-tuning of component-specific failure analysis and the application of probabilities of failure to arrive at true core damage frequencies.

Furthermore, the reader will note that many single-component minimal cut sets resulted from fault tree analysis for two of the four selected top events. Breakage of one or more graphite tubes and inadequate cooling of the drain tanks both contain a combined ten unique single-component minimal cut sets. These cut sets may alarm engineers seeking to understand the safety of the LFTR. Again, it is important to emphasize that these results are based on the system design descriptions available while conducting this research. In the absence of detailed reactor and subsystem schematics, assumptions were made about the basic principles of operation of the LFTR. Some of these single-component minimal cut sets represent true vulnerabilities in the LFTR design, while others simply reflect a lack of adequate design detail to be evaluated further. The distinction between which minimal cut sets are of concern and those easily mitigated by plant design will only become apparent in future phases of development. Nonetheless, given the existing design descriptions, these single-component minimal cut sets offer engineers risk-mitigation opportunities as described in the results and conclusions section of this study.

CHAPTER 7

CONCLUSION AND RECOMMENDATIONS

In the "Technology Assessment of a Molten Salt Reactor Design" (2015) there were 36 hazards identified that posed great risk to safety and integrity of the LFTR. Engineering evaluation and technical study revealed four of these hazards as potential causes of reactor core damage. These primary hazards were selected for further initiating event analysis:

- Unintentional control rod withdrawal
- Breakage of one or more graphite tubes
- Improper or inadequate cooling of the drained fuel salt
- Partially thawed piece of salt plug or solid mass obstructs piping to the drain tank.

A thorough review of existing initiating event lists maintained by the Nuclear Regulatory Commission and the EPRI as well as consideration of existing PSAs for Generation IV Reactor Designs resulted in a list of potential initiating events whose feasibility was further evaluated using logical determination. Fault tree analysis and event tree analysis results were consolidated to reveal one key list of basic initiating events capable of leading to one or more of the four primary hazards identified. The majority of these initiating events are shared in common with pressurized water reactors:

1. Operator Error
2. Rod control interlock failure
3. Instrumentation and control circuitry or protective logic failure
4. Seismic events
5. Fire within the plant
6. Loss of instrument air
7. Loss of AC or DC control power
8. Continuous rod withdrawal casualty

9. Core geometry failure
10. Loss of off-site power without Scram

Initiating events determined to be unique to the LFTR design are

11. Improper handling of graphite tubes during maintenance or inspection
12. Chemical processing plant failure
13. Loss of fuel salt flow
14. Loss of coolant salt flow
15. Fuel salt channel blockage
16. Drain tank cooling mechanism design deficiency
17. Freeze valve design deficiency
18. CO_2 / Intermediate Coolant Salt Heat Exchanger failure

Fault tree analysis and event tree analysis also revealed important opportunities for risk mitigation to preclude the occurrence of a top level hazard by interrupting the event sequence with engineered safety features. The safety features identified are

- Redundant blanket-gas control valves for a liquid control rod system as well as overpressure protection to mitigate the risk of liquid control rod ejection
- Fuel salt expansion volume to accommodate the thermal expansion and contraction of fuel salt during all modes of reactor operation
- Fuel salt pressure relief system to mitigate pressure surges caused by thermal expansion or heat exchanger failure
- Air-driven cooling circuit with demonstrated success for use in the drain tanks. Avoid the use of water-cooled systems to prevent inadvertent introduction of a neutron moderator
- Perforated colander upstream of drain piping to prevent graphite or other debris from hydraulic transport into the drain piping or drain tanks

- Squirrel-cage induction motors for use as fuel salt and coolant salt pumps instead of synchronous motors, obviating the need to design a sealed electric drive motor housing outside of the primary containment (Appendix A contains further discussion).

Though probabilities of occurrence were unavailable for this study, future research may find the groundwork provided by fault tree analysis and event tree analysis crucial in quantifying risks for the LFTR. The basic initiating events identified and the vulnerabilities detected during analysis offer planners the opportunity to review system design descriptions and properly mitigate some of the most important hazards for the Flibe Energy LFTR.

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APPENDIX A

TECHNICAL REVIEW OF "WHAT-IF ANALYSIS TABLES" (EPRI 2015, A-1 to A-39)

Hazard Scenario	Justification for Inclusion/Exclusion from Study	References
Unintentional control rod withdrawal	**Included:** • **Mode:** Failure of liquid control rod with overpressurization of control gas • **Effect:** Prompt criticality (see SL-1 casualty) causing vaporization of core materials • **Effect:** Rapid thermal expansion of fuel salt with no expansion volume causing rupture of reactor vessel	- (Thatcher n.d., 11) - (Electric Power Research Institute 2015, 3-9) - (Carruth 1989) - (Boyd 2008, 463) - (International Atomic Energy Agency 2010, 16)
Loss of blanket salt	Excluded: • MSRE assessed no hazard due to higher pressure of blanket salt compared to fuel salt • Neutronic computations needed to support the theory of increase in reactivity due to a loss of blanket salt	- (Electric Power Research Institute 2015, A-3) - (Kasten 1967, 18)
Premature criticality during filling	Excluded: • Precluded by procedural compliance and operator supervision	- (Electric Power Research Institute 2015, A-4)
Inflow of contaminants or unexpected isotopic ratio in the fuel salt	Excluded: • Operational/administrative procedures can be used to implement changes to the rate of fuel addition • Safety systems involving fast control of reactivity (boron tipped control rods) provide gross reactivity control	(Electric Power Research Institute 2015, A-7)
Breakage of one or more graphite tubes	**Included:** • **Mode:** Neutron irradiation and embrittlement of graphite structural components • **Mode:** Pressure or temperature surge resulting in mechanical failure • **Mode:** Chemical corrosion of graphite due to fluoride fuel salt interaction • **Effect**: Increased reactivity due to increase in fuel salt inventory in the reactor core region • **Effect:** Blockage of fuel salt channels and higher neutron flux due to off-design clearances in fuel salt channels • **Effect:** Possible transport of graphite contaminant to the drain line or fuel salt drain tank, jeopardizing operability of the	- (Electric Power Research Institute 2015, A-12) - (Kasten 1967, 18) - (Beall, et al. 1964, 219-221) -(Salama and El-Morshedy 2011)

	freeze plug and drain tank safety systems	
Inadvertent release of fission product gas from reactor cell or containment	Excluded: • Several layers of designed defense by Flibe LFTR: - Radiation sensors in containment cell cover gas - Operate off gas handling system at lower pressure than containment cell - Double walled pipe system with radiation sensors	- (Electric Power Research Institute 2015, A-13)
Hydrogen reacts with fluorine in the chemical processing system	Excluded: • Procedural and administrative controls would preclude mixing hydrogen and fluorine. • Industry practices are being reviewed for hydrogen management and storage. • Careful separation of processes is being designed for the Flibe Energy LFTR.	- (Electric Power Research Institute 2015, A-23) - (Kasten 1967, 12)
Excess pressure in the helium bubbler	Excluded: • Overpressurization could occur to a closed discharge valve. Recommend mitigation by inclusion of multiple redundant reliefs and periodic testing requirements. • More likely result would be inability to remove fission products from the reactor, requiring plant shutdown. No risk of catastrophic failure.	- (Electric Power Research Institute 2015, A-25) - (Beall, et al. 1964, 62)
Minor failure in the primary heat exchanger	Excluded: • Because the coolant salt is kept at higher pressures (~10-15 bar) than the fuel salt loop (~1-2 bar), a failure of the primary heat exchanger would introduce coolant salt to the fuel loop, reducing reactivity by displacing fuel. This casualty does not represent a hazard for a power excursion or core damage.	- (Electric Power Research Institute 2015, A-26)
Major failure in the primary heat exchanger	Excluded: • The result of a major primary heat exchanger failure is a sudden increase of non-fissile material into the fuel salt loop, causing a down-power excursion and rapid lowering of median temperature. This would require plant shutdown and draining of the contaminated fuel salt, but does not present a hazard to core integrity.	- (Electric Power Research Institute 2015, A-26)
Sealed housing for the electric drive motors for pumps fail	Excluded: • Flibe Energy LFTR proposes pump design from the ORNL MSBR (Robertson, Smith, et al. 1968, 44). These pumps are vertical-shaft	- (Electric Power Research Institute 2015, A-29) - (Robertson,

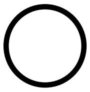
	sump-type single-stage centrifugal pumps driven by electric motors that require access for maintenance. This technology preceded the development of squirrel cage induction motors, which have since become industry standard. • Squirrel cage induction motors currently comprise 97% of motors used in PWRs and 94% of motors used in BWRs. They are resistant to degradation in the harsh environment within the containment structures of nuclear power plants and are proven to have a service life of ~40 years. Recommend using these induction motors to obviate the need to create sealed electric drive motor housings outside the primary containment.	Smith, et al. 1968, 44) -(Villaran and Subudhi 1996, 2-1 and 2-13)
Inadequate removal of Pa or U from the blanket salt	Excluded: • Insufficient batch filling and shutdown of processing and reactor system will occur. • No hazard for core damage is discussed in the LFTR design or found in a review of the literature.	- (Electric Power Research Institute 2015, 4-13)
Electrolytic cell is improperly operated	Excluded: • Improper loading of metallic lithium and thorium into the metallic bismuth stream could result in inadequate contact with the blanket salt, but will not change reactivity significantly. Low loading will cause the electrolytic reaction to cease. • Only potential hazard would be bismuth entering the reactor core where degradation of Hastelloy-N material may result. This is precluded by mechanical design separating the fluids and procedural/administrative processes to prevent introduction of bismuth.	- (Electric Power Research Institute 2015, 4-14 and A-31)
Potassium hydroxide is released	Excluded: • Potassium hydroxide is an industrial safety concern and may result in chemical exposure to workers if released • No radiological hazard exists	- (Electric Power Research Institute 2015, A-34)
Improper or inadequate cooling of the drained fuel salt	**Included:** • **Mode:** Improper selection of passive cooling mechanism (i.e. water-cooled), which may encourage criticality in the drain tank • **Mode:** Inadequate design resulting in criticality or excessive temperatures, causing failure of the drain tank to the	- (Electric Power Research Institute 2015, 3-22) -(Robertson 1965, 220) - (Beall, et al. 1964, 230)

	environment • **Effect:** The presence of a moderator in the drain tank (water or graphite) could enable criticality leading to temperature rise and structural failure.	-(Haubenreich, et al. 1968, 68-69)
Partially thawed piece of salt plug or solid mass obstructs piping to drain tank	**Included:** • **Mode:** Partial thaw of the freeze valve, rupture of freeze valve piping due to thermal expansion, or graphite obstruction of drain tank piping • **Effect:** Inability to separate the fuel contents from the graphite moderator in the core region, resulting in sustained fission during casualty operations • **Effect:** Thermal expansion of fuel salt causing rapid pressure increase in the reactor vessel • **Effect:** Potential rupture of drain piping due to thermal expansion of freeze plug material • **Effect:** Potential fissions outside the core if graphite obstructs the drain piping or enters the drain tanks	-(Robertson, Smith, et al. 1968, 47) - (Electric Power Research Institute 2015, A-10 and A-36) -(Brovchenko, et al. 2013, 338) - (Beall, et al. 1964, 231)

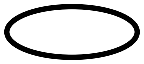
APPENDIX B

FAULT TREE ANALYSIS KEY

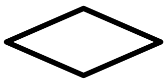
PRIMARY EVENT SYMBOLS



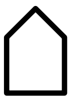
- Basic Event: A basic initiating fault requiring no further development.



- Conditioning Event: Specific conditions or restrictions that apply to any logic gate (used primarily with PRIORITY AND and INHIBIT gates).



- Undeveloped Event: An event that is not further developed either because it is of insufficient consequence or because information is unavailable.

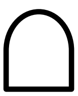


- External Event: An event that is normally expected to occur

INTERMEDIATE EVENT SYMBOLS

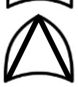
- Intermediate Event: A fault event that occurs because of one or more antecedent causes acting through logic gates.

GATE SYMBOLS

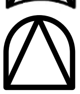


- AND-Gate: Output fault occurs if all of the input faults occur.

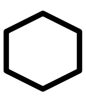
- OR-Gate: Output fault occurs if at least one of the input faults occur.



- EXCLUSIVE OR-Gate: Output fault occurs if exactly one of the input faults occurs.

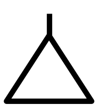


- PRIORITY AND-Gate: Output fault occurs if all of the input faults occur in a specific sequence (the sequence is represented by a conditioning event drawn to the right of the gate).

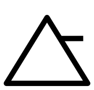


- INHIBIT-Gate: Output fault occurs if the (single) input fault occurs in the presence of an enabling condition (the enabling condition is represented by a conditioning event drawn to the right of the gate).

TRANSFER SYMBOLS



- Transfer in: Indicates that the tree is developed further at the occurrence of the corresponding Transfer Out (e.g. on another page)



- Transfer out: Indicates that this portion of the tree must be attached at the corresponding Transfer In.

(Vesely, et al. 1981, IV-3)

APPENDIX C

DERIVATION OF FAULT TREE ANALYSIS MINIMAL CUT SETS

Minimum cut sets for each fault tree analysis diagram were developed using the "Bottom-up procedure" described in the *Fault Tree Handbook* (Vesely, et al. 1981, XI-4) for Boolean Equations. This procedure takes into account each intermediate event from the lowest leading up to the top event and translates the Boolean logic gates from the Fault Tree Analysis into their mathematical equivalent. By substituting basic events and conditioning events for each intermediate event, the minimum cut sets can be determined and qualitatively evaluated to learn more about the failures leading to each top event.

Uncontrolled Rod Withdrawal Minimum Cut Set Development

$E1_8 = B1_1 + B1_2$
$E1_7 = B1_4 + B1_5 + B1_6 \bullet C1_3 + B1_7 \bullet C1_3$
$E1_6 = B1_3$
$E1_5 = E1_7 + E1_8$
$= B1_4 + B1_5 + B1_6 \bullet C1_3 + B1_7 \bullet C1_3 + B1_1 + B1_2$
$E1_4 = B1_3 + C1_1$
$E1_3 = B1_1 \bullet C1_1$
$E1_2 = E1_5 \bullet C1_2$
$= (B1_1 + B1_2 + B1_4 + B1_5 + B1_6 \bullet C1_3 + B1_7 \bullet C1_3) \bullet C1_2$
$= B1_1 \bullet C1_2 + B1_2 \bullet C1_2 + B1_4 \bullet C1_2 + B1_5 \bullet C1_2 + B1_5 \bullet C1_2 \bullet C1_3 + B1_7 \bullet C1_2 \bullet C1_3$
$E1_1 = B1_1 \bullet B1_2 + B1_3 \bullet C1_1$
$T1_1 = E1_1 + E1_2$
$= B1_1 \bullet B1_2 + B1_3 \bullet C1_1 + B1_1 \bullet C1_2 + B1_2 \bullet C1_2 + B1_4 \bullet C1_2 + B1_5 \bullet C1_2 + B1_5 \bullet C1_2 \bullet C1_3 + B1_7 \bullet C1_2 \bullet C1_3$

Therefore:
$T1_1 = (B1_1 \bullet B1_2)+(B1_3 \bullet C1_1)+(B1_4 \bullet C1_2)+(B1_5 \bullet C1_2)+(B1_2 \bullet C1_2)+(B1_1 \bullet C1_2)+$
$(B1_6 \bullet C1_2 \bullet C1_3)+(B1_7 \bullet C1_2 \bullet C1_3)$

Breakage of One or More Graphite Tubes Minimum Cut Set Development

$E2_{12} = E1_3 = B1_1 \bullet C1_1$
$E2_{11} = B2_7 \bullet B2_8 \bullet C2_2$
$E2_{10} = E2_{12}$
$E2_9 = B2_3 + B2_4 + B2_5 + B2_6 + E2_{11}$
$= B2_3 + B2_4 + B2_5 + B2_6 + B2_7 \bullet B2_8 \bullet C2_2$
$E2_8 = B2_3 + B2_4 + B2_5 + B2_6$
$E2_7 = C2_1 \bullet E2_9$
$= C2_1 \bullet (B2_3 + B2_4 + B2_5 + B2_6 + B2_7 \bullet B2_8 \bullet C2_2)$

$= B2_3 \bullet C2_1 + B2_4 \bullet C2_1 + B2_5 \bullet C2_1 + B2_6 \bullet C2_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2$

$E2_6 = B2_2$

$E2_5 = B2_1$

$E2_4 = B2_1$

$E2_3 = E2_6 + E2_7 + E2_8$

$= B2_2 + B2_3 \bullet C2_1 + B2_4 \bullet C2_1 + B2_5 \bullet C2_1 + B2_6 \bullet C2_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 + B2_3 + B2_4 + B2_5 + B2_6$

$E2_2 = E2_{10} = E2_{12} = E1_3 = B1_1 \bullet C1_1$

$E2_1 = E2_4 + E2_5 = B2_1 + B2_1 = B2_1$

$T2_1 = E2_1 + E2_2 + E2_3$

$= B2_1 + B1_1 \bullet C1_1 + B2_2 + B2_3 \bullet C2_1 + B2_4 \bullet C2_1 + B2_5 \bullet C2_1 + B2_6 \bullet C2_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 + B2_3 + B2_4 + B2_5 + B2_6$

Therefore:

$T2_1 = B2_1 + B2_2 + B2_3 + B2_4 + B2_5 + B2_6 + (B1_1 \bullet C1_1) + (B2_3 \bullet C2_1) + (B2_4 \bullet C2_1) + (B2_5 \bullet C2_1) + (B2_6 \bullet C2_1) + (B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2)$

Inadequate or Improper Cooling of the Drain Tanks Minimum Cut Set Development

$E3_7 = B3_4$

$E3_6 = B3_3 = T2_1$

$E3_5 = B3_5 + B3_6$

$E3_4 = E3_6 + E3_7 = B3_4 + T2_1$

$= B3_4 + B2_1 + B1_1 \bullet C1_1 + B2_2 + B2_3 \bullet C2_1 + B2_4 \bullet C2_1 + B2_5 \bullet C2_1 + B2_6 \bullet C2_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 + B2_3 + B2_4 + B2_5 + B2_6$

$E3_3 = B3_1 \bullet B3_2$

$E3_2 = E3_5 + B3_7$

$= B3_5 + B3_6 + B3_7$

$E3_1 = E3_3 + E3_4$

$= B3_1 \bullet B3_2 + B3_4 + B2_1 + B1_1 \bullet C1_1 + B2_2 + B2_3 \bullet C2_1 + B2_4 \bullet C2_1 + B2_5 \bullet C2_1 + B2_6 \bullet C2_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 + B2_3 + B2_4 + B2_5 + B2_6$

$T3_1 = E3_1 + E3_2$

$= B3_1 \bullet B3_2 + B3_4 + B2_1 + B1_1 \bullet C1_1 + B2_2 + B2_3 \bullet C2_1 + B2_4 \bullet C2_1 + B2_5 \bullet C2_1 + B2_6 \bullet C2_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 + B2_3 + B2_4 + B2_5 + B2_6 + B3_5 + B3_6 + B3_7$

Therefore:

$T3_1 = B2_1 + B2_2 + B2_3 + B2_4 + B2_5 + B2_6 + B3_4 + B3_5 + B3_6 + B3_7 + (B1_1 \bullet C1_1) + (B3_1 \bullet B3_2) + (B2_3 \bullet C2_1) + (B2_4 \bullet C2_1) + (B2_5 \bullet C2_1) + (B2_6 \bullet C2_1) + (B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2)$

Obstruction of the Drain Piping Minimum Cut Set Development

$E4_{11} = B4_{10} + B4_{11}$

$E4_{10} = B4_{10} + B4_{11}$

$E4_9 = B4_{10} + B4_{11}$

$E4_8 = E4_9 + E4_{10} + E4_{11}$

$= B4_{10} + B4_{11} + B4_{10} + B4_{11} + B4_{10} + B4_{11} = B4_{10} + B4_{11}$

$E4_7 = B4_{10} + B4_{11} + T2_1$

$= B4_{10} + B4_{11} + B2_1 + B1_1 \bullet C1_1 + B2_2 + B2_3 \bullet C2_1 + B2_4 \bullet C2_1 + B2_5 \bullet C2_1 + B2_6 \bullet C2_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 + B2_3 + B2_4 + B2_5 + B2_6$

$E4_6 = B4_4 + B45 + B4_6 + B4_7 + B4_8 + B4_9$

$E4_5$ $= E4_7 + E4_8 = B4_{10} + B4_{11} + B4_{10} + B4_{11} + T2_1$
$= B4_{10} + B4_{11} + B2_1 + B1_1 \bullet C1_1 + B2_2 + B2_3 \bullet C2_1 + B2_4 \bullet C2_1 + B2_5 \bullet C2_1 + B2_6 \bullet C2_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 + B2_3 + B2_4 + B2_5 + B2_6$

$E4_4$ $= E4_6 \bullet B4_3$
$= (B4_4 + B45 + B4_6 + B4_7 + B4_8 + B4_9) \bullet B4_3$
$= B4_4 \bullet B4_3 + B4_5 \bullet B4_3 + B4_6 \bullet B4_3 + B4_7 \bullet B4_3 + B4_8 \bullet B4_3 + B4_9 \bullet B4_3$

$E4_3$ $= B4_1 \bullet B4_2$

$E4_2$ $= C4_1 \bullet E4_5$
$= C4_1 \bullet (B4_{10} + B4_{11} + B2_1 + B1_1 \bullet C1_1 + B2_2 + B2_3 \bullet C2_1 + B2_4 \bullet C2_1 + B2_5 \bullet C2_1 + B2_6 \bullet C2_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 + B2_3 + B2_4 + B2_5 + B2_6)$
$= B4_{10} \bullet C4_1 + B4_{11} \bullet C4_1 + B2_1 \bullet C4_1 + B1_1 \bullet C1_1 \bullet C4_1 + B2_2 \bullet C4_1 + B2_3 \bullet C2_1 \bullet C4_1 + B2_4 \bullet C2_1 \bullet C4_1 + B2_5 \bullet C2_1 \bullet C4_1 + B2_6 \bullet C2_1 \bullet C4_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 \bullet C4_1 + B2_3 \bullet C4_1 + B2_4 \bullet C4_1 + B2_5 \bullet C4_1 + B2_6 \bullet C4_1$

$E4_1$ $= E4_3 + E4_4$
$= B4_1 \bullet B4_2 + B4_4 \bullet B4_3 + B4_5 \bullet B4_3 + B4_6 \bullet B4_3 + B4_7 \bullet B4_3 + B4_8 \bullet B4_3 + B4_9 \bullet B4_3$

$T4_1$ $= E4_1 + E4_2$
$= B4_1 \bullet B4_2 + B4_4 \bullet B4_3 + B4_5 \bullet B4_3 + B4_6 \bullet B4_3 + B4_7 \bullet B4_3 + B4_8 \bullet B4_3 + B4_9 \bullet B4_3 + B4_{10} \bullet C4_1 + B4_{11} \bullet C4_1 + B2_1 \bullet C4_1 + B1_1 \bullet C1_1 \bullet C4_1 + B2_2 \bullet C4_1 + B2_3 \bullet C2_1 \bullet C4_1 + B2_4 \bullet C2_1 \bullet C4_1 + B2_5 \bullet C2_1 \bullet C4_1 + B2_6 \bullet C2_1 \bullet C4_1 + B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 \bullet C4_1 + B2_3 \bullet C4_1 + B2_4 \bullet C4_1 + B2_5 \bullet C4_1 + B2_6 \bullet C4_1$

Therefore:

$T4_1 = (B4_1 \bullet B4_2) + (B4_4 \bullet B4_3) + (B4_5 \bullet B4_3) + (B4_6 \bullet B4_3) + (B4_7 \bullet B4_3) + (B4_8 \bullet B4_3) + (B4_9 \bullet B4_3) + (B4_{10} \bullet C4_1) + (B4_{11} \bullet C4_1) + (B2_1 \bullet C4_1) + (B2_3 \bullet C4_1) + (B2_4 \bullet C4_1) + (B2_5 \bullet C4_1) + (B2_6 \bullet C4_1) + (B2_2 \bullet C4_1) + (B1_1 \bullet C1_1 \bullet C4_1) + (B2_3 \bullet C2_1 \bullet C4_1) + (B2_4 \bullet C2_1 \bullet C4_1) + (B2_5 \bullet C2_1 \bullet C4_1) + (B2_6 \bullet C2_1 \bullet C4_1) + (B2_7 \bullet B2_8 \bullet C2_1 \bullet C2_2 \bullet C4_1)$

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