Applying Risk Informed Methodologies To Improve the Economics of Sodium-Cooled Fast Reactors

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Submitted to the Department of Nuclear Science and Engineering in partial fulfillment of the requirements for the degree of Master of Science in Nuclear Science and Engineering

Abstract

In order to support the increasing demand for clean sustainable electricity production and for nuclear waste management, the Sodium-Cooled Fast Reactor (SFR) is being developed. The main drawback has been its high capital and operating costs in comparison with traditional light water reactors. In order to compete, the SFR must be shown to be economically competitive. This study makes use of the proposed Technology Neutral Framework (TNF) being developed by the U.S. NRC. By applying this risk-based approach to safety, rather than the traditional approach of applying deterministic requirements, it will be shown that significant savings can be realized without compromising fundamental safety.

A methodology was developed using the Technology Neutral Framework to judge design alternatives based on risk significance that provide acceptable safety within the framework at less cost. The key probabilistic metrics of Risk Achievement Worth and Limit Exceedence Factor will be used to assess whether a system or component plays an important safety function. If not the system, structure or component either can be eliminated, modified or its safety grade can be reduced resulting in cost savings. In addition, assessments were made to determine how to improve thermal efficiency by raising reactor exit temperature and by applying other design alternatives to reduce costs as evaluated on a safety, reliability and economic basis. This methodology was applied in a series of case studies demonstrating the value of the approach in design. The probabilistic risk assessment, the reference economic model and the Technology Neutral Framework tools required for this methodology are described.

A reference economic model for a pool-type SFR was developed using the G4-ECONS model since it is an acceptable standard model for economic analysis. Since cost predictions for sodium cooled fast reactors are highly uncertain, the results of the economic analysis are used to estimate the ***relative*** improvement in cost as a function of the design alternatives proposed by the TNF methodology approach. This study used generic and comparative numbers for the ALMR and S-PRISM reactors for cost of components of the SFR, to identify capital cost drivers for further study and cost reduction. For comparative purposes, the light water reactor (LWR) economic model in the G4-ECONS model was used and benchmarked to current LWR data.

As a result of the case studies in which the methodology was applied, it was shown that the capital cost of the SFR could be reduced by almost 18% ($336 million) over the reference design and the levelized generating costs could be reduced by over 10% (almost 1 cent/kw-hr). These savings come largely from improvements in thermal efficiency, elimination of the energetic core disruptive accident as a design basis event and simplification of the reactor shutdown system based on risk analysis and safety significance. Should this methodology be applied to the entire plant design, it is expected that significant additional savings could be identified.

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Table of Contents

Chapter 1 - Introduction.......... 9
1.1 Thesis Objectives.......... 9
1.2 Sodium Fast Reactor Background.......... 10
1.2.1 The Sodium Fast Reactor.......... 13
1.2.2 History of SFR Development.......... 14
1.2.3 Economic Issues.......... 16
1.3 Thesis Outline.......... 18
Chapter 2 – Methodology and Framework.......... 19
2.1 Developing the Framework and Methodology.......... 19
2.1.1 Framework and Methodology Description.......... 19
2.1.2 Required Tools for this Framework.......... 21
2.1.3 Technology Neutral Framework Background.......... 22
2.2 The Probabilistic Risk Assessment Model.......... 22
2.2.1 Using the Technology Neutral Framework.......... 23
2.2.2 Probabilistic Risk Assessment.......... 24
2.2.3 Determine Safety Requirements Using the TNF.......... 25
2.2.4 Using PRA Analysis.......... 28
2.3 Using Importance Measures to Analyze Safety within the TNF.......... 32
2.3.1 Identifying with PRA and Importance Measures.......... 34
2.4 Identifying Potential Design Alternatives.......... 35
2.4.1 Identifying with the Economic Data.......... 35
2.5 Development of the Economic Model.......... 37
2.6 Methodology Example.......... 37
Chapter 3 – Development of the Economic Model.......... 39
3.1 Overview.......... 39
3.2 Gathered Data.......... 40
3.2.1 Capital Cost.......... 40
3.2.2 Operations and Maintenance Costs.......... 41
3.2.3 Fuel Cycle Costs.......... 42
3.3 Development of a Reference Model.......... 43
3.3.1 Using the G4ECONS model.......... 43
3.3.2 Levelized Unit Electricity Cost.......... 45
3.3.3 Description of the ALMR.......... 46
3.4 Results from Reference Model.......... 48
3.4.1 LUEC using the ALMR estimated costs.......... 48
3.4.2 Capital cost of Sodium Fast Reactors compared to modern LWR.......... 49
3.4.3 Capital cost of ALMR compared to modern LWR.......... 50
3.4.4 Adjusted Total Capital Cost and LUEC.......... 56
3.5 Limitations of a Reference Model.......... 57
3.6 Uses of the Reference Model to Make More Economical.......... 60
3.6.1 Possible ways to reduce LUEC.......... 60
3.7 Summary.......... 61
Chapter 4 – Demonstration of Methodology through Case Studies.......... 62
4.1 Case Studies to Demonstrate Methodology.......... 62
4.2 Case Study 1 – Removing Energetic Core Disruptive Accidents from Licensing Basis Events.......... 62
4.2.1 Possible Dose from ECDA.......... 63
4.2.2 Establishing the Frequency of ECDA.......... 65
4.2.3 Economic Benefit of Not Designing for ECDA.......... 66

4.2.4 Cost of Traditional Containment 67
4.2.5 Safety Issues with Containment 72
4.2.6 Potential Savings from other Design Alternatives 72
4.2.7 Increasing Core Outlet Temperature 73
4.2.8 Summary 73
4.3 Case Study 2 – Possible Design Alternatives that may Improve Efficiency 74
4.3.1 Design Alternative – Increase Core Outlet Temperature 75
4.3.3 Design Alternative – Change Power Conversion System 78
4.3.4 Summary 81
4.4 Case Study 3 - Manufacturing the Steam Generator as Non-Safety Grade 81
4.4.1 Risk-Informed Special Treatment Requirements for Reactors 81
4.4.2 Use of PRA to Confirm Safety Standards 83
4.4.3 Economic Impact 85
4.5 Case Study 4 - Control Rods 88
4.5.1 Use of PRA to Identify 88
4.5.2 Economic Impact 90
Chapter 5 – Summary and Future Work 91
5.1 Summary 91
5.2 Future Work 92
References 94
Appendix A – G4-ECONS Data for ALMR 98

List of Figures

Figure 1.1 - Predicted Energy Use by Fuel in the United States (1980-2030) ... 11
Figure 1.2 - Loop-Type and Pool-Type SFRs ... 14
Figure 1.3 - Japanese Method of Reducing Cost ... 16

Figure 2.1 - The Proposed Risk-Based Framework ... 19
Figure 2.2 - Frequency Consequence Curve from NUREG-1860 ... 23
Figure 2.3 - TNF Acceptability ... 27
Figure 2.4 - End State Classes from ALMR PRA ... 29
Figure 2.5 - Sample Event Tree from ALMR PRA ... 31

Figure 3.1 – Pool-Type SFR Nuclear Island Diagram ... 47
Figure 3.2 - US Nuclear Power Plant Capacity Factor (1973-2007) ... 59

Figure 4.1- Event Tree Showing Dominant Sequence ... 66
Figure 4.2 - Relative Size of Containment – ABR-1000 vs. ALMR ... 69
Figure 4.3 - Efficiency Comparison with Varying Core Outlet Temperature ... 80
Figure 4.4 - Sample Event Tree from PRISM ... 84
Figure 4.5 - Fault Tree for a Single Control Rod Insertion ... 89
Figure 4.6 - Fault Tree for RSS for Initiators which Require 1 Control Rod for Shutdown ... 89

List of Tables

Table 1.1 - World Net Electricity Consumption by Region (Billion kw-hr) 11
Table 1.2 - Gen IV Systems and Best Case Deployment Dates 12
Table 1.3 - Significant Worldwide SFRs 15

Table 2.1 - RAW values for systems in the PRISM design 33
Table 2.2 - Sample LEF Values for the PRISM 34
Table 2.3 - 1994 ALMR Direct Capital Cost Breakdown 36

Table 3.1 - Summary of Collected Data: Capital Costs (2007$) 40
Table 3.2 - Comparison of Percentage Capital Costs: ALMR and S-PRISM 41
Table 3.3 - Summary of Collected Data: Operation & Maintenance Costs (2007$) 41
Table 3.4 - Summary of Collected Data: Fuel Cycle Costs 42
Table 3.5 - Generic Inputs for G4-ECONS model 44
Table 3.6 - ALMR Performance Data 47
Table 3.7 - ALMR Inputs to G4-ECONS (2007$) 48
Table 3.8 - Unadjusted LUEC Results using ALMR numbers 49
Table 3.9 - Overnight Costs for Some Proposed Nuclear Plants [Du and Parsons, 2008] 49
Table 3.10 - Relative Cost by Country 50
Table 3.11 - Comparison of Benchmark LWR and Bellefonte LWR (2007$) 51
Table 3.12 - Direct Comparison of Capital Costs for LWR and SFR (2007$) 53
Table 3.13 - Modified Comparison of Capital Costs (2007$) 55
Table 3.14 - MIT Study LWR Cost Estimates (2007$) 55
Table 3.15 - Expected Future Cost by Fuel Type [Du and Parsons, 2008] 56
Table 3.16 - Adjusted LUEC for LWR and ALMR compared to NERA Proportions of Generating Cost (2007$) 57
Table 3.17 - LUEC with various Capacity Factors 59
Table 3.18 - Changes Required to Lower LUEC 5% 60
Table 3.19 - Summary of Economic Data 61

Table 4.1 - Generic Release Categories for ALMR 64
Table 4.2 - Containment Characteristics ABR-1000 vs. ALMR 70
Table 4.3 - COA 21 Comparison from G4-ECONS model 71
Table 4.4 - Maximum Effect of Containment on LUEC 71
Table 4.5- Total Savings from Removal or Simplification of Equipment 73
Table 4.6 - Effect on LUEC by Raising Core Outlet Temperature 73
Table 4.7 - Summary of Design Changes ($2007) 74
Table 4.8 - Effects of Higher Efficiency on LUEC 75
Table 4.9 - Results of Safety Analysis for 550C 76
Table 4.10 - Core Outlet Temperatures of existing SFRs 78
Table 4.11 - Effect of S-CO2 Cycle on Capital Cost and LUEC ($2007) 80
Table 4.12 - Maximum Potential Savings through Efficiency Gains ($2007) 81
Table 4.13 - Quality Standards from Regulatory Guide 1.26 86
Table 4.14 - ALMR High Grade Industrial Standards 87

Table 5.1 - Total Economic Benefits from Potential Design Alternatives 91

Table A.1 - ALMR Input Values for G4-ECONS 98
Table A.2 - ALMR outputs from G4-ECONS model 102

Chapter 1 - Introduction

1.1 Thesis Objectives

The objective of this thesis is to develop a methodology to reduce the capital cost and increase the competitiveness of sodium cooled fast reactors using a risk informed process in technology selection and design. Rather than applying traditional deterministic regulatory requirements to the design of sodium cooled fast reactors, the newly developed Technology Neutral Framework (TNF) [U.S. NRC, NUREG-1860] will be used to identify licensing basis events which will be used to judge safety of the plant. The TNF is a new methodology for licensing of nuclear reactors using probabilistic safety analysis and establishing a safety goal based on acceptance criteria which are dose based. Within this framework, certain event sequences when modeled in the PRA have a very low probability of causing significant consequences. By studying these events, opportunities for design simplification and cost reduction can be made without compromising safety. Deterministically based design criteria do not allow for such improvements in design or cost reduction. This approach may present the opportunity for refinement of the reactor design by revealing components that are unnecessary or possibly over-designed to compensate for requirements imposed according to current deterministic licensing requirements.

This project is part of a larger multi-university Nuclear Energy Research Initiative Project (Project # 08-020). The project team is made up of members from Massachusetts Institute of Technology, Ohio State University, and Idaho State University. The main goal of the overall project is to propose a methodology using risk-based methods to improve the Sodium-Cooled Fast Reactor and to develop and describe tools that support this methodology. Specifically, this thesis focuses on the economics of design choices and options using a probabilistic risk assessment outputs to judge the acceptability of design options that could reduce costs within the context of the TNF.

The objective is not to design a sodium cooled fast reactor but develop a systematic process to be used in design. The challenge of this project is obtaining reliable cost estimates in sufficient detail to test the methodology. To avoid absolutes, this project will focus on comparative assessments of designs and cost estimates completed in the past. The most well documented is the Advanced Liquid Metal Reactor [Gluekler, 1997] since it has detailed costs at the component level and a completed Probabilistic Risk Assessment. The goal is to show meaningful reductions in cost which will then be compared to light water reactors. The main tool that will be used in economic analysis is the G4-ECONS model, developed by the Generation IV International Forum (GIF) Economic Modeling Working Group. Once a reasonable model has been developed, areas for reduction in cost will be determined and specific items can be addressed. With the use of the G4ECONS model, sensitivity analyses will be performed to quantify the possible savings through specific changes to capital or operational cost, efficiency, and availability. It is expected that this methodology could be used in the design of any future reactor since the methodology will be generic.

1.2 Sodium Fast Reactor Background

Today, the world's population is growing at a rate of more than 1% annually [U.S. Census Bureau], with all people striving for a better quality of life. Growing along with population is the ever-increasing demand for energy, and all of the life-changing benefits that come with it. However, in recent times there has also been an increased awareness of the detrimental effect of this growth in demand on the environment, as traditional forms of power production tend to create pollution and deplete the natural resources in all regions of the world. Therefore, there must be a movement towards the development of technology that can produce the required energy without these negative impacts. A leader in clean, safe and cost-effective power production is nuclear energy. In their *2009 Annual Energy Outlook* [Energy Information Administration, 2009], the Energy Information Administration predicted a continued rise in electricity demand shown in Figure 1.1, indicating that the demand for nuclear power would continue to rise as well. According to the Table 1.1, the worldwide electricity demand is predicted to continue to grow at 2.3% [Energy Information Administration, 2004].

Figure 1.1 - Predicted Energy Use by Fuel in the United States (1980-2030)

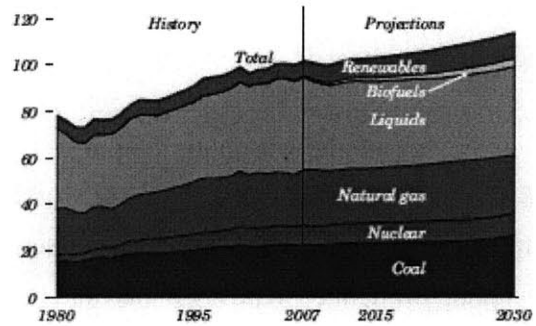


Table 1.1 - World Net Electricity Consumption by Region (Billion kw-hr)

Region/Country	2001	Projections 2010	2015	2020	2025	Average Annual Percent Change, 2001-2025
Industrialized Countries						
North America	**4,036**	**4,839**	**5,306**	**5,792**	**6,314**	**1.9**
United States*	3,386	4,055	4,429	4,811	5,207	1.8
Canada	500	578	630	680	728	1.6
Mexico	150	206	247	301	379	3.9
Western Europe	**2,246**	**2,486**	**2,659**	**2,839**	**3,029**	**1.3**
Industrialized Asia	**1,014**	**1,132**	**1,208**	**1,279**	**1,354**	**1.2**
Japan	788	870	920	965	1,012	1.0
Australia/New Zealand	226	262	288	314	342	1.8
Total Industrialized	**7,296**	**8,456**	**9,173**	**9,910**	**10,697**	**1.6**
EE/FSU						
Former Soviet Union	1,397	1,666	1,862	2,044	2,202	1.9
Eastern Europe	418	515	585	662	739	2.4
Total EE/FSU	**1,815**	**2,181**	**2,447**	**2,706**	**2,941**	**2.0**
Developing Countries						
Developing Asia	**2,650**	**3,723**	**4,508**	**5,342**	**6,274**	**3.7**
China	1,237	1,856	2,322	2,825	3,410	4.3
India	554	751	896	1,053	1,216	3.3
South Korea	231	318	371	419	468	3.0
Other Developing Asia	628	797	919	1,045	1,181	2.7
Middle East	**476**	**635**	**723**	**818**	**926**	**2.8**
Africa	**384**	**499**	**602**	**716**	**808**	**3.1**
Central and South America	**668**	**864**	**1,000**	**1,196**	**1,425**	**3.2**
Total Developing	**4,179**	**5,721**	**6,833**	**8,072**	**9,434**	**3.5**
Total World	**13,290**	**16,358**	**18,453**	**20,688**	**23,072**	**2.3**

At the end of 2008, there were 438 nuclear reactors in operation worldwide, producing 16% of total electricity, according to the International Atomic Energy Agency [IAEA, 2009]. In response to these increasing energy demands, the Agency predicts that nuclear energy will increase in importance and require new construction over the next 25 years.

Realizing that new technology would be necessary to meet these worldwide energy demands in a sustainable manner, ten countries – Argentina, Brazil, Canada, France, Japan, the Republic of Korea, South Africa, Switzerland, the United Kingdom and the United States – agreed to a system of international cooperation in these endeavors of research and development. Representatives from each of these ten countries joined together to form the Generation IV International Forum (GIF). The GIF defined four goal areas of focus for these advanced reactors: sustainability, economics, safety and reliability, and proliferation resistance.

Six potential systems were identified as the most promising technologies to focus on, in order to streamline the efforts on these specific projects. These six systems and their best-case deployment dates are listed in Table 1.2. As seen in the table, the Sodium Fast Reactor (SFR) is viewed as the most developed of the Gen IV technologies and will most likely be the first of these systems ready for deployment [U.S. DOE and GIF, 2002].

Table 1.2 - Gen IV Systems and Best Case Deployment Dates[1]

Sodium Fast Reactor	2015
Very High Temp Reactor	2020
Gas Cooled Fast Reactor	2025
Molten Salt Reactor	2025
Super-Critical Water Reactor	2025
Lead-Cooled Fast Reactor	2025

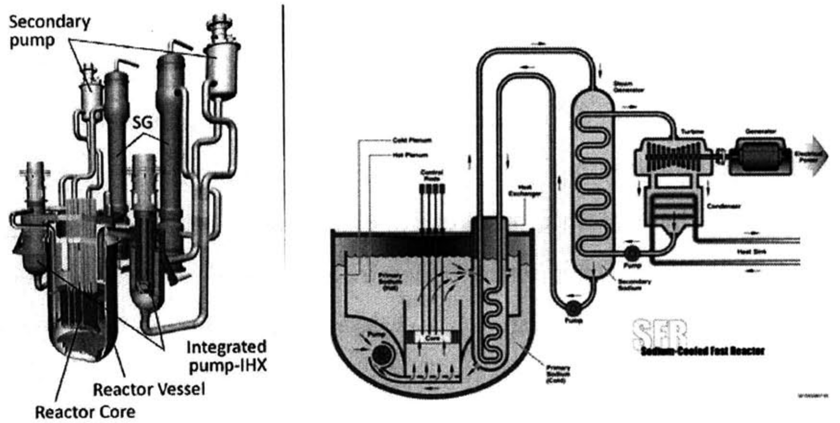
[1] Taken from the Gen-IV Roadmap, though not judged to be realistic at this point

1.2.1 The Sodium Fast Reactor

The Sodium Fast Reactor uses the fast neutron spectrum to maintain fission and produce energy. The majority of natural uranium is the isotope U-238, making up about 99.3%. The remaining 0.7% is U-235, the isotope required for thermal fission in modern light water reactors. The fast neutrons are used to breed plutonium from the U-238, and these plutonium isotopes then undergo fission to produce heat. Therefore, the fast reactors can utilize uranium much more efficiently than a thermal reactor. Because water acts a moderator and will slow the neutrons out of the fast spectrum, liquid metals such as sodium are used as the coolant in these fast reactors, transferring the heat from the core to a power conversion system used to produce electricity. Besides the advantage of more efficient use of natural uranium, SFRs can also be used to breed fuel since they can be designed to produce more fuel than they consume by the use of an external ring of U-238, where the plutonium is bred for reprocessing and subsequent recycle. In addition, fast reactors can be used for transmutation of nuclear waste with long half-lives into less troublesome isotopes that will decay on a much shorter timescale [World Nuclear Association, 2009].

The GIF has ranked the SFR as a top prospect for the support of its goals in the four areas previously mentioned. It is top-ranked in sustainability due to its closed fuel cycle and potential for actinide management. The sustainability and actinide missions of the sodium cooled fast reactors are the main drivers for developing this technology. It has also been ranked as good in the other three areas: safety, economics, and proliferation resistance. Since there are several operating sodium fast reactors around the world, most notably in Russia, France and Japan, this technology is seen as being deployable with much less research and development than may be required for some of the other technologies [U.S. DOE and GIF, 2002].

Figure 1.2 - Loop-Type and Pool-Type SFRs



Current Loop-type Design
i.e. JSFR

Current Pool-type Design
i.e. EFR (EU), ABR (US)

In Figure 1.2, basic schematics of both the pool-type and loop-type SFRs are shown. On the left, the pool-type SFR has all reactor internals located within the primary sodium pool, including the intermediate heat exchanger. The primary sodium never leaves the pool, making a loss of coolant accident extremely unlikely. The pool-type will have a larger reactor vessel, but will be simpler to construct and will have a smaller reactor building since more of the required equipment will be located within the reactor vessel [Zhao, 2009].

The loop-type SFR has a more traditional reactor vessel with a primary sodium loop which is connected to an external intermediate sodium-to-sodium heat exchanger which is then connected to a steam generator for the production of power. The loop-type reactor has a smaller, but more complicated reactor vessel, with more components located outside of the vessel itself. This means that the external loops will require shielding, but permits easier in-service inspection and maintenance which is one of its major advantages.

1.2.2 History of SFR Development

Sodium fast reactors had their start in 1946 with the US Clementine reactor at Los Alamos National Laboratory. The initial concern was the supply of uranium to support the development of nuclear power. While the US held the early lead in sodium reactor development with the Experiment Breeder Reactor I and II, the US lead was lost to France and Russia who maintained a fast reactor program while the US dropped its efforts in the 1980's. The initial application was to breed plutonium for electricity production. In total, there have been about 20 SFRs constructed worldwide. The most significant are shown in Table 1.3 [IAEA, 2006].

Table 1.3 - Significant Worldwide SFRs

Name	**Country**	**Power (MWe)**	**Year**	**Loop or Pool**
Clementine	US		1946	
EBR-1	US	0.2	1951	
BR 5	Russia		1959	Loop
Dounreay FR	UK	15	1959	Loop
EBR-2	US	20	1963	Pool
Fermi 1	US	66	1963	Loop
BOR 60*	Russia	12	1969	Loop
BN350	Kazakhstan	130	1972	Loop
Phenix*	France	250	1973	Pool
Prototype FR	UK	270	1974	Loop
KNK II	Germany	21	1977	Loop
BN600*	Russia	600	1980	Pool
Fast Flux Test Fac.	US	400	1980	Loop
Superphenix	France	1240	1985	Pool
FBTR*	India	13	1985	Loop
MONJU*	Japan	280	1994	Loop

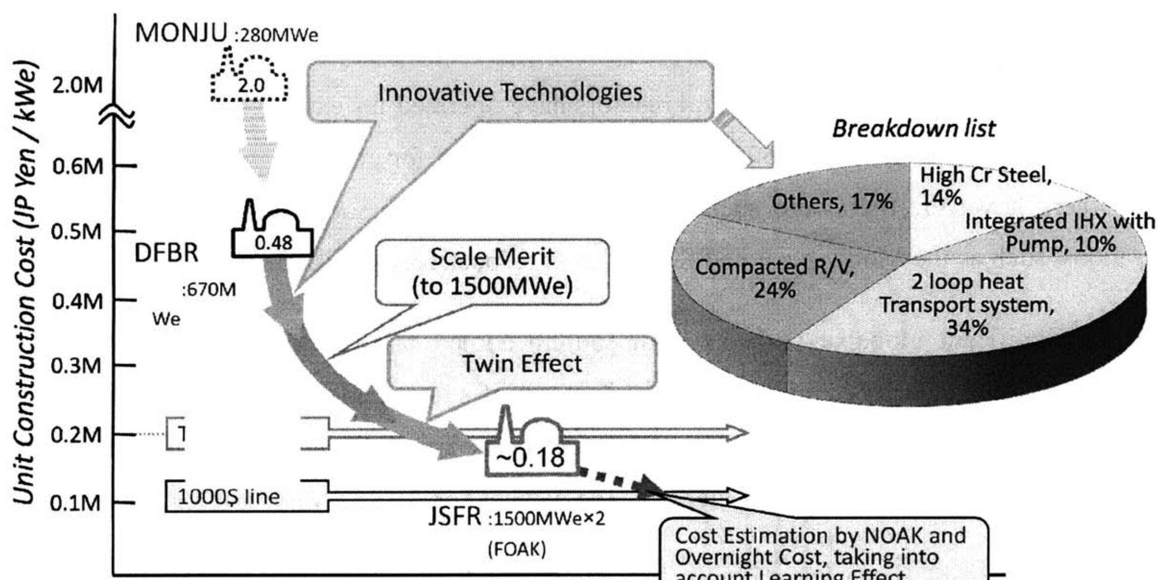
Of these reactors, five (marked by *) are operating or scheduled to resume operations as of 2008. The EBR-2 was one of the most significant in demonstrating the safety potential of the SFR. It was operated for 30 years without a major accident, and was used to demonstrate the passive safety nature of pool type SFRs. Several reactors have operated for extended periods, including Fermi 1 for 15 years and the BN600 that has been operating for close to 30 years. There have been others however, that were shutdown prematurely due to sodium leaks and unreliable operations, such as the Monju plant and SuperPhenix respectively [Carlson, 2009].

There is significant global experience with sodium cooled fast reactors but the operational reliability has not met expectations. The capital cost of sodium cooled fast reactors is higher than light water reactors and, at present, the fuel costs are also higher due to the need to reprocess and re-fabricate plutonium fuel from the blanket zones.

1.2.3 Economic Issues

The challenge of developing an economically competitive reactor has been recognized by the Generation IV initiative as a major obstacle impeding the deployment of these fast reactor systems. The major challenge is the relatively high initial capital cost. As a result of the Generation IV initiative, countries such as Japan and France have focused their design efforts on redesigning the plant within the constraint of the safety requirements which are more challenging for a sodium cooled system than a water system. The Japanese Atomic Energy Administration's (JAEA) philosophy for capital cost reduction is to reduce the amount of material used in construction by designing a larger plant while reducing the amount of steel required per unit electricity produced. The JAEA believe that the benefits of scale will be realized through a large, monolithic reactor with several loops. They also have attempted to design the loops while using as little piping as possible to reduce additional cost. Finally, the JAEA is taking advantage of advances in technology by assuming the development of new materials such as high-strength steels and new components such as integrated heat exchangers and pumps because they foresee this technology being available by the predicted SFR deployment dates. This approach is shown in Figure 1.3 [JAEA, 2009].

Figure 1.3 - Japanese Method of Reducing Cost



The unit construction cost of Monju is expressed as the construction cost divided by electric power.
The unit construction cost of DFBR and JSFR are evaluated value

General Electric has taken a different approach to making the SFR economically competitive. Their approach uses smaller reactors that allow the benefits of factory fabrication of components and passive safety systems to be realized, while constructing each plant on a shorter timescale [Boardman, 2000]. It is unclear at this time which approach will prove to be more successful at reducing costs.

The economics of the SFR are driven in part by the need to maintain overall safety and the uncertainty of the future licensing requirements of these advanced reactors. In past designs, a large focus has been placed on dealing with the events such as fuel failure and energetic core disruptive accidents. In the early days of the United States fast reactor breeder program, the construction of the Clinch River Breeder Reactor (CRBR) was attempted, but ultimately canceled due to escalating costs of construction and an unsure licensing and political climate.

The CRBR was estimated in 1972 to cost $699 million based on initial designs and estimates. By the time the project was ultimately canceled after 11 years of development, the estimated cost had ballooned to over $2.5 billion, and the schedule had been delayed to the point where the reactor would not come online earlier than 1989. These increasing costs and schedule delays were mainly caused due to funding restrictions and constantly changing licensing requirements, resulting in multiple redesigns and the numerous additional safety systems [U.S. DOE, 1983]. Learning from this project, it becomes obvious that a better understanding of the licensing environment and the requirements on these reactors is necessary. This can ensure that the construction and startup occurs in a timely manner and within budget, preventing situations like CRBR from occurring again.

Finally, a major issue that has affected the Sodium Fast Reactors has been their unreliability due to costly technical problems with sodium systems. In France, SuperPhenix was shut down repeatedly through its ten year operating life due to several sodium leaks [IAEA, 2006]. In Japan, Monju encountered a thermo-well weld failure and sodium leak, resulting in a fire [Carlson, 2009]. These failures have led to the permanent shutdown of the SuperPhenix and the prolonged shutdown of the Monju plant in Japan. These operational issues significantly affect the economics and perception of fast reactors as unreliable producers of electricity and need to be addressed before SFRs are deployed in large numbers.

1.3 Thesis Outline

Chapter 2 – Methodology and Framework

Chapter 2 describes the development of the risk-based methodology that will be proposed for use in this project. The process is described in a step-by-step manner, demonstrating the progression from deciding on a design alternative, confirming that the design alternative meets safety requirements, and finally to the determination of economic benefit. The process of confirming safety compliance using this risk-based methodology and the Technology Neutral Framework is described in detail.

Chapter 3 – Development of the Economic Model

Chapter 3 details the process of developing the reference economic model. The chapter begins with a look into the current range of estimates for the different components of SFR cost: capital cost, fuel cycle, operations and maintenance, and decommissioning and disposal. Then using the ALMR as a reference design, an economic model is developed in as much detail as possible using the G4-ECONS model. The validity of this economic model is discussed, as well as the limitations.

Chapter 4 – Demonstration of Methodology through Case Studies

Chapter 4 is the culmination of the project, where the methodology is demonstrated to identify possible economic gain in terms of reduced electricity cost. Methods of identifying potential design alternatives are shown using risk-based methods and quantified using the economic model. Each step in the methodology is shown to illuminate the use of this methodology. One example, the removal of the Energetic Core Disruptive Accident from possible Licensing Basis Events, is shown as a major catalyst for savings. Another example, manufacturing the Steam Generator as non-safety grade, initially appears to have the potential for large savings, but ultimately results in little gains after performing the economic analysis. Overall, these case studies demonstrate the use of the different tools that were developed and described in the previous chapters.

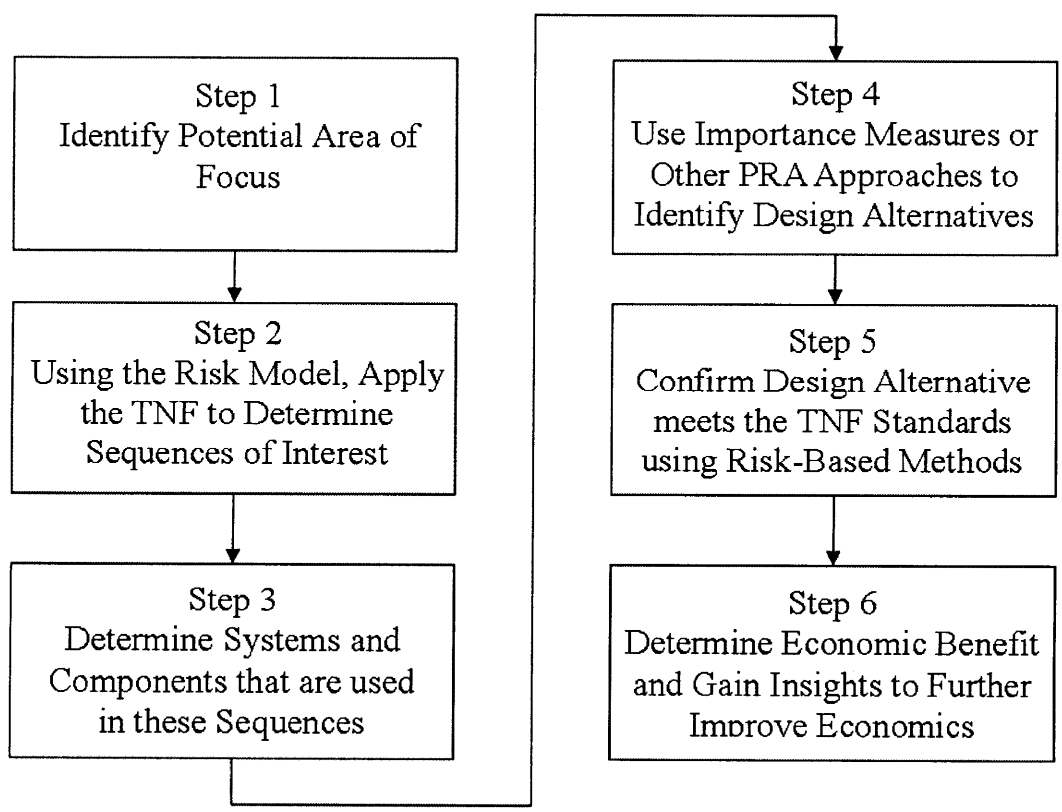
Chapter 2 - Methodology and Framework

2.1 Developing the Framework and Methodology

2.1.1 Framework and Methodology Description

In an effort to reduce the cost of the Sodium Fast Reactor to levels where it can be economically competitive in the electricity market, a risk-based methodology is proposed. The main goal of this methodology is to enable a process through which economic improvements can be made on existing reactor designs, while still conforming to safety requirements. The methodology will utilize the flexibility in design allowed in the Technology Neutral Framework proposed by the U.S. NRC in NUREG-1860. Once the necessary tools are developed, this methodology and framework will be a useful method to reduce the overall cost of electricity. Figure 2.1 graphically illustrates the basic methodology which is explained below.

Figure 2.1 - The Proposed Risk-Based Framework



This process begins by identifying an area of focus. This may be done several ways: through the identification of major cost drivers, through deterministic improvements such as increasing efficiency, or through identification of areas with unnecessarily higher safety margins than required under the Technology Neutral Framework.

Once the area of focus is determined, the risk-based methods must be used to determine all potential effects that might affect the safety. The most effective tool for doing this is the reactor-specific PRA, which will be central to the development of a safety model. All sequences that may be affected by a proposed change must be identified. These sequences are defined by event trees and fault trees that describe the reactor's response to different scenarios based on the performance of the components in each system.

Once these sequences are identified, a list of the systems, structures and components (SSCs) that perform the necessary functions can be compiled. The PRA will show the level of contribution for each of the SSCs and their overall importance to the safety case. At this point, different risk-based methods may be used to identify the importance of a system, structure or component to safety. These risk-based methods include the use of importance measures such as the Risk Achievement Worth (RAW) and the Limit Exceedance Factor (LEF). The systems, structures or components with a low RAW or high LEF are candidates for removal, lowering of safety grade or design modification without significantly affecting the safety of the plant while possibly reducing the cost. These will be explained in detail later in this chapter.

The proposed design alternative must be checked against the TNF safety framework to ensure that it continues to meet the safety standard as established in the Technology Neutral Framework. During this step, confirmation of the safety analysis of the plant will be required to support the event and fault tree analysis of the plant to be sure that the change can be implemented without significant changes in the safety performance of the plant. If the design alternative is shown to not meet the standard, the idea must be reformulated in such a way that it will comply. If the design alternative does meet the standard, it can continue to the next stage.

Once the design alternative has been shown to meet all standards of safety, the economic benefit of the proposed change must be determined. There are two main economic figures that are important to the scope of this framework: overnight capital cost given in \$/kwe and Levelized

Unit Electricity Cost (LUEC) in mills/kw-hr. The LUEC is the most important, since this represents the base cost of electricity production for this reactor. The busbar LUEC for the Sodium Fast Reactor can be directly compared to a busbar LUEC for any other power plant type, showing the relative cost of the SFR against light water reactors or traditional fossil fuel power plants. This direct comparison will show how competitive the SFR can be against other power generation sources. The capital cost represents the upfront investment that must be made to build a Sodium Fast Reactor. For nuclear plants in general and for sodium cooled fast reactors, the capital cost is the largest contributor to the cost of electricity. The larger capital cost represents increased risk to potential investors, especially until the technology has been demonstrated to be sufficiently reliable to justify this large initial investment.

The economic benefit will be determined by how the design alternative affects either the LUEC or the overnight capital cost. It is important that all aspects such as the impact on reliability, maintenance or operability of the design alternative are taken into consideration, not only the effects on capital cost. At this point, the economic consequences of the design alternative should be reviewed, as the process may have illuminated other areas that should be considered as areas for possible economic improvement.

2.1.2 Required Tools for this Framework

The two main tools that will be essential to this methodology are a Probabilistic Risk Assessment (PRA) model and an economic model. The PRA model is an integral part of this methodology, as it is used to identify sequences of interest and to check that the design alternative meets the safety requirements as specified in the Technology Neutral Framework. The economic model will be very important for the final step of the methodology, to determine the overall economic effect of the design alternative. The economic model must provide results in terms of the busbar LUEC and overnight capital cost in order to allow for a full economic consequence analysis of the potential design alternatives.

Both of these models should be easily modified to account for changes that will be suggested through the proposed design alternatives. They should also have a large range of potential inputs that can be modified to allow for the most flexibility in design analysis. If possible, these models should be user-friendly and present the results in a clear and concise manner, for the greatest ease of use.

Traditional design analysis tools are also necessary to complete this analysis. These tools include safety and thermal analysis codes to confirm that the designs proposed do not compromise the overall safety or performance of the plant. The results of these analyses are then fed back into the PRA model for reanalysis to confirm TNF acceptance.

2.1.3 Technology Neutral Framework Background

As these advanced reactor types are being developed, the Nuclear Regulatory Commission (NRC) has begun to realize that the current licensing process may impede these new technologies. The current licensing requirements in the NRC's Code of Federal Regulations (CFRs) are largely deterministic which were created to license light water reactors (LWRs). The NRC is developing a preliminary framework of a risk-informed and performance based licensing structure that may be used to license future non-water nuclear power plants and to allow for a more risk-informed design of light water reactors. This framework has been called the Technology Neutral Framework (TNF), since it will be focused on determining requirements based on safety and risk, regardless of the technology.

The TNF would allow for a broader use of technology specific risk information using a probabilistic risk assessment developed for each reactor design. This allows the safety analysis and regulatory oversight to focus on the items most important to the safety for that design. The framework would stress safety performance as the metric for acceptability, giving the designers more flexibility to decide on features most appropriate to their design [U.S. NRC, NUREG - 1860].

2.2 The Probabilistic Risk Assessment Model

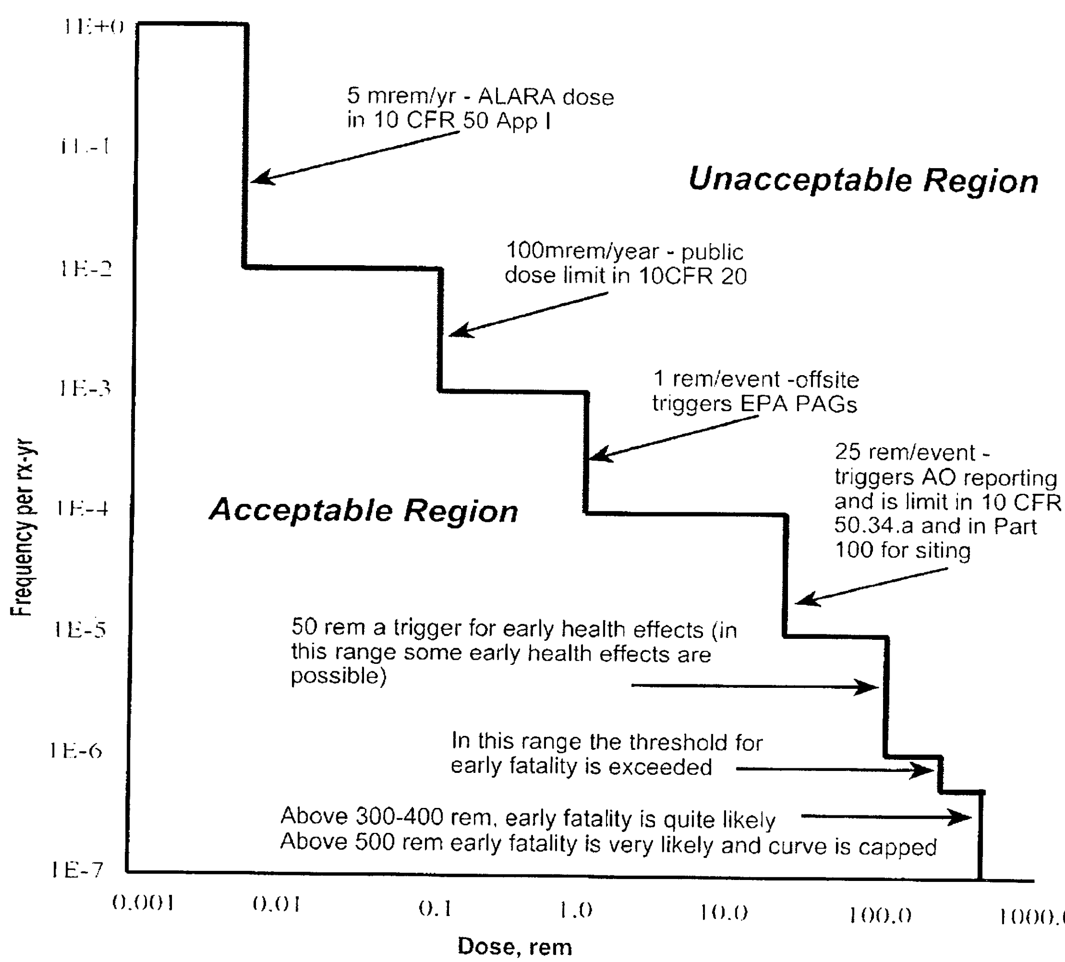
In order to use the proposed framework, a probabilistic risk assessment model is needed to support the Technology Neutral Framework approach. A probabilistic safety model will be largely design specific, since individual probabilities and frequencies are assigned based on the SSCs of the particular design. In this thesis, two available PRAs will be used: one of the

Advanced Liquid Metal Reactor (ALMR) [El-Shiekh, 1994] and the other the PRISM reactor [Hackford, 1986].

2.2.1 Using the Technology Neutral Framework

The Technology Neutral Framework presents an opportunity to make use of the PRA to systems, structures and components in terms of safety importance. Using the TNF approach, Licensing Basis Events (LBE) for which must be designed are determined using the results of the PRA. Using PRA, events are classified according to their frequency and consequences as represented as a possible public radiation dose. Figure 2.2 shows the proposed frequency-consequence curve [U.S. NRC, NUREG 1860].

Figure 2.2 - Frequency Consequence Curve from NUREG-1860



Any event with a mean frequency greater than 10^{-7} per reactor year is subject to the requirements of this curve. An event needs not to be considered as a LBE if the event can be shown through the PRA to have a point estimate frequency below 10^{-8} or if the mean frequency is below 10^{-7} per reactor year. Therefore, if all possible events initiated by failures from a single system, structure or component fall below these limits, this item may be targeted for possible removal or simplification and thus potential cost reduction.

2.2.2 Probabilistic Risk Assessment

The main way to determine frequencies and consequences of certain events is to use Probabilistic Risk Assessment. Through the use of PRA, fault trees and event trees can be constructed and the overall frequencies for different possible end states are produced. Based on the consequences associated with each of these end states, the worst-case scenario for each event can be identified. This combination of a frequency and consequence can be used with the TNF to determine whether or not the sequence falls within the acceptable or unacceptable region of the Frequency-Consequence Curve in Figure 2.2.

The main tool that can be used in the PRA analysis of Sodium Fast Reactors is the collection of event trees developed in the ALMR [El-Shiekh, 1994] and PRISM PRA [Hackford, 1986] reports. Since the reference reactors used for this study are based on a pool-type ALMR reactor, the fault trees, event trees, initiating event frequencies and system reliabilities from these reports will be representative of the pool-type design used as a reference model. There are two main types of PRA analysis: Level 1 and Level 2/3. Level 1 PRA estimates the frequency of core damage. This is usually based on the safety equipment included in the specific reactor designs, so the confidence in Level 1 PRA results tends to be relatively high. Level 2 and 3 PRA estimate the source terms, magnitude of releases and the possible consequences from these releases. These analyses tend to contain more unpredictable variables, and will have more uncertainty [U.S. NRC, Fact Sheet, 2007].

The Appendices of the ALMR PRA and PRISM PRA contain event trees for many different initiating events, describing the possible sequences that could occur and the final core damage state that would result from each. These have been analyzed to best understand the sequences with extremely low frequencies, as well as the sequences that may require special consideration within a risk-informed methodology.

2.2.3 Determine Safety Requirements Using the TNF

Using the TNF, there is a specific process that must be followed to determine the safety requirements. The design-specific PRA should be used to select all sequences that represent potentially risk-significant accident challenges. These should include all frequent, infrequent and rare initiating event sequences. Once this list is compiled, the PRA analysis can be used to determine where each of these events falls in respect to the Frequency-Consequence Curve. Licensing Basis Events (LBEs) are determined through a process of binning sequences which all share similar phenomenology. If any sequence in a bin has a point estimate frequency higher than 10^{-8}, then the sequences must be analyzed further to determine the mean frequencies. At this point, if the mean frequency of any sequence in the bin is above 10^{-7}, then this bin will be considered an LBE. The highest frequency of all sequences in each bin is selected and paired with the highest consequence of all sequences in the same bin. This is then the point that is plotted on the Frequency-Consequence curve as an LBE.

A stringent requirement of the TNF regarding the LBEs states that only SSC's that are considered as safety grade items can be credited in the LBE analysis and are subject to special treatment. Therefore, any function or capability of an SSC that is not safety grade must be set to have a failure probability of 1.0 for guaranteed failure. This means that only safety-related components are credited for prevention or mitigation of any event. This requirement is very important for the analysis of whether or not items should be manufactured as safety grade or non-safety grade. A component is not important as a safety grade item if its safety functions can be removed from the PRA with minimal consequences. Removal of the safety grade status on a component can result in significant cost reduction, which needs to be considered in achieving the goal of economic performance.

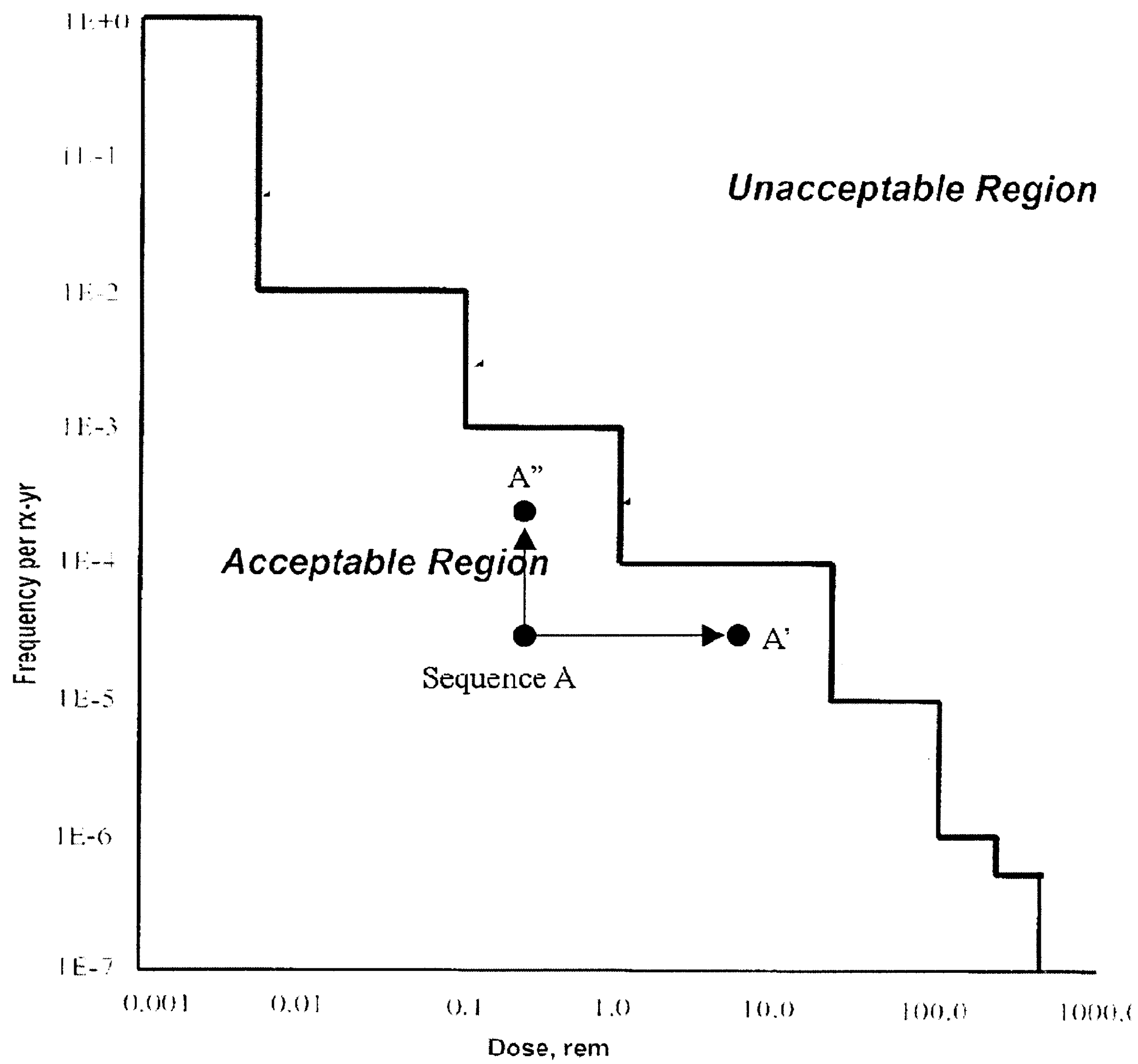
There are two ways to meet the requirements of the TNF. The first is to determine that the PRA sequence will have a frequency point estimate of less than 10^{-8}/yr. The NUREG specifically instructs to "drop" these from consideration as a Licensing Basis Event. The second method is to confirm that the event falls within the acceptable region of the curve based on risk and consequence.

If a sequence has a point estimate of greater than 10^{-8}/yr and through uncertainty analysis can be shown to have a mean frequency of greater than 10^{-7}/yr, then it must be analyzed further. For such events that must be considered, the consequence of its end state must be determined through Level 2 and 3 analyses to ensure that the associated frequency and consequence will fall within the acceptable region of the curve. These higher frequency events may still have room for savings if the consequences remain within the limits of the TNF for the determined event frequency.

Any changes to the design of the reactor must be checked against these TNF standards using the PRA approach. This is most easily done in the situation where an SSC is removed entirely or changed from safety-significant component to a commercial grade component. In this instance, all functions of this component within the PRA can be set to always fail.

Sequences with mean frequencies higher than 10^{-7} must fall within the acceptable region of the Frequency-Consequence Curve. The higher frequencies do not necessarily mean that the design change (ex. Component or system removed, simplified, or lowered in safety grade status) is unacceptable. For example, in Figure 2.3, Sequence A lies within the acceptable region of the curve and would meet the safety requirements of the TNF. However, there is room under the curve in both directions indicating that changes in the design may be permitted. If a design alternative was proposed that resulted in a higher dose while maintaining the same frequency, the resulting sequence could be plotted at point A'. Likewise, a design alternative could be proposed that would result in a higher frequency without significantly changing the dose, moving the sequence to point A". For the sequence illustrated above, either of those design alternatives would be acceptable under the TNF, and the one with the greatest economic benefit should be selected.

Figure 2.3 - TNF Acceptability



This analysis is more complicated when an operating condition is changed. For example, if the design core outlet temperature was increased, the probabilities of failure for many different components may be affected in many different sequences. These probabilities must be determined through modeling and other analyses before the PRA analysis can be performed [U.S. NRC, NUREG-1860].

2.2.4 Using PRA Analysis

When using the TNF as the standard for safety acceptance criteria, Level 1 PRA analysis is needed. A plant specific PRA will include a collection of event trees that describe possible accident scenarios and the probabilities of success or failure at each possible opportunity for prevention or mitigation. At the end of each Level 1 event tree, there are many possible end states that may be reached depending on the success or failure of each of the SSCs within the tree, which defines a set of similar consequences for the end states. Although each possible end state will be different depending on the mitigating circumstances, groups of similar end states are binned together as classes. These classes identify the defining attribute that describes the end state. Figure 2.4 is a page taken out of the ALMR PRA [El-Sheikh, 1994] describing the possible classes that can result from the various event trees.

Figure 2.4 - End State Classes from ALMR PRA

DVL	Double vessel (reactor vessel and containment vessel) leak
LOF3-4	Loss of power to the primary EMPs with 3 or 4 synchronous machines operating
LOF3-4CD	Same as LOF3-4
LOF2CD	Loss of power to the primary EMPs with 2 synchronous machines operating
LOF1CD	Loss of power to the primary EMPs with only 1 synchronous machine operating
LOF0CD	Loss of power to the primary EMPs with none of the synchronous machines operating
OP-4D	Less than 10% safe overpower operation for 4 days
OK	Safe shutdown or power operation.
RUTOP	Reference ATWS UTOP
RULOF	Reference ATWS ULOF
RULOHS	Reference ATWS ULOHS
SD	Same as OK for safe shutdown
2SIGSD	Shutdown with the decay heat level at the 2-sigma level (115% of nominal)
SPO	Same as OK for safe power operation
BUTOPLHS	Benign combined UTOP/ULOHS accident (with consequences similar to those of reference ALMR ATWS accidents)

Core Damage Categories:

A	Creep rupture of the fuel clad for up to 25% of the fuel assemblies
B	Category A plus eutectic attack of up to 25% more fuel assemblies
C	Fuel melting and dispersal of up to 25% of the core
D	Whole core meltdown

Based on preliminary Level 2/3 consequence analysis, the estimated dose can be calculated for each of the possible end states. Depending on the sequence frequency, the associated sequence end state must be labeled as acceptable or unacceptable, according to the TNF guidelines.

In order to more easily analyze the event trees given in a PRA, the event trees with their associated probabilities can be entered into a code such as the Systems Analysis Program for Hands-on Integrated Reliability Evaluation (SAPHIRE). SAPHIRE allows the user to modify probabilities or remove components from the event trees entered from the PRA. The code will then output the new probabilities of reaching each possible end state. This capability allows the user to track the effect of changes made to any system, especially the probabilities of reaching the most damaging end states.

Figure 2.5 is one event tree from the ALMR PRA showing the possible progression of events following a 0.6-1.0g Seismic Event. The event tree shows the initiating event frequency and all possible mitigating steps with their associated probabilities of success and failure. For this event tree, the possible end states are OK, A and C. Referring to Figure 2.4, OK will always be considered acceptable and both A and C may be unacceptable, since they result in core damage. It must be determined using Level 2/3 analysis whether end states A and C are acceptable or unacceptable. Once the criteria for acceptable and unacceptable are established, the associated probabilities can be determined. In this example however, due to the low initiating event frequency of 8×10^{-5} and the high scram reliability, most of the sequence probabilities fall well below the 10^{-8} threshold for point estimates, and will be screened from consideration as potential LBEs. The point estimates that are above the threshold may be considered in the licensing process.

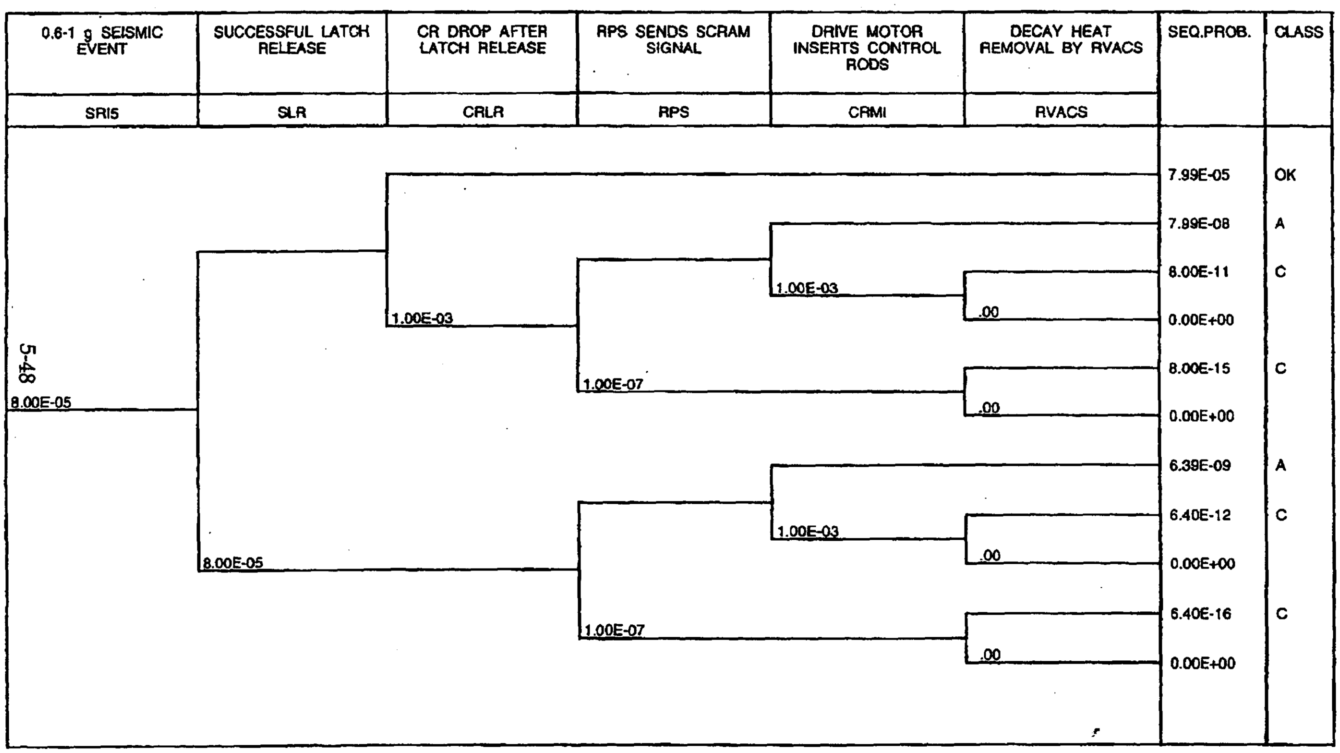


Figure 5-25 0.6 - 1.0 g Seismic Event

2.3 Using Importance Measures to Analyze Safety within the TNF

The SAPHIRE code can be used to process PRA results in a relatively easy manner to identify contributions of systems, structures and components to the overall safety of the plant. There are many useful importance measures that can be used as metrics in determining the safety importance of SSC's. Those considered in this analysis are Risk Achievement Worth (RAW) and Limit Exceedance Factor (LEF).

Risk Achievement Worth

One importance measure that can be used is the Risk Achievement Worth (RAW). The RAW for a component is the conditional probability of overall system failure given component failure divided by the failure probability of the system under normal conditions [Cheok, et al. 1998]:

Equation 2.1 - Risk Achievement Worth

$$RAW = \frac{P(System\ Failure|Component\ Failure)}{P(System\ Failure)}$$

The RAW value of components can be easily calculated using the SAPHIRE code, simply by changing the probability of failure for every function of the component to 1.0. The probability of system failure is determined by taking the sum of the probabilities for all classes that have been identified as failed end states.

The RAW value that is obtained can be 1.0 or higher. A RAW value of 1.0 indicates that the system does not rely heavily on that component, but a high RAW value does not provide sufficient information to determine the importance of the component. A high RAW does not necessarily mean that the component cannot be changed or modified, instead it merely indicates that the component and its functions should not be eliminated. Table 2.1 presents several RAW values for systems in the PRISM design, using a large release of radioactivity as the failure criteria [Johnson and Apostolakis, 2009].

Table 2.1 - RAW values for systems in the PRISM design

Event	RAW (Large Release)
Reactor Protection System/Plant Control System Signal	1.55×10^2
Reactor Shutdown System (Scram)	7.87×10^6
Pump Coast Down	2.021
Nominal Inherent Reactivity Feedback	1.992
Operating Power Heat Removal	1.01
Shutdown Heat Removal through the Intermediate Heat Transfer System	1.90×10^3
Reactor Vessel Air Cooling System (RVACS)	2.42×10^4

The RAW value is an indicator that can show directly the significance of each component in the overall safety case of the reactor. If this component is not important is can be removed or have it safety classification changed. As previously explained, any component that is not manufactured to safety grade standards cannot be credited in the LBE analysis. For example, if a design option was to remove the pump coastdown from the PRISM reactor design, the PRA and supporting safety analysis would have to be modified so that any increase in safety provided by the pump coastdown is not accounted for during any event tree. The probability of failure under these new conditions would be the previous probability multiplied by the RAW value, 2.021 in the case of PRISM. In analyses such as these, where SSCs are being removed or downgraded out of the safety grade categories, the RAW value can be a very quick and useful tool to use in this methodology.

Limit Exceedence Factor

Another importance measure that can be of use in the safety analyses under the TNF methodology is the Limit Exceedence Factor (LEF). This importance measure is valuable because it measures importance of a specific component with respect to the maximum allowable risk within a system. The LEF is the probability of failure for a component that causes the overall sequence frequency to be equal to the limit divided by the original component failure probability.

Equation 2.2 - Limit Exceedence Factor

$$LEF = \frac{P(Comp. Fail.) so\ that\ P(System\ Fail.) = Limit}{P(Component\ Failure)}$$

Specifically under the TNF, the numerator of the LEF represents the frequency of failure that would make the overall system failure for the most dominant sequence a value of 10^{-8}/yr. For any system with extremely low frequencies, the LEF will be larger than unity, with larger values of LEF indicating more margin to the limit. Table 2.2 shows several values of LEF as determined from the PRISM PRA. Items with large LEF values may be identified as possible components that can be targeted for simplification [Johnson and Apostolakis, 2009].

Table 2.2 - Sample LEF Values for the PRISM

Event	LEF
Reactor Protection System/ PCS Signal	1.9E7
Reactor Shutdown System (SCRAM)	38
Shutdown Heat Removal through Intermediate Loop	7.8
Reactor Vessel Air Cooling System (RVACS)	7.2

2.3.1 Identifying with PRA and Importance Measures

In the previous description, it was shown how importance measures such as the RAW and LEF could be used in the safety analyses of the methodology. The RAW value for any given component can be used to identify possible candidates for removal from a specific design. Any component with a RAW value close to unity is not very important for meeting the safety requirements under the TNF, and therefore they may be removed completely or at least removed simply from the safety grade category. Removing a component completely can potentially save on capital cost, and removing a component from safety grade may save on capital cost and has the potential to also simplify the licensing process.

The LEF metric assessed the margin of the SSCs, indicating components that provide excessive margin or reliability beyond what is required in the TNF. These components may serve an important function in the safety framework, which would lead to a high RAW, but may not be required to be as reliable to ensure frequencies below 10^{-8}/yr. Identifying components with high LEF values may allow for simplification of systems, which can lead to potential capital cost savings.

Finally, the PRA model can be used in conjunction with the Frequency Consequence Curve to determine components that may be more reliable (very low failure rate or too much redundancy for function) than necessary, even for frequencies above the 10^{-8} threshold. Any changes in design will require a safety re-analysis and an updated Level 2/3 consequence analysis to confirm that the doses are within acceptable levels.

2.4 Identifying Potential Design Alternatives

The risk assessment approach is a method for identifying systems component and structures that are not safety significant followed by subsequent confirmatory analysis and determination of potential cost savings. Another approach is to focus on high capital cost areas of the plant to see if changes in design can be made to reduce these costs. Using an economic model of the plant, areas of high cost and potential high value reductions can be identified. Savings in capital cost or increases in efficiency or capacity factor will have the greatest impact on the levelized cost of electricity. Therefore, these are the areas that should be concentrated on most heavily.

2.4.1 Identifying with the Economic Data

Once the economic model has been developed, it can be used as a tool to identify high cost areas within the specific design. Capital cost has been determined as the largest contributor to the LUEC. As will be shown in Chapter 3, a reduction in capital cost by 7.4% can result in a LUEC reduction of 5%. Table 2.3 contains an abridged list by category of the Direct Capital Costs (Equipment) of one ALMR reactor block, developed for the 1994 ALMR Cost Estimate [Gokcek, 1995]. These costs represent the overnight value of the reactor block, and do not account for the construction cost. The second column shows the percentage of total direct capital cost for each category.

Using this information, specific areas of construction can be identified as the largest contributors to the total capital cost. For example, the steam generator makes up almost 10.5% of the total capital cost, the turbine generator makes up almost 9%, and the reactor internals almost 8%.

These large percentages may indicate that there could be room for significant overall savings and can provide an initial indication of areas on which to focus effort. In a case study in Chapter 4, possible savings through the steam generator are explored.

Table 2.3 - 1994 ALMR Direct Capital Cost Breakdown

(Equipment Cost Only)

Account No		1994 k$	
211	Yard Work	$16,116.00	2.22%
212	Reactor Facilities	$72,804.00	10.02%
213	Turbine Generator Building	$9,678.00	1.33%
218N	Maintenance Shop	$11,087.00	1.53%
21 Total		**$137,410.00**	**18.92%**
220A.211	Reactor Vessels	$25,602.00	3.53%
220A.212	Reactor Internals	$55,595.00	7.65%
220A.213	Control Rod Drives	$9,676.00	1.33%
220A.221	Primary Heat Transport System	$46,241.00	6.37%
220A.222	Intermediate Heat Transport System	$48,400.00	6.66%
220A.223	Steam Generator System	$75,702.00	10.42%
220A.231	Back-up Heat Removal System	$1,612.00	0.22%
220A.15	Fuel Handling and Storage	$7,677.00	1.06%
220A.268	Maintenance Equipment	$25,878.00	3.56%
220A.27	Instrumentation and Control	$20,260.00	2.79%
220A.31	Support Engineering	$38,414.00	5.29%
225	Fuel Handling	$10,805.00	1.49%
22 Total		**$403,029.00**	**55.49%**
231	Turbine Generator	$64,601.00	8.89%
23 Total		**$95,424.00**	**13.14%**
242	Station Service Equipment	$19,554.00	2.69%
245	Electric Structure and Wiring	$11,233.00	1.55%
246	Power and Control Wiring	$10,346.00	1.42%
24 Total		**$47,735.00**	**6.57%**
252	Air,Water and Steam Service Sys	$8,790.00	1.21%
253	Communication Equipment	$8,642.00	1.19%
25 Total		**$25,788.00**	**3.55%**
262	Mechanical Equipment	$16,899.00	2.33%
26 Total		**$16,899.00**	**2.33%**
	Total Direct Capital Costs	**$726,285.00**	

2.5 Development of the Economic Model

The main goal of this work is to find ways to reduce the overall cost of electricity for the given reference reactor plant design. The cost of electricity will have many components, and any design alternative that may affect one or more of these components. In Chapter 3 the process will be described through which an example economic model will be developed. Consistent with the reference design used in the safety model, the economic model is based on the ALMR.

The most important part in the development of the reference model is to determine the validity of the inputs that are being used. There has not been considerable experience in building Sodium Fast Reactors in the past, so it may be difficult to obtain reliable information. However, it will be possible to compare the cost of SFRs against the cost of a modern light water reactor using published information. Modern LWR costs will be easier to determine to assess the relative differences in SFR cost data.

2.6 Methodology Example

In Chapter 4, several case studies will be performed that will demonstrate the use of the methodology and the tools described in this chapter. In order to clearly demonstrate the methodology shown in Figure 2.1, a step-by-step breakdown is described below:

Step 1 – Identify Potential Area of Focus

Based on the economic data presented in Table 2.3, identify one of the costly systems, structures and components that should be targeted. This large cost driver may be an area where changes can result in potential economic savings.

Step 2 – Using Risk Model, Apply TNF to Determine Sequences of Interest for Safety Case

Using the PRA, identify that the sequences of interest were those that used this SSC to mitigate the damage states in case of an accident.

Step 3 – Determine Systems and Components that are used in these Sequences

Determine what specific systems, structures and component were used in the sequences that might be removed or modified.

Step 4 – Use Importance Measures or Other PRA Approaches to Identify Design Alternatives

Calculate the RAW and LEF values for the SSC to see if it plays a significant safety role and could be modified.

Step 5 – Confirm Design Alternative meets the TNF Standards using Risk-Based Methods

Set the component failure probability to 1.0 to assess impact. Confirm with safety analysis and other factors (overall plant reliability) to see if this change is warranted and if the TNF is satisfied.

Step 6 – Determine Economic Benefit and Gain Insights to Further Improve Economics

Calculate the economic gain from the proposed design modification and tabulate overall system savings for each proposed change.

Step 7 – Identify other Non-PRA Based Changes

Consider other improvements such as means to increase thermal efficiency (raising core outlet temperature, different power conversion systems, improvement in capacity factor, etc) that require design modifications which may have an impact on the safety of the plant. Apply steps 1-6 again to assess safety impact and economic value.

Chapter 3 – Development of the Economic Model

3.1 Overview

The economic model to be used in this framework was developed as a tool to determine the economic benefits of possible design alternatives. The first step in developing the economic model was to gather available information for the cost of the Sodium Fast Reactor. Due to the relative lack of experience in industry, the availability of accurate and reliable cost information was limited to cost estimates provided by developers. For the purpose of this analysis, the cost estimates of developers were assumed and not challenged. Even when corrected for inflation, these numbers appear to be low relative to light water rector cost estimates despite the obvious differences in complexity of design of the two types of plants. Four major cost areas were identified in the lifetime of an SFR: Capital Costs, Operations and Maintenance Costs, Fuel Costs and Decommissioning Costs. All available information in various levels of detail was gathered, as well as expert opinion on the expected cost for these major areas.

The most detailed data was available for the Advanced Liquid Metal Reactor (ALMR) which was selected for use in construction of the economic model. The categorized inputs were entered into the G4-ECONS model, producing the outputs in the form of the Levelized Unit Electricity Cost (LUEC) and the Total Capital Cost.

The data was then analyzed to confirm the validity of the model by comparing several applicable categories to similar information from an LWR model used in the Generation IV roadmap exercise [US DOE and GIF, 2002]. Adjustments were made to compensate for the inconsistencies between the two models. These inconsistencies were largely in the area of balance of plant systems and site infrastructure.

Using the economic model, sensitivity analyses were performed to determine the effects of changing the possible inputs. Five main areas of interest were identified: capital cost, operations and maintenance cost, fuel cost, overall efficiency, and capacity factor. Any changes to these areas would have a significant effect on the overall LUEC. The individual effects of reductions in each of these areas were then shown.

3.2 Gathered Data

3.2.1 Capital Cost

Available capital cost information was gathered for the Clinch River Breeder Reactor (CRBR) [U.S. DOE, 1983], Japanese Sodium Fast Reactor (JSFR) [Ono et al, 2007], S-PRISM [Boardman, 2001], and the Advanced Liquid Metal Reactor (ALMR) [Gokcek et al, 1995]. The cost data is generally presented as lumped figures at a relatively low level of detail. These figures can be used to develop a range of capital cost estimation available from industry. The results are presented in Table 3.1 corrected for inflation up to 2007 using the escalation factors presented in the G4-ECONS guidelines [GIF, 2007].

Table 3.1 - Summary of Collected Data: Capital Costs (2007$)

	Net Capacity (MWe)	Total Capital Cost ($M)	Total Capital Cost ($/kwe)
CRBR	380	5,032	13,242
JSFR	1500	2,445	1,630
S-PRISM*	1651	3,024	1,832
ALMR*	622	1,538	2,472

*S-PRISM is a modular reactors with multiple small reactors per plant

The Clinch River Breeder Reactor was a prototype design in the 1970's and was eventually abandoned due to the escalating costs. In contrast, the other three data points are based on more modern SFR designs and are all within the same relative range of capital cost.

Only the S-PRISM and the ALMR have more detailed cost breakdowns that can be presented in terms of the GIF Code of Accounts [GIF, 2007] to provide some comparison value. Due to the large difference in electrical generating capacity, the main item of comparison was the percent of total overnight capital cost as a fraction of the total overnight capital cost for each of the categories. These percentages nearly matched across all accounts, varying by no more than 3%. See Table 3.2 for details. This implies a certain level of reliability between these two data sets.

The most detailed cost information available was with the ALMR. In the "1994 Capital and Busbar Cost Estimate", complete detailed cost estimation is presented according to the GIF Code of Accounts (COA) [Gokcek et al, 1995], including breakdowns within each code. This level of detail will allow for the development of a fairly detailed reference economic model for the pool-type Sodium Fast Reactor.

Table 3.2 - Comparison of Percentage Capital Costs: ALMR and S-PRISM

COA	Categories	ALMR % of Total Cost	S-PRISM % of Total Cost	Diff. in % of Total Cost
20	Land and Land Rights	1%	0%	1%
21	Structures and Improvements	13%	10%	3%
22	Reactor Plant Equipment	38%	37%	0%
23+25	Turbine Plant and Heat Rejection	11%	11%	0%
24	Electric Plant Equipment	4%	5%	-1%
26	Misc. Plant Equipment	2%	2%	1%
33	Construction Services	8%	6%	2%
31	Engineering and home office	5%	3%	3%
35	Field supervision	5%	3%	1%
44	Owners Cost	14%	12%	2%

3.2.2 Operations and Maintenance Costs

Available O&M Costs were also gathered from SFR sources to determine reasonable estimates to use in the economic modeling of the reactor. The results are presented in Table 3.3. As with the capital cost, these numbers were provided in different levels of detail, ranging from a simple quote for mills/kw-hr to a more detailed breakdown into categories such as labor costs and consumables.

Table 3.3 - Summary of Collected Data: Operation & Maintenance Costs (2007$)

	O&M Costs (mills/kw-hr)
ALMR (622 MWe Plant)	11.92
IFR (1488 MWe Plant)	13.01
S-PRISM (1520 MWe Plant)	7.63
Nuclear Industry (NEI)	12.90

The most detailed set of O&M numbers provided is the 622 MWe ALMR [Gokcek et al, 1995]. Since the breakdown level of detail required for the G4ECONS model was available for the ALMR, these values will be used for the reference model. It was not possible to explain the large discrepancy between the S-PRISM and ALMR costs since the S-PRISM numbers were not available at a suitable level of detail. The ALMR quoted O&M cost is within an acceptable range when compared to the NEI average values for the industry [Nuclear Energy Institute, May 2008]. Whether this SFR value is correct is not challenged in this thesis.

3.2.3 Fuel Cycle Costs

To determine reasonable estimates for use in the economic model, Fuel Cycle Costs were collected from various estimates such as the S-PRISM [Boardman, 2000], IFR [Lineberry et al, 1986], the S-PRISM Fuel Cycle Study [Dubberly, 2003], and the ALMR [Gokcek et al, 1995]. Like the capital costs and O&M costs, these were provided at various levels of detail. The most useful were those that were provided in terms of mills/kW-hr, since this was the input required for the economic model. Unlike the previous categories, there is less concrete cost data on the fuel cycle, since at this point there are still competing options for reprocessing and types of fuel. These options are beyond the scope of this project, and will not be discussed further. The results are presented in Table 3.4. These varying SFR values are compared to the Reference LWR value from the GIF Model [GIF, 2008]. The Global Nuclear Energy Partnership (GNEP) reports were reviewed, but there was no economic data available that described the fuel cycle cost input to a power reactor.

Table 3.4 - Summary of Collected Data: Fuel Cycle Costs

	Fuel Cost (mills/kw-hr)
S-PRISM (1997$)	5.0
IFR (1986$)	7.0
S-PRISM Fuel Cycle Study (2003$)	4.6
ALMR (1994)	10.22
Reference LWR (2007$)	9.07

For consistency, the number that will be applied to the G4ECONS model will be the ALMR value from the "1994 Capital and Busbar Cost Estimate." This number has been supplied as the fuel cycle busbar cost for a Central Fuel Recycle Facility [Gokcek et al, 1995]. Again, the accuracy of this value is not challenged in this thesis.

3.3 Development of a Reference Model

The economic model should be based on a reasonably accurate reference model even if only used for a comparative analysis. But with little data available for the cost of a Sodium Fast Reactor, it becomes difficult to assess the validity of the collected values. The best tool that can be developed from the data is a model that is consistent with the available information. This model will be based closely on the ALMR, since the plant design and applicable features represent a basic pool-type Generation IV SFR.

Once the reference model has been developed and an estimated cost breakdown has been produced, the cost savings or excess expenditures for any modifications can be applied to these categories to judge the relative savings possible. In this manner, the overall effect on capital cost and busbar generation costs can be realized for each design modification that is proposed using the Technology Neutral Framework.

According to the G4 Estimating Guidelines [GIF, 2007], top-down cost estimation can be sufficient to approximately estimate costs when there is little cost data available. As long as consistent estimating techniques are followed, comparisons can be made between design alternatives. This basic model can also be modified and refined as more information becomes available.

3.3.1 Using the G4ECONS model

The G4-ECONS model is an economic modeling tool that was developed by the GIF/Economic Modeling Working Group as a tool for Generation IV reactor cost modeling [GIF, 2008]. The model is designed to utilize the user input results from a top-down or bottom-up estimation process of the capital, fuel cycle, and operational costs. These inputs are processed based on the reactor specifications and the desired reactor application (electricity, process heat, etc.), and levelized output costs are displayed. Based on the level of detail of the inputs to the model, many different cost breakdowns within the calculations are available, such as within the fuel cycle or financing models. The user can make use of these features if enough detail is available. If less detailed information is available, lumped values can be input for the O&M, Fuel Cycle, or Capital Cost categories by applying the lumped figures to the "Contingency" category.

The first few generic data inputs are designed to describe reactor performance and the economic climate which will influence financing and most importantly the Levelized Unit Electricity Cost (LUEC). Reactor inputs include total thermal power, thermal efficiency, capacity factor, and years to construct. These shall be provided by the designer of the reactor. Several of these generic data inputs are not used, depending on the level of detail for the specific reactor design. For example for the ALMR, the site size and cost of land was not required because the total cost of land was already provided in another section. Economic data inputs include cost of capital, the economic life of the reactor, and the time to construct. These shall be determined by the builder as best guess estimates based on current conditions. The financial inputs used in the reference model are based on the Nuclear Energy Institute's 2008 predictions shown in Table 3.5 [Nuclear Energy Institute, August 2008]. A notable financial input missing in the G4-ECONS model is the applicable tax rate. Therefore, all monetary outputs in this thesis will be provided without the effects of taxes.

Table 3.5 - Generic Inputs for G4-ECONS model*

Reactor Plant Description	2-block ALMR
Year Adjust	1
Hours in a Day	24
Days in a Year	365
Site Size (Acres)	n/a
Site Size (Hectares)	n/a
Reactor Net Electrical Capacity	1244
Reactor Average Capacity Factor over Life	85%
Thermodynamic Efficiency (net)	37%
Plant Economic and Operational Life	40 years
Years to Construct	6 years
Cost per Acre for land	n/a
Average craft labor rate	n/a
Cost of Capital	11.04%

*Based on the ALMR as described in Table 3.6

For the Sodium Fast Reactor, the desired application is electricity production, and therefore the main economic concerns are the total capital cost and the electricity cost once operational. The capital costs are expressed as $/kwe and the LUEC is expressed as mills/kw-hr. The most useful part of the G4-ECONS model is that these outputs are quickly produced as the inputs are changed. Therefore, depending on the level of input detail, quick sensitivity analyses can be performed by adjusting the input parameters.

3.3.2 Levelized Unit Electricity Cost

The ultimate goal of economizing the Sodium Fast Reactor is to reduce the cost of producing electricity to levels comparable with light water reactors. By achieving similar levels of electricity production cost, the SFR may be a desirable choice by utilities for the next generation of nuclear power reactors. Using the G4-ECONS model, the LUEC is made up of four parts:

1. Annualized Capital Cost, including financing
2. Operations and Maintenance
3. Fuel Cycle
4. Decommissioning and Decontamination

The annual Fuel Cycle and Operation and Maintenance inputs to the LUEC are straight-forward: the total annualized cost divided by the total electricity generated in one year.

The Decommissioning and Decontamination input to the LUEC is an annual contribution to a sinking fund, with a goal amount to be accumulated over the operating life of the plant. This goal amount is given as 33% of the direct capital costs [GIF, 2008]. The total annual amount is then divided by the total electricity production for one year. For the ALMR model with a direct capital cost of $1906 million, this will be $635 million dollars at the end of life.

The Annualized Capital Cost is the annual payment that must be made against the initial construction loans. This figure includes the initial overnight capital costs, the interest during construction, and the interest accrued over the operational period. This annual figure is divided by the total electricity production for one year. The total electricity production is calculated using the thermal power, efficiency and capacity factors given for the reactor [GIF, 2007]. These values are calculated by the G4-ECONS model using the cost inputs provided by the user.

The values of each of these will be determined for the SFR and will be shown in Table 3.8 later in this chapter.

3.3.3 Description of the ALMR

In order to provide a context for the cost figures that follow, the ALMR reference reactor is described below. The ALMR is an SFR design developed by General Electric based on the Integral Fast Reactor Technology. An ALMR plant utilizes modular reactor modules arranged into reactor blocks, each comprised of two 840 MWt pool-type sodium fast reactors producing 622 MWe [Gluekler, 1997]. The performance data is given in Table 3.6. A basic diagram of the nuclear island for the reference pool-type SFR model is shown in Figure 3.1 [Gokcek et al, 1995]. Each block of the plant would have two reactors and steam generator sets such as these.

Table 3.6 - ALMR Performance Data

No. of Reactors/Power Block	2
No. of Power Blocks	2
Thermal Reactor Power	3360 MWt
Electrical Power for 2 Power Blocks	1244 MWe
Net Station Efficiency	37%
Plant Capacity Factor	85%
Steam Conditions (Superheat)	15.16 MPa/430 C
Primary Sodium Inlet/Outlet Temp	360/500 C
Secondary Sodium Inlet/Outlet Temp	327/477 C
Fuel	Metal (U-0.23Pu-0.1Zr)
Average Fuel Burnup	106 MWd/kg
Average Fuel Linear Power, BOL/EOL	20/18 W/mm
Refueling Interval	24 months
Containment Leak Rate	<1% (7 kPa, 20 C)

Figure 3.1 – Pool-Type SFR Nuclear Island Diagram

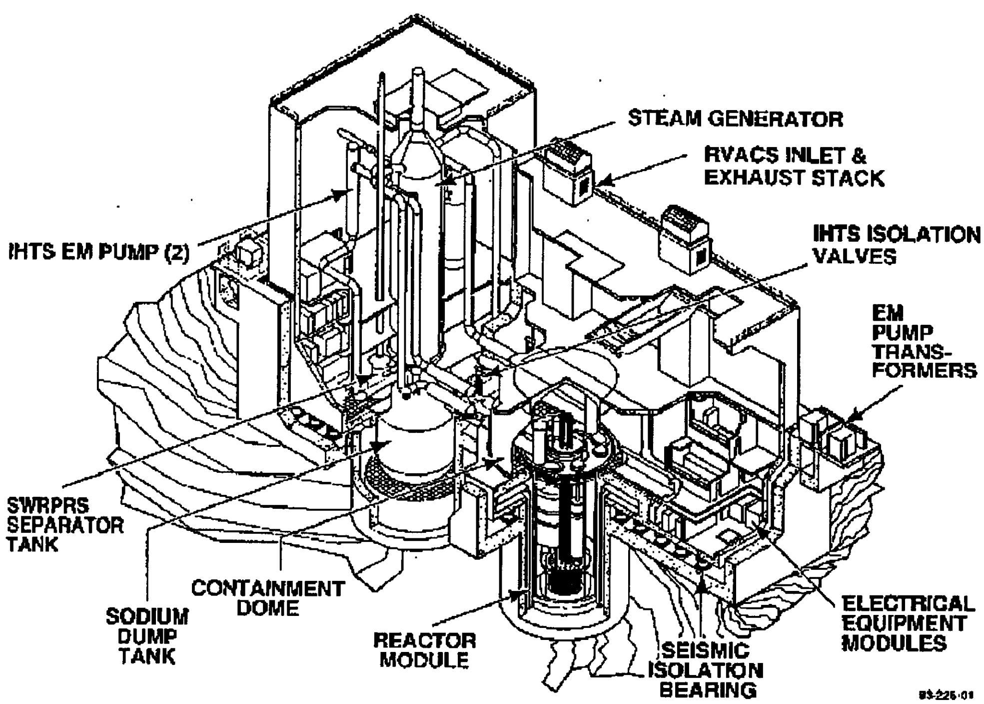


Figure 3-3 REACTOR & STEAM GENERATOR FACILITY GENERAL ARRANGEMENT

3.4 Results from Reference Model

3.4.1 LUEC using the ALMR estimated costs

The information presented in the "1994 Capital and Busbar Cost Estimate" for an Nth-of-a-kind (NOAK) ALMR were input to the G4-ECONS model [Gokcek et al, 1995]. These high level inputs on the Code of Account level are shown in Table 3.7. More detailed data was used in the actual model (see Appendix A). With this information and the G4-ECONS model, the results in Table 3.8 are obtained. As stated earlier, these values do not take into account the effects of taxes.

Table 3.7 - ALMR Inputs to G4-ECONS (2007$)

COA	Category Description	($M)
	Reactor Net Electrical Capacity	1,244 MWe
	Reactor Average Capacity Factor over Life	85%
	Thermodynamic Efficiency (net)	37%
	Plant Economic and Operational Life	40 years
11	Land and land rights	23.855
21	Buildings, Structures, & Improvements on Site	358.80
22	Reactor Plant equipment	908.69
23	Turbine/Generator Plant equipment	241.94
24	Electrical equipment	130.28
25	Water intake and heat rejection plant	39.50
26	Miscellaneous plant equipment	53.22
31	Design Services at A/E Offices	99.10
33	Design services at plant site	121.32
35	Construction supervision at plant site	218.78
	TOTAL FOR SERVICES (31-35)	**439.20**
46	Other Owners' capital investment costs	318.69
	Contingency value in $M	384.38
	OVERNIGHT TOTAL	**2898.55**

Table 3.8 - Unadjusted LUEC Results using ALMR numbers

	Total Annual Cost (1994M$/yr)	Mills/kw-hr (1994$)	Mills/kw-hr (2007$)*
Capital Cost w/ financing**	70.07	35.47	51.21
Operations and Maintenance	46.60	7.82	11.82
Fuel Cycle	49.25	10.63	15.14
D&D	1.68	0.36	0.47
TOTAL	**167.60**	**54.28**	**78.64**

*Scaled using a factors from the Cost Estimating Guidelines: Capital Cost – 1.444, O&M – 1.512, Fuel – 1.425, D&D – 1.305

**Financing costs using inputs: cost of capital – 11.04%, economic life – 40 years, construction – 6 years

3.4.2 Capital cost of Sodium Fast Reactors compared to modern LWR

If Sodium-Cooled Fast Reactors are to be introduced in the United States, their cost of electricity must be equal or lower than comparable alternatives. In this case, these alternatives would be light water reactors. There have been recent estimates for the cost of LWRs that can be used for comparison, including five estimates evaluated in the 2008 MIT study, "Update on the Cost of Nuclear Power" shown on Table 3.9 below [Du and Parsons, 2008]. As can be seen, the overnight capital cost of new proposed light water reactors ranges from a low of $3500/kwe to over $4700/kwe in 2007$. These numbers offer significant doubts in the estimates of SFR costs presented by developers.

Table 3.9 - Overnight Costs for Some Proposed Nuclear Plants [Du and Parsons, 2008]

	Owner [A]	Name of Plant [B]	Design [C]	Capacity MW [D]	Projected Commercial Operation Date [E]	Overnight Cost US 2007 $/kW [F]
[2]	FPL	Turkey Point 5 & 6	ESBWR	3,040	2018-2020	3,530
[3]	Progress Energy	Levy County 1 & 2	AP1000	2,212	2016-2017	4,206
[4]	SCEG/Santee-Cooper	V.C. Summer 2 & 3	AP1000	2,234	2016-2019	3,787
[5]	Southern	Plant Vogtle 2 units	AP1000	2,200	2016-2017	4,745
[6]	NRG	South Texas 3 & 4	ABWR	2,700	2014-2015	3,480

From Du and Parsons, CEEPR Working Paper 09-004.

There is much debate over the capital cost of a Sodium Fast Reactor relative to the present-day cost of a light water reactor. Table 3.10 below shows some dated comparative studies on the relative costs of SFRs and LWRs. Most predictions indicate that total capital cost for an SFR plant is greater than that for a similar LWR plant.

Table 3.10 - Relative Cost by Country

Country	**Unit Capital Cost Relative to LWR**
France[2]	1.26
Great Britain[3]	1.1
Russia[4]	1.3
Germany[3]	1.16

3.4.3 Capital cost of ALMR compared to modern LWR

To provide a relative means of comparison of LWRs and SFRs, the G4-ECONS model provides a benchmark LWR model described as a Gen III+ PWR, with a rated electrical capacity of 1300 kwe. This benchmark model has a complete set of cost data, broken down to the COA level of detail [GIF, 2008]. A more modern source of LWR capital cost data is the 2005 TVA Cost/Schedule/COL Project Proposal for its Bellefonte Site. The proposed project at Bellefonte is a two-unit addition, 1,371 kwe per unit [Toshiba, 2005].

For ease of comparison, the Nth-of-a-Kind cost data for a 2-block ALMR was selected from the ALMR report, and then costs were scaled up from 1994$ to 2007$ using the factors of inflation provided in the G4 Guidelines. The Bellefonte costs were halved to simulate a 1-unit model, and these costs were scaled from 2004$ to 2007$. In this way, all reactor models in the comparison were rated for about 1300 kwe.

The first comparison is between the benchmark LWR provided in the G4ECONS model and the Bellefonte estimate, shown in Table 3.11. This was to confirm the reliability of the benchmark LWR data provided in the G4-ECONS model. Since the Bellefonte units are proposed as

[2] M. Rapin. *Fast Breeder Reactor Economics*. Royal Society Discussion Meeting, London, UK. 1989
[3] Troyanov, M.F. et al. "In Current Conditions, is it more expensive to build fast reactors than thermal reactors?" *Atomic Energy*. Vol. 78, No. 1 1995
[4] Poplavskii, et al. "BN-800 as a New Stage in the Development of Fast Sodium-Cooled Reactors." *Atomic Energy*. Vol. 96, No. 6 2004

additional reactors on an existing site, certain categories will be lower, such as Land and Land Rights (COA 11), Buildings, Structures and Improvements (COA 21) and Water Intake and Heat Rejection Plant (COA 25). Many of these facilities already exist and will only require additions made to accommodate the new units. Thus these areas should be disregarded in the comparison.

This table shows that the G4-ECONS standard model for LWRs is quite acceptable compared to current cost estimates. In all other areas, with the exception of the Turbine Equipment, the costs are similar, with the Bellefonte Plant costs exceeding by 10-20%. This difference is most likely due to an increase in costs at a pace exceeding inflation, reflected by the higher cost of the Bellefonte estimate. These similar categories are bolded in Table 3.11. The consistency of these overnight capital cost results bolsters the reliability of the cost estimation methods used by the GIF modeling group.

Table 3.11 - Comparison of Benchmark LWR and Bellefonte LWR (2007$)

COA	Case Description	Reference LWR in G4ECONS	Bellefonte 1-unit
	Reactor Net Electrical Capacity	1,300	1,371
	Reactor Average Capacity Factor over Life	90%	90%
	Thermodynamic Efficiency (net)	33%	33%
	Plant Economic and Operational Life	40	40
11	Land and land rights	6.000	0
21	Buildings, Structures, & Improvements on Site	440.18	294.84
22	**Reactor Plant equipment**	**454.09**	**498.23**
23	Turbine/Generator Plant equipment	430.82	335.37
24	**Electrical equipment**	**125.58**	**151.28**
25	Water intake and heat rejection plant	91.39	21.00
26	**Miscellaneous plant equipment**	**82.42**	**79.17**
31-35	**Total Construction Services**	**615.29**	**592.21**
46	**Other Owners' capital investment costs**	**312.65**	**375.26**
	Contingency value in $M	382.85	536.07
	OVERNIGHT TOTAL	2941.270	2873.43

In order to determine the relative validity of the ALMR capital costs, a direct comparison can now be performed with some confidence. For a simplified explanation, only the comparison between the benchmark PWR and the ALMR model will be discussed. Most capital costs are difficult to directly compare without a much more detailed cost breakdown, since these costs are very dependent on the reactor design. The design of an SFR is very different from that of a PWR, so a comparison of these design specific costs would yield no useful insight into their relative accuracy.

However, two main categories in the COA breakdown should be design independent: Electrical Equipment (COA 24) and Other Owner's Capital Investment Cost (COA 46). The Electrical Equipment category includes all switchgear, transformers, protective systems, and other on-site electrical necessities. These costs should be solely dependent on the electrical capacity of the facility, thus independent of reactor design. The Other Owner's Capital Investment Cost includes the general infrastructure required such as roads, railways, administrative buildings, transmission lines and other costs that are non-reactor specific [Nuclear Energy Institute, August 2008]. These should also be independent of reactor design. The direct comparison of capital costs is presented in Table 3.12. In comparison, these two categories are consistent between the ALMR and the benchmark LWR; both values are within 2% of each other. This supports the consistency of estimation methods used by both economic modeling groups, indicating that there may be relative accuracy between the two sets of data.

Table 3.12 - Direct Comparison of Capital Costs for LWR and SFR (2007$)

COA	Case Description	Reference LWR	2-Block ALMR
	Reactor Net Electrical Capacity (MWe)	1,300	1,288
	Reactor Average Capacity Factor over Life	90%	85%
	Thermodynamic Efficiency (net)	33%	37%
	Plant Economic and Operational Life (years)	40	40
11	Land and land rights	6.000	23.855
21	Buildings, Structures, & Improvements on Site	440.18	358.80
22	Reactor Plant equipment	454.09	908.69
23	Turbine/Generator Plant equipment	430.82	241.94
24	Electrical equipment	125.58	130.28
25	Water intake and heat rejection plant	91.39	39.50
26	Miscellaneous plant equipment	82.42	53.22
31	Design Services at A/E Offices	96.59	99.10
33	Design services at plant site	139.88	121.32
35	Construction supervision at plant site	378.82	218.78
	TOTAL FOR SERVICES (31-35)	**615.29**	**439.20**
46	Other Owners' capital investment costs	312.65	318.69
	Contingency value in $M	382.85	384.38
	OVERNIGHT TOTAL	**2941.27**	**2898.55**
	Total Capital Cost (with financing) $/kwe	**3215.43**	**3311.43**

*Financing costs using inputs: cost of capital – 11.04%, economic life – 40 years, construction – 6 years

In comparing the other components of the total capital cost, there are large differences in the accounts for Buildings, Structures & Improvements (COA 21), Reactor Plant Equipment (COA 22) and Construction Supervision at Plant Site (COA 35). The Reactor Plant cost for the ALMR is double that of the LWR. The reactor plant and internals of an SFR are much more complex than an LWR, so this premium is expected. Four areas that have been identified to drive this cost premium are the complexity of the automatic control systems, the number of heat-transfer loops required to allow for passive safety measures, the need for an intermediate loop to prevent a primary sodium and water reaction, and the use of sodium instead of water [Troyanov et al, 1995].

There are several categories of the LWR that are more expensive than the SFR. The Buildings and Structures cost for the LWR exceeds the SFR by almost 25%. This can be attributed to the cost of the large, concrete and steel containment required for an LWR that is not included in the design for an ALMR. Finally, the Construction Supervision at Plant Site costs for the LWR are almost 75% more than those for the SFR. This is because of the modular, in-factory construction of the ALMR units that leads to a much simpler construction process on-site [Boardman, 2001]. An LWR has the majority of construction occur on-site.

The only major discrepancy among the three different sets of data is the Turbine/Generator Plant Equipment (COA 23). This category should be relatively similar from one design to the next, since the Rankine cycle remains the same regardless of the core design providing the heat. The other categories that make up direct capital cost, Heat Rejection and Miscellaneous Equipment (COA 25 and 26), should also be relatively similar between the two designs. It is not clear why the ALMR has assumed such a low value for the turbine generator compared to the LWR. Therefore, for the sake of consistency in the economic model, the ALMR values were replaced with those of the reference PWR in the bolded accounts shown in Table 3.13. This raised the overnight total to a value of $3140.84 M, which resulted in a total capital cost of $3460/kwe. This value accounts for the financing costs, which is determined using the user inputs: 11% cost of capital, 40 years economic life and 6 years of construction.

*Table 3.13 - Modified Comparison of Capital Costs (2007$)**

COA	Case Description	Reference LWR	2-Block ALMR
	Reactor Net Electrical Capacity (MWe)	1,300	1,288
	Reactor Average Capacity Factor over Life	90%	85%
	Thermodynamic Efficiency (net)	33%	37%
	Plant Economic and Operational Life (years)	40	40
11	**Land and land rights**	**6.000**	**6.000**
21	Buildings, Structures, & Improvements on Site	440.18	358.80
22	Reactor Plant equipment	454.09	908.69
23	**Turbine/Generator Plant equipment**	**430.82**	**430.82**
24	**Electrical equipment**	**125.58**	**125.58**
25	**Water intake and heat rejection plant**	**91.39**	**91.39**
26	**Miscellaneous plant equipment**	**82.42**	**82.42**
31	Design Services at A/E Offices	96.59	99.10
33	Design services at plant site	139.88	121.32
35	Construction supervision at plant site	378.82	218.78
46	**Other Owners' capital investment costs**	**312.65**	**312.65**
	Contingency value in $M	**382.85**	**382.85**
	OVERNIGHT TOTAL	**2941.27**	**3140.84**
	Total Capital Cost (with financing) $/kWe	**3215.43**	**3459.86**

*Bolded Categories were equalized for consistency
**Financing costs using inputs: cost of capital – 11.04%, economic life – 40 years, construction – 6 years

When these estimates are compared to the overnight costs from the MIT Study shown in Table 3.9, the following summary in Table 3.14 is presented for evaluation:

Table 3.14 - MIT Study LWR Cost Estimates (2007$)

Owner	Plant	Capital Cost $/kwe
TVA	Bellefonte	2,930
FPL	Turkey Point	3,530
Progress Energy	Levy County	4,206
SCEG	VC Summer	3,787
Southern	Plant Vogtle	4,745
Sodium Fast Reactor	ALMR	3,140

It is quite clear that despite the additional complexities of the sodium cooled fast reactor, developers believe that the capital cost could be lower than the level of modern LWRs. The MIT updated economic study determined that the expected future cost of LWRs should be about

$4000/kwe. These results are shown on the Table 3.15 below which compares other electricity generating alternatives with and without a carbon charge and variations in cost of capital [Du and Parsons, 2008].

Table 3.15 - Expected Future Cost by Fuel Type [Du and Parsons, 2008]

	Update $2007				
			LCOE		
	Overnight Cost	Fuel Cost	Base Case	w/ Carbon Charge $25/tCO2	w/ same cost of capital
	$/kW	$/mmBtu	¢/kWh	¢/kWh	¢/kWh
Nuclear	4,000	0.67	8.4		6.6
Coal	2,400	2.60	6.2	8.3	
Gas	900	7.00	6.5	7.4	

From Du and Parsons, CEEPR Working Paper 09-004.

3.4.4 Adjusted Total Capital Cost and LUEC

Based on the results of the MIT study, the two sets of capital costs were scaled up by a factor of 1.244 so that the LWR capital costs with financing would be equal to $4000/kwe. This same scaling factor increased the ALMR capital cost to 4304 $/kwe. Using these scaled figures, the adjusted LUEC increased due to the increased capital cost, to 75.56 mills/kw-hr for the benchmark LWR and 92.34 mills/kw-hr for the SFR. The results of this escalation are shown in Table 3.16.

Table 3.16 - Adjusted LUEC for LWR and ALMR compared to NERA Proportions of Generating Cost (2007$)

	LWR Mills/kw-hr	ALMR Mills/kw-hr	NERA Proportions of Generating Cost
Capital Cost w/ financing	56.87 (75.3%)	64.79 (70.2%)	60-75%
Operations and Maintenance	9.35 (12.4%)	11.93 (12.9%)	8-15%
Fuel Cycle*	9.07 (12.0%)	15.15 (16.4%)	5-10%
D&D	0.28 (0.4%)	0.47 (0.5%)	0-5%
Total LUEC	**75.56**	**92.34**	
Total Capital Cost	**$4000/kwe**	**$4304/kwe**	

* LWR Fuel Cycle is based on once-through cycle, ALMR fuel cycle is based on steady-state recycling facility

Within the NEI's document entitled, "The Cost of New Generating Capacity in Perspective", current nuclear power generating costs are in the range of 83.40 mills/kw-hr [Nuclear Energy Institute, 2009]. The results from the G4-ECONS model analysis for LWRs fall within this range. Also, each of the values as a percentage of total electricity cost nearly falls within the ranges expected by the NERA Economic Consulting Group's "Proportions of Generating Cost" [NERA, 2006]. This agreement further enforces the level of comparable reliability in these numbers.

These results show that the capital cost of the ALMR versus the LWR is not the only contributor that results in higher generation costs. Instead, the ALMR exhibits a higher cost in each of the four categories that make up the LUEC. The net effect is that given these assumptions, the levelized cost of power is 22% more expensive than LWRs with 14% higher cost for the capital portion of the plant.

3.5 Limitations of a Reference Model

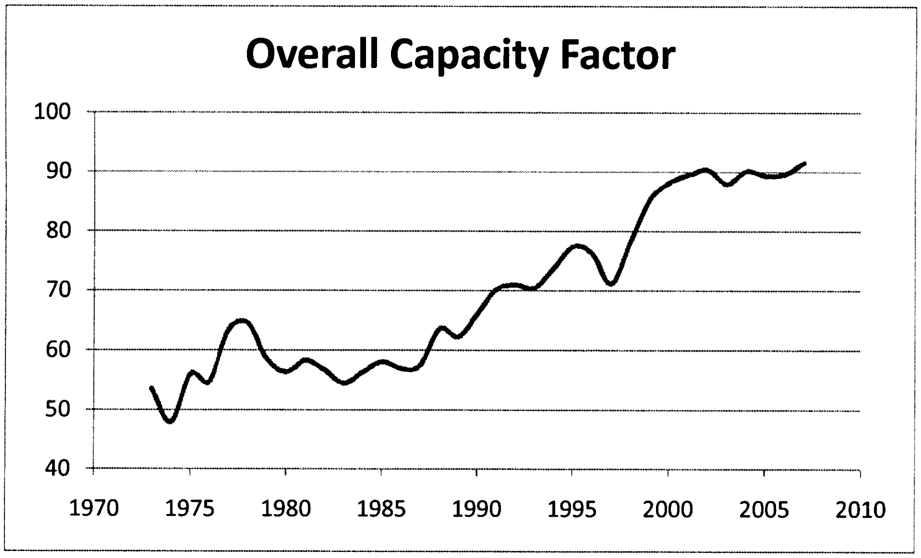
While the use of a reference model can enable an analyst to draw conclusions and determine economic trends, the reference model has limitations that must also be realized. The biggest limitation that must be remembered is that the results produced are only as reliable as the data that is input into the model. Care must be taken to obtain as reliable values as possible for the input data, or it must be noted that the results can only be reliable to such a level. This reference model uses the capital cost values for the ALMR, scaled from 1994$ to 2007$. The values have also been scaled to closely mimic the known increases in LWR costs over that same period. Although several checks have been performed to confirm the reliability of these capital costs, it must be acknowledged that absolute conclusions should not be made based on these values. However, this model should be a satisfactory tool to make comparative judgments based on possible design changes to reactors of similar design.

There are other variables besides capital cost that are difficult to predict at the incipience of development of this new technology:

- Fuel Cycle Information at Steady State
- Capacity Factor for Nth of a Kind (NOAK) Reactor
- Risk-Premium and Corresponding Cost of Capital

These variables will have a large impact on the LUEC, since these are major contributors to the calculations. Based on initial estimates, the fuel cycle component can make up anywhere from 10-20% of the overall levelized electricity cost [Gokcek et al, 2005]. The deployment of SFR technology would require the development of new fuel cycle facilities, and the estimates of these costs have very large uncertainties at this point. The capacity factor is also a large unknown. The assumption of 85% is used in developing the reference model may be too low for an advanced SFR which would significantly alter the economics. For example, a 5% increase in capacity factor lowers the cost of electricity by almost 6%. Modern-day light water reactors have achieved capacity factors of over 90%, but it had taken many years to get to this level of performance [Energy Information Administration, 2008]. See Figure 3.2 below for the capacity factor of US Nuclear Power Plants from 1973-2007.

Figure 3.2 - US Nuclear Power Plant Capacity Factor (1973-2007)



It is unknown whether or not Sodium Fast Reactors will require a similar learning curve to achieve capacity factors at levels as high as LWRs. If lessons learned from the LWR program can be applied to speed the development of the SFR program, these levels may be achieved more quickly. A sensitivity of the LUEC versus Capacity Factor is shown in Table 3.17.

Table 3.17 - LUEC with various Capacity Factors

Capacity Factor	LUEC (mills/kw-hr)
50 %	156.98
60 %	130.81
70 %	112.13
80 %	98.11
85 %	92.34
90 %	87.21

As can be seen in the table, even if a competitive level of electricity generation is reached for an assumed capacity factor, any loss of capacity will quickly balloon those costs to much higher levels. However, it is believed that the prior issues that have plagued SFRs, such as sodium leaks and low reliabilities, will no longer be a problem with the advancements in these technologies that have occurred. Whether this can be realized is not known at this time.

The cost of capital is another area that will be difficult to determine without a serious effort and detailed understanding of the current economic conditions. This expected rate of return to investors will require a risk premium, and this premium will vary depending on the perceived risk of their investment. Considering an NOAK reactor, the cost of capital for an SFR should be similar to that of an LWR. The financing structure will also be heavily dependent on who is constructing the plant, whether it is built as part of a regulated entity or as an unregulated plant [Nuclear Energy Institute, February 2009].

3.6 Uses of the Reference Model to Make More Economical

3.6.1 Possible ways to reduce LUEC

Based on the results from the G4-ECONS model, there are five possible ways to reduce the LUEC, thus making the Sodium Fast Reactor more economically competitive.

1. Reduce Capital Cost
2. Reduce Operations and Maintenance Costs
3. Reduce Fuel Cycle Costs
4. Increase Capacity Factor
5. Increase Efficiency

Utilizing these five possible changes, each affects the LUEC in a different way due to their importance in the calculation. The first three changes affect the inputs to the annual cost, with the capital cost input as the largest contribution. The final two changes will affect the amount of electricity produced annually, with higher energy production leading to lower unit prices. Therefore, as an example of their effects on the LUEC, Table 3.18 shows the required change to lower the LUEC of the ALMR by 5%.

Table 3.18 - Changes Required to Lower LUEC 5%

Capital Cost	Reduce by 7.4%
O&M	Reduce by 37.8%
Fuel Cycle	Reduce by 32.9%
Capacity Factor	Increase by 4.5% to 89.5%
Efficiency	Increase by 2% to 39%

As these results show, the most effective ways to reduce the LUEC cost for the SFR is to target the capital cost, the efficiency or the capacity. Any reduction in fuel cycle cost or O&M cost is less effective in reducing the overall generation cost, as major savings must be realized to result in similar reductions.

In order to quantify the economic impact of any change proposed, the effects of such a change on any of these five categories must be determined. Using the G4-ECONS model with the detailed inputs for the ALMR, the overall effect on the LUEC and Total Capital Cost can be realized quickly.

3.7 Summary

With the G4-ECONS model, the reference model was developed for the Sodium-Cooled Fast Reactor. These values were compared to several modern cost estimates of LWRs to confirm relative reliability of the costs. A relative comparison of the costs of the SFR and the benchmark LWR was then developed, as shown in Table 3.19. Using the model, possible methods of reducing the cost of electricity were identified, with reducing capital cost and increasing efficiency being two of the most effective options.

Table 3.19 - Summary of Economic Comparison (2007$)

	Benchmark LWR	SFR
Total Capital Cost	$4,000/kwe	$4304/kwe
LUEC	75.56 mills/kw-hr	92.34 mills/kw-hr
SFR relative to LWR	–	+ 22.2%

Chapter 4 – Demonstration of Methodology through Case Studies

4.1 Case Studies to Demonstrate Methodology

Several case studies were performed using the methodology described in an effort to reduce the cost of the Sodium Fast Reactor. Due to the limitations of available data, these case studies were selected to show the potential of this methodology in both probabilistic and deterministic situations, using the ALMR as the reference design to evaluate certain design alternative. The case studies considered are:

1. Removing energetic core disruptive accidents from licensing basis events
   a. Removal or simplification of structures and equipment
   b. Raising the core outlet temperature to increase thermal efficiency
2. Evaluation of alternative power conversion cycles
3. Assessing whether the steam generator can be manufactured as non-safety grade to reduce its cost
4. Eliminating unnecessary safety equipment on a risk basis such as the excess number of control rods

In this chapter, the economic benefit of each of the design alternatives will be presented in terms of savings in the cost of electricity (LUEC). The assumptions for the calculation of the LUEC are the same as those developed in Chapter 3, and will be based on the ALMR economic data presented in Appendix A.

4.2 Case Study 1 – Removing Energetic Core Disruptive Accidents from Licensing Basis Events

One major class of accidents covered within the PRA is the Energetic Core Disruptive Accident (ECDA). This class describes all accidents that result in the energetic removal of fuel material from its designated position in the core. These accidents have been a major issue in previous licensing discussions, as there has been serious concern over the possible consequences. These accident types do have the potential for serious consequences if they were to occur, but it can be shown that they are also extremely unlikely when analyzed using a risk-based approach to safety.

As previously described in Chapter 2, the TNF describes the process for determining Probabilistic Licensing Basis Events (LBEs) and the licensing requirements. The TNF also describes the method through which sequences can be screened from consideration. If the mean frequency of the sequence is lower than 10^{-8} per year, the event can be screened from consideration. This screening method is the major difference between the deterministic and probabilistic approach to licensing within the TNF, and it may allow for major savings in the SFR design.

4.2.1 Possible Dose from ECDA

If the ECDA cannot be screened below the threshold, it must comply by the standards for LBEs. Therefore, if the frequency does not fall below 10^{-8}, the dose resulting from this event must be less than the corresponding limit imposed by the stairstep curve in Figure 2.2. From this curve, the highest limit for dose corresponds with events having frequencies as low as 10^{-7}, which is 500 rem. Therefore, if the dose is higher than 500 rem, it cannot be licensable unless the point estimate of the frequency is pushed below the 10^{-8} level to be screened from consideration.

Using Generic Accident Progression methods based on several SFR PRAs and collections of event trees, the Generic Release categories were developed to group potential accidents by their potential consequences. This method is extremely valuable for the evaluation of accidents using the TNF as a guide, since it can immediately illuminate which sequences will have large potential consequences. The 12 release categories are described and the potential doses listed in Table 4.1 [Denning, 2009].

Table 4.1 - Generic Release Categories for ALMR

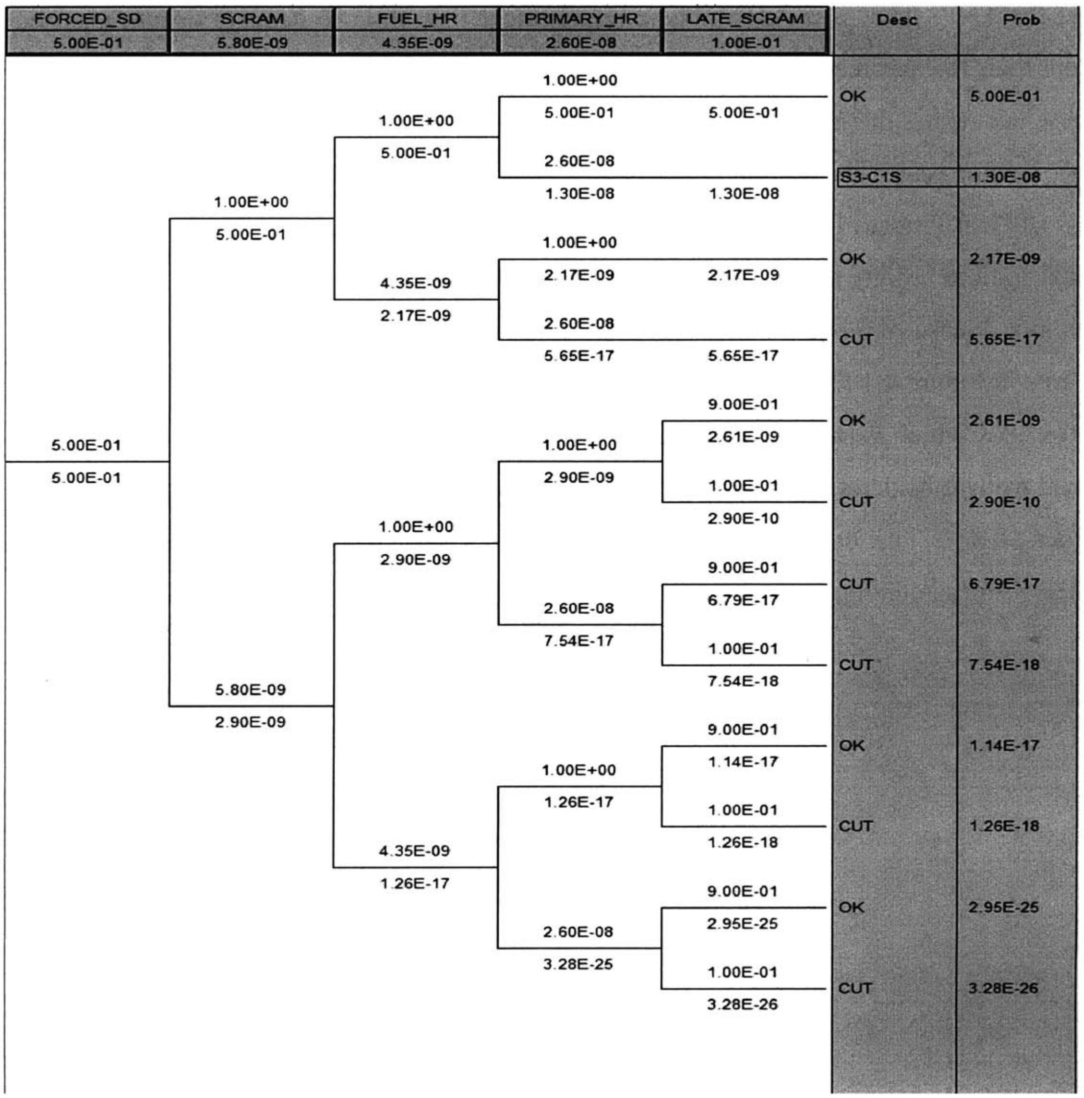
Release Category	Release Characteristics	Dose (rem) at One Mile		
		5^{th}	Median	95^{th}
1	Large Release of contaminated sodium to containment. Containment intact other than design leakage	2.5E-4	2.5E-2	0.25
2	Small release of contaminated sodium to containment. Containment intact other than design leakage	2.5E-6	2.5E-4	2.5E-3
3	Large release of contaminated sodium to containment. Containment open	4.3E-2	4.3	430
4	Small release of contaminated sodium to containment. Containment open	4.3E-4	4.3E-2	0.43
5	Substantial fuel melting. Pool scrubbing. Primary system intact or containment intact	0.2	1.3	2.3
6	Minor fuel melting. Pool scrubbing. Primary system intact or containment intact	2.0E-2	0.13	0.23
7	Substantial fuel melting. Pool scrubbing. Primary system and containment failed	77	47	810
8	Minor fuel melting. Pool scrubbing. Primary system and containment failed	7.7	47	81
9	Fuel melting. Energetic event. Limited pool scrubbing. Primary system failed. Containment intact	2.7	12	100
10	Fuel melting. Energetic event. Limited pool scrubbing. Primary system failed. Early containment failure	790	2,000	11,000
11	Failed primary system. Core uncover. Oxidizing environment. Containment intact.	1.4	11	110
12	Failed primary system (includes failure of vessel and guard vessel by molten fuel). Core uncover. Oxidizing environment. Containment failed.	160	1,100	11,000

The important value to note in this table is the 95^{th} percentile dose at one mile. This dose must be less than 500 rem unless the mean estimate of the frequency of the corresponding event can be pushed below the 10^{-7} threshold. Therefore, the release categories of interest for this analysis are #7, 10 and 12. According to their descriptions within the table, each of these events represent situations where there is core damage and the containment is assumed to have failed.

4.2.2 Establishing the Frequency of ECDA

Within the TNF, there is a deterministic requirement that all core damage events must be less frequent than 10^{-5} per reactor year. However, in order to gain economic benefit by not designing for these accidents, the mean estimate of this frequency must be shown to be less frequent than 10^{-7} per reactor year. The dominant sequence that results in the energetic scenario from the PRA is Loss of Heat Removal during a shutdown transient. The progression of this scenario leads to eventual sodium boiling and a recriticality of the core with a late energetic expulsion. This event results in a release of 11,000 rem for the 95th percentile release. This sequence is shown in the event tree in Figure 4.1 [Johnson, 2009]. However, the point estimate frequency of this sequence is 1.30 x 10^{-8}, which is below the threshold for screening. In addition, all unprotected events are screened from consideration due to the highly reliable scram system, with a failure probability on the order of 10^{-6}. The level of assurance on the scram system helps to push all related sequences well below the screening limit [Apostolakis and Johnson, 2009].

Figure 4.1- Event Tree Showing Dominant Sequence



4.2.3 Economic Benefit of Not Designing for ECDA

By not considering the ECDA events as the basis of the design requirements, several simplifications can be made and justified with the TNF approach to safety. To determine the economic gain from eliminating the ECDA as a design basis accident, the ABR-1000 has been designed to meet the requirements in the existing licensing framework, codified in 10 CFR Part 50, which tends to use the more deterministic method of identifying design basis accidents. These accident scenarios have required additional redundant safety features [Grandy and

Seidensticker, 2007]. The ABR-1000 has three redundant decay heat removal loops and two independent safety-grade scram systems in addition to an ultimate scram system included in its design. It also has a much larger and more robust reactor containment building. These additional systems and structures can add significant capital cost, increasing the LUEC

Even though the design of the ALMR and S-PRISM were not done using the TNF approach, the designers chose to eliminate the ECDA as a design basis event. Thus the designs of the ALMR and S-PRISM already take these simplifications into account, including the use of only one scram system, one decay heat removal system, and the simplified containment structure. These design alternatives can result in significant economic savings, but they may not be allowed by the deterministic approach to licensing, with its requirements for defense in depth and redundancy in safety systems

Thus comparing the costs of these systems in the ABR-1000 with those of the ALMR, now justified by the TNF methodology, one can make an estimate of the cost savings for these simplifications. The following sections give examples of the analysis used to estimate savings since no data on this level of detail is provided for the ABR-1000, ALMR or S-PRISM. .

An additional benefit of not designing for the ECDA events is a possible increase in the core outlet temperature to take advantage of available thermal margins. This will allow the reactor to operate at a higher efficiency, thus decreasing costs. The benefits of increasing the temperature will be discussed in the following section on increasing efficiency.

4.2.4 Cost of Traditional Containment

A traditional dome-shaped concrete containment such as those on modern LWR plants is an extremely expensive structural component. According to the estimates used in the G4-ECONS reference LWR, the containment dome cost makes up over 5% of the total overnight cost of the plant. In the development of the Sodium Fast Reactor, there have been varying opinions on whether or not a comparatively robust containment design would be required. The Japanese support the effort to use a simplified and less robust non-traditional containment, stressing that the main design concern must be focused on radionuclide confinement rather than pressure resistance due to the relatively low pressure of the primary system compared to LWRs. The

Japanese have also determined that ECDA is a highly improbable event, and therefore the containment does not need to have the strength of a traditional containment [Shimakawa et al., 2002].

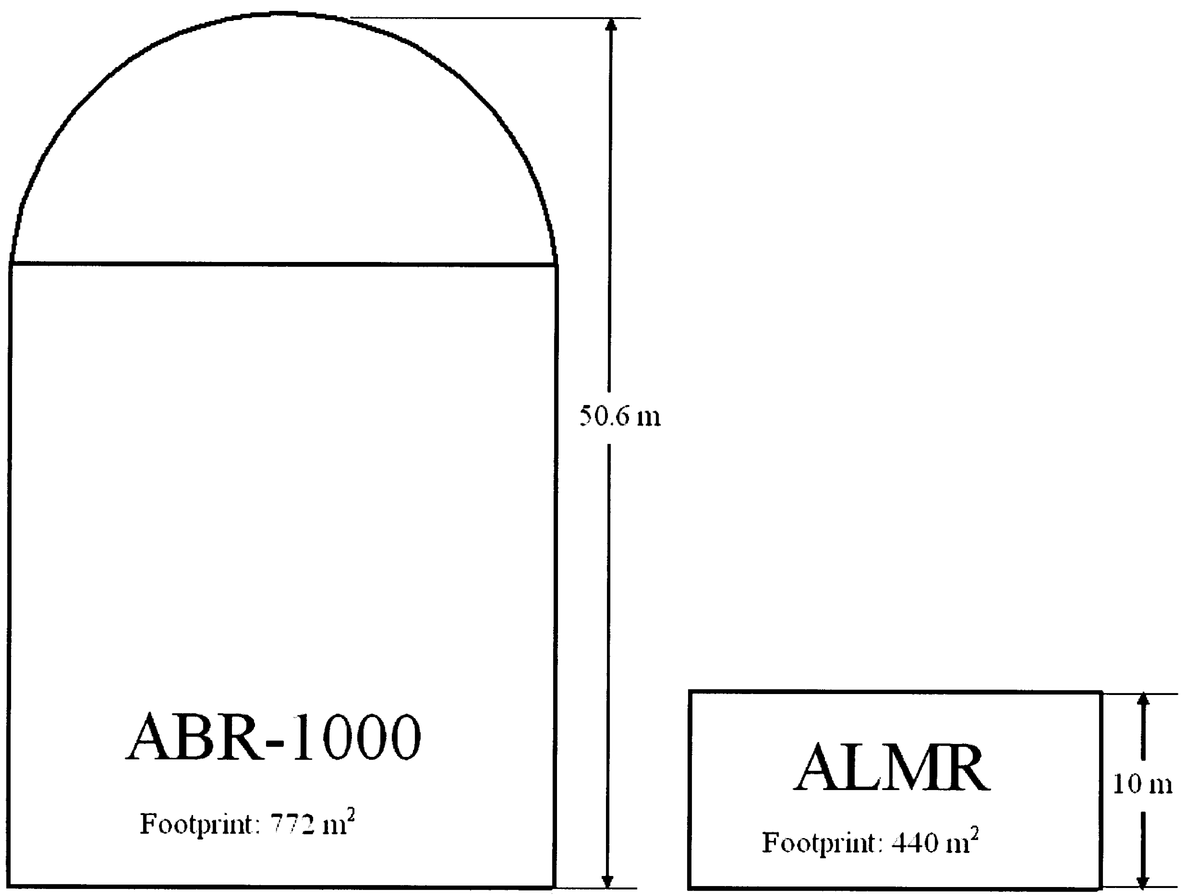
If it is determined that the SFR will require a conventional containment, the containment will be constructed to standards similar to those for the ABR-1000. However, if it is determined that a smaller, rectangular containment can be used instead, a containment building similar to that of the ALMR should be constructed.

In the ABR-1000, the reactor building is a conventional steel-lined reinforced concrete structure, in the shape of a cylindrical base and a hemispherical dome. It was designed to be similar to that of a traditional light water reactor, with sufficient room within the containment dome to perform maintenance tasks required during the lifetime of the reactor. The entire nuclear island and containment is seismically isolated to help protect against seismic events [Grandy and Seidensticker, 2007].

Each of the two reactors in one block of the ALMR has a steel-lined upper containment structure 10 meters tall, with a width of 20 meters and a length of 22 meters. Located between the two upper containments is a shared auxiliary service room, 8 meters tall, 9 meters wide and 34 meters in length. This room contains the primary Na service and cover gas cleanup systems in addition to the primary sodium storage tanks. This containment volume is a low leakage pressure retaining steel-lined concrete room and it provides access to the components on the top of the reactor vessel [Boardman, 2001]. The Nuclear Island is seismically isolated by a single, seismically isolated platform holding the reactor, its safety equipment, the intermediate heat-transport system, and the steam generator [Gluekler, 1997].

Each block of the ALMR consists of two reactors, and has a total power rating of 622 MWe. Each reactor within the block is contained within its own rectangular containment. The design of the ABR-1000 is based on a single smaller reactor, only producing 380 MWe. For a quick size comparison, the containment around one reactor of the ALMR will be examined alongside the dome-shaped containment of the ABR-1000 in Figure 4.2. The use of the containment of one reactor is done only to attain a comparison where both designs will have comparable power ratings. The reference ALMR is comprised of four reactors, two per block, and has a rated power of 1244 MW, 622 MW per block.

Figure 4.2 - Relative Size of Containment – ABR-1000 vs. ALMR



As can be seen in the figure, the traditional containment of the ABR-1000 is considerably larger than that of the ALMR. The traditional containment is five times taller and has almost twice the footprint. With such a considerable differential in size, the cost to construct the ABR-1000 will definitely exceed that of the ALMR. In fact, the surface area of the two containments has a ratio of almost 4 to 1, indicating that the traditional containment may require nearly four times the building materials of concrete and steel to construct. The comparative dimensions are presented in Table 4.2.

Table 4.2 - Containment Characteristics ABR-1000 vs. ALMR

	ABR-1000	ALMR (1 reactor)
Electrical Power	370 MW	322 MW
Containment Dimensions	30.5 m diameter 50.6 m tall	20 m x 22 m 10 m tall
Footprint	772 m^2	440 m^2
Total Containment Surface Area	4845 m^2	1280 m^2
Maximum Leak Rate	0.1%/day @10 psig	>1% /day @ 5 psig
Maximum Design Pressure	10 psig	5 psig
Seismic Isolation	Entire Reactor Building	Entire Reactor Building

*Only 1 reactor building is used for this size comparison in order to show comparable power ratings

To evaluate the cost of the traditional containment versus the smaller containment of the ALMR, the known cost of the AP1000 containment will be used as a reference along with the comparative costs of the reference LWR and ALMR in the G4-ECONS model.

The most recent economic data available on the cost of a traditional containment comes from the Westinghouse design for the AP1000 LWR reactors. According to the World Nuclear News, these containments, which measure 36 meters in diameter and 65 meters in height, will be built at a cost of $150 million [Industry Talk, 2008]. While a light water reactor containment will need to be built much thicker and sturdier to withstand the higher design basis pressures than required for an SFR, this provides a relative figure of an up-to-date cost of a traditional containment. It would not be unrealistic for the ABR-1000 containment alone to still cost more than $100 million.

The Code of Account (COA) 21 describes all costs involved with building structures and site improvements during the construction. This includes the cost of all main, auxiliary and support buildings, including the turbine buildings, steam generator buildings, warehouses, maintenance shops and the reactor containment, with the reactor containment as the dominating cost in this category. Without considering the type of containment that will be built, all costs in COA 21 should be the same, leaving the containment cost as the only major difference in this category between a traditional LWR and the ALMR.

As can be seen in Table 4.3, the aggregate value of COA 21 is much more expensive for the reference LWR with the traditional containment than for the ALMR with the smaller containment. The difference between the reference LWR and the ALMR is over $81 million. With the traditional containment built to the standards of the ABR-1000, the savings would still be around $50-80 million. This is a reasonable amount to save based on the comparative sizes of the two different containment types and the corresponding amounts of building materials required.

Table 4.3 - COA 21 Comparison from G4-ECONS model

	2007$
Reference LWR (COA 21)	$440.18 M
ALMR (COA 21)	$358.80 M
Difference	$ 81.38 M
ALMR Direct Capital Cost	$ 1906.95 M
Percentage of ALMR Total Direct Capital Cost	**4.3%**

If the conclusion can be reached that the traditional containment is not necessary, this results in no change to the existing reference model, since it already takes credit for this smaller containment. However, the use of this smaller containment does present significant savings over the use of the larger traditional one, as shown in Table 4.4. The difference in capital cost between the two options results in a savings of 4.3%. These capital cost savings translate into a 1.6% reduction in the electricity cost.

Table 4.4 - Maximum Effect of Containment on LUEC

	With Traditional Containment (+$81M)	ALMR Containment	% Change
Direct Capital Cost (2007$)	1988.33 M	1906.95 M	-4.3%
LUEC (Mills/kw-hr)	93.87	92.34	**-1.6%**

4.2.5 Safety Issues with Containment

The removal of ECDA events from the possible LBEs is the main safety issue that allows for a smaller and less rugged containment. As mentioned previously, release categories #7, 10, and 12 in Table 4.1 all describe events where there is core damage and the containment has failed. According to the safety analysis, there are no probabilistic LBEs more frequent than the 10^{-8} threshold that pose a major threat to containment [Johnson and Denman, 2009].

There are other safety issues that may need to be addressed as well before this design alternative can be approved. Some examples of these safety issues are the consideration of how the containment would perform under certain Beyond Design Basis Events (BDBE) such as an Aircraft Impact or a Large Sodium Spray or Pool Fire. According to investigations performed by Argonne National Laboratory, the less rugged containment should be able to survive the impact from a commercial aircraft, mainly due to the safety contributions of the seismic isolators [Kulak, 2003]. In addition, GE Nuclear Energy Division has performed an assessment on the robustness of the less rugged containment, and they have concluded that it can accommodate large sodium spray and pool fires without producing containment pressures that may rupture containment [Boardman, 2001].

4.2.6 Potential Savings from other Design Alternatives

If it can be shown that only one safety-grade decay heat removal system is required, such as RVACS in the ALMR model, there may be savings possible. The ABR-1000 has three independent and redundant heat removal loops and two independent safety-grade scram systems in its design [Grandy and Seidensticker, 2007]. By the removal of two heat removal loops, there may be the potential for minor savings on capital cost of the reactor. This savings will not have a major impact on the overall LUEC, as the maximum potential benefit may be about \$8.9 million, less than 0.3% of the total direct capital cost. The simplification of the scram system may allow for savings up to about \$8.1 million [Gokcek et al, 1995]. A summary of these savings is shown on Table 4.5 below. The total savings is estimated to be \$17 million, reducing capital cost by 0.54 percent and reducing electricity cost by 0.31 percent.

Table 4.5- Total Savings from Removal or Simplification of Equipment

	Total Savings from Change	% of Total Direct Capital Cost
Decay Heat Removal Loops	$8.9 million	0.28%
Scram System Simplification	$8.1 million	0.26%
TOTAL	$17.0 million	0.54%
Effect on LUEC	**– 0.29 mills/kwe**	**– 0.31%**

Although these savings are not significant, there is a significant value in using the TNF methodology should this methodology be rigorously applied with actual cost information based on specific designs of sodium cooled fast reactors.

4.2.7 Increasing Core Outlet Temperature

Increasing the Core Outlet Temperature may be the most effective design alternative that arises from the removal of ECDA events from Licensing Basis. The current design temperature of 510°C is a result of conservatism in design because of the uncertainty of requirements, especially those surrounding ECDA. Without the requirement to design for ECDA, these conservatisms can be relaxed. The increased temperature allows the reactor to operate at a higher efficiency, resulting in reduced electricity cost. The development of this design alternative through the project methodology is shown in greater detail in Section 4.3.2, but the economic benefits of raising the temperature can be seen in Table 4.6.

Table 4.6 - Effect on LUEC by Raising Core Outlet Temperature

	Cycle Efficiency	LUEC (Mills/kw-hr)	% Change
Rankine Cycle at 510°C	37%	92.34	–
Rankine Cycle at 550°C	38%	89.91	-2.43 (-2.6%)

4.2.8 Summary

The removal of Energetic Core Disruptive Accidents from Licensing Basis Events is an important example of how this methodology can be used to reduce the cost of the SFR. By demonstrating that these events have a frequency of less than 10^{-8}/yr, they fall into a region that is "screened out" under the TNF licensing process. Without the need to design for these improbable accidents, the designer can have greater flexibility in choosing safety components, and may be able to reduce costs or enhance operating performance. The overall benefits of designing without these events are shown in Table 4.7.

Table 4.7 - Summary of Design Changes ($2007)

Option	Direct Capital Cost ($ million)	Capital Cost Reduction ($ million)	LUEC (mills/kw-hr)	LUEC Reduction (mills/kw-hr)
No ECDA credit, large containment and added safety features	$2005.33	—	93.87	—
Smaller containment and removal of unneeded safety features (ALMR design)	$1906.95	-$99.38	92.34	– 1.53 (1.6%)
ALMR design operated at increased temp (550°C)	$1906.95	-$99.38	89.91	– 3.96 (4.2%)

4.3 Case Study 2 – Possible Design Alternatives that may Improve Efficiency

In Chapter 3, the economic benefits of different design changes were discussed. One of the most effective ways to reduce overall electricity cost was to target the efficiency of the reactor. Shown on Table 4.8 are the effects of higher thermal efficiency on the Levelized Unit Electricity Cost. The thermal power, capacity factor and the initial efficiency were each given in the reactor specifications for the ALMR [Gokcek et al, 1994]. The total annualized cost and the LUEC were calculated using the G4ECONS program and the economic inputs from the economic model. In order to quantify the economic benefit of increasing overall efficiency, the total electrical power and respective LUEC were calculated with an efficiency of 38% and 39%, representing a 1% or 2% increase in efficiency. With a higher efficiency, the reactor will produce more electricity as reflected in the increase in electrical power. As a rough estimate, it can be said that each 1% increase in efficiency will reduce the LUEC by about 2.5%.

Equation 4.1 and Equation 4.2 show how the efficiency can affect the LUEC for a given reactor design. The overall efficiency of the reactor will affect the annual energy production, with an increase in efficiency leading to an increase in kw-hr per year. The total annualized cost is an output from the G4-ECONS model, and represents the total expenses (capital recovery, interest, O&M, and fuel costs) for the plant each year. Therefore, as long as the total annualized cost remains constant, the larger denominator in Equation 4.1 will lead to a reduced overall LUEC.

Equation 4.1 - Levelized Unit Electricity Cost

$$LUEC = \frac{Total\ Annualized\ Cost}{Annual\ Electricity\ Production}$$

Equation 4.2 - Annual Electricity Production

$$Annual\ Electricity\ Prod. \\ = Thermal\ Power\ \times Eff. \times 24\,hours/day \times 365\ days \times Cap.Factor$$

Table 4.8 - Effects of Higher Efficiency on LUEC

	ALMR Initial Conditions	With 1% increase in Efficiency	With 2% increase in Efficiency
Thermal Power	1680 MWth	1680 MWth	1680 MWth
Capacity Factor	85%	85%	85%
Efficiency	37%	38%	39%
Electrical Power	1243 MWe	1277 MWe	1310 MWe
Total Annualized Cost	$ 829.38 M/year	$ 829.38 M/year	$ 829.38 M/year
LUEC (mills/kwe)	92.34	89.46(-2.6%)	87.60 (-5.1%)

Most design alternatives will not only affect the efficiency, but also will require an economic analysis of the capital cost changes or any effects on availability or capacity factor of the reactor. Therefore, the complete economic analysis of these changes will present the possible economic benefits along with the possible drawbacks.

4.3.1 Design Alternative – Increase Core Outlet Temperature

A design alternative that may increase efficiency is increasing the core outlet temperature. By increasing the core outlet temperature, the overall efficiency of the cycle will increase. Previous work has been done by Alexander Ludington to show potential increases in efficiency through various design alternatives [Ludington, 2009].

Possible Safety Issues

Before any changes can be suggested to raise the core outlet temperature, it must be confirmed as safe within the TNF model. To do this, the effect of raising the temperature must be assessed by evaluating the impacts on fuel behavior, accident analysis and changes in frequencies of fuel damage to be input into the PRA model. The area that will be affected is the increased likelihood of the initiators that may lead to Core Disruptive Accidents due to the additional heat in the core if an accident was to occur. As discussed in the Section 4.2.2, the analysis of ECDA results in the probability of all sequences falling below the limit to be screened under the TNF. Even if the probabilities of the initiators were increased due to the higher temperatures, there is sufficient margin provided by the scram system that prevents the overall failure probabilities of all unprotected events from rising above the 10^{-8} threshold. Higher temperatures raise the frequency or consequence of some accident sequences that resulted in sodium boiling or minor fuel pin damage, so these new frequencies must be confirmed. Any sequences that have risen above the threshold into a region of consideration must not fall into the generic release categories #3, 7, 10 or 12 as described in Table 4.1. This has been confirmed through work performed by Matt Denman and Brian Johnson through core and fuel performance modeling as well as PRA methods. Due to the number of sequences analyzed to confirm the safety analysis, the conclusions have been summarized in Table 4.9. All unprotected events remain below the 10^{-8} threshold. All sequences that result in sodium boiling and fuel pin damage still resulted in no release [Denman, Johnson and Nitta, 2009].

Table 4.9 - Results of Safety Analysis for 550C

Sequences	**Conclusions**
All Unprotected Events	Frequency remains below 10^{-8} threshold
Sodium Boiling	All sequence end states acceptable, no release
Fuel Pin Damage	All sequence end states acceptable, no release

Since the increase in core outlet temperature will be allowed according to the TNF safety criteria, the feasibility of this design alternative will depend on the economic feasibility. The two possible detrimental effects of the higher temperatures could be increased corrosion rate of the primary loop materials or the failure of the fuel cladding. Typical experience originating from light water reactors shows that an increase in temperature by only 5 to 10°C can double the rate of corrosion, due to the corrosive environment created between the water and steel [Jones, 1991]. Fortunately, the material compatibility between sodium and stainless steels allows for extended operation under most temperatures with little or no corrosion. "Our experience in decommissioning EBR-II," says John Sackett, Argonne's deputy associate laboratory director for Argonne-West, "shows that materials and components in the core can operate in liquid sodium without significant damage or corrosion. We removed components from the sodium pool after 30 years and found them just as shiny as the day they went in. We saw original marks that welders and other craftsmen had made 30 years earlier when they created the component."[Baurac, 2002] Therefore, corrosion should not be a limiting factor even at higher temperatures.

It appears that the main restriction on higher operating temperatures will be the fuel and clad material limits due to eutectic formation and burnup. With the use of metal fuel, such as in the S-PRISM, the eutectic liquidus temperature is dependent on burnup, so an onset of eutectic formation must be assumed to use as a baseline. With this temperature, the 2-sigma clad temperature calculations must be performed to determine the core outlet temperature, while keeping the hottest fuel pin below this maximum temperature [Denman, 2009].

Potential Core Outlet Temperature

The core outlet temperature can be increased by flattening the coolant temperature distribution within the fuel assembly. Flattening can be achieved by placing vertical ribs on the inner wall of the hexagonal assembly cans, which will reduce flow in the non-heated edge subchannels. This design will create better mixing of the coolant within each assembly, allowing for a more even temperature distribution across the channels. In this way, the average core outlet temperature could be increased while maintaining the temperature of the hot channel within peak cladding temperature limits. With ribs in place, it has been shown that the average core outlet temperature can be increased almost 15°C without raising the cladding temperatures in the hottest channel [Memmott, 2009]

Another potential method of flattening the temperature profile is through the use of TRU grading or diluent grading to flatten the core power profile. This option is not as effective as it can greatly affect the refueling cycle and can be difficult to control over a long period of burnup.

These drawbacks may result in a refueling cycle period that is too short to be economically feasible [Denman, 2009].

These and other design options allow for a large range in core outlet temperatures for existing SFRs. Some of these temperatures are listed in Table 4.10.

Table 4.10 - Core Outlet Temperatures of existing SFRs

	Core Outlet Temperature (°C)
ALMR (USA) [Gluekler, 1997]	500
ABR-1000 (USA) [Grandy, 2007]	510
JSFR (Japan) [Ichimiya, 2007]	550
EFR (Europe) [Pay, 2009]	545
BN-600 (Russia) [IAEA, 2007]	550

Based on the potential options and the conclusions drawn from the analysis of the fuel and potential corrosion issues, the initiating event frequency should not change and the consequences associate with the events modeled in the PRA suggest that an outlet temperature of 550°C is possible for metallic fuel as well as oxide fuel. At this time, there has not been further analysis on the economic benefits and tradeoffs for metallic and oxide fuel.

As previously shown in Table 4.8, the increase in temperature leads to increased efficiency and ultimately lower electricity cost. As will be shown in the following section, the results of higher temperatures are magnified even further if the use of the Supercritical CO_2 cycle can be adopted.

4.3.2 Design Alternative – Change Power Conversion System

Another design alternative identified and analyzed is the option to change the Power Conversion System (PCS). One attractive option is to use the Supercritical CO_2 (SCO_2) cycle instead of the traditional Rankine cycle.

From the safety standpoint for licensing under the Technology Neutral Framework, the type of PCS will have little effect on any existing sequence. It is not considered a safety grade system, and any mitigation or prevention benefits will have no effect on the safety case for licensing. However, non-safety grade systems can affect the performance of the reactor during accident sequences, and these potential effects must be analyzed. If an alternative power conversion system is selected, the impact of this system must be analyzed to assess potential accident scenarios that affect the core and radiation release. Replacing the steam with SCO_2 has some advantages in that water is removed as a potential reactive agent. However, SCO_2 operates at very high pressures, as high as 20 MPa [Hejzlar et al, 2006], and a failure of the intermediate heat exchanger could introduce high pressure CO_2 into the primary system, potentially overpressurizing it and voiding the core. The likelihood and consequences of these events need to be analyzed in the TNF framework to assess whether the safety goals are still met and whether the economic benefits can be realized.

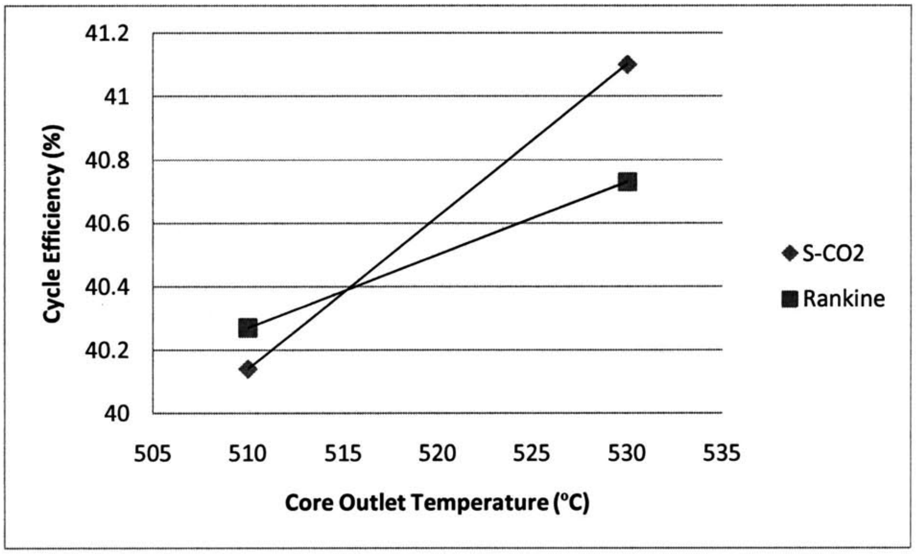
Economic Benefits of SCO_2

The economic benefits of using the Supercritical CO_2 cycle instead of the Rankine cycle are twofold: there is the potential for very large savings in capital cost due to the reduced size of the turbomachinery and there may be the potential for operation at higher efficiency than with the traditional Rankine cycle.

The potential for reduced capital cost stems from the large difference in size between the components required for a Rankine cycle versus those required for a Brayton cycle. Based on work performed for a previous study by Vaclav Dostal of MIT, a SCO_2 PCS may be as much as six times smaller than a Rankine PCS with a comparable power rating. In COA 23 – Turbine and Generator Plant Equipment, the SCO_2 cycle cost was about one-third of the cost of the Rankine Cycle [Dostal, et al, 2005]. This results in significant savings, as COA 23 is a $335 million (2007$) category. If the designer was able to reduce capital cost in this category by two-thirds, it would be a saving of $223 million, almost 12% of the direct capital cost of the SFR.

The second potential economic benefit lies with the performance of the Supercritical CO_2 PCS at higher temperatures. Based on efficiency and performance modeling done by Alexander Ludington of MIT, trends were developed to understand the relationship between increasing temperature and efficiency. As shown in the previous section, increasing the core outlet temperature results in higher overall efficiencies. When comparing the S-CO_2 cycle and the Rankine cycle as seen in Figure 4.3, the slope of the line describing these relationships are different. The slope for the S-CO_2 cycle is much steeper, indicating that the efficiency will increase more for the same change in outlet temperature. There is a critical point where these efficiencies are the same for either cycle, and this point has been determined at 518°C. Therefore, for any core outlet temperature higher than this critical point, the S-CO_2 cycle will be more efficient. With a core outlet temperature of 550°C, this efficiency gain through switching PCS options may be about 0.75% [Ludington, 2009].

Figure 4.3 - Efficiency Comparison with Varying Core Outlet Temperature



	S-CO2	Rankine
Efficiency at 510°C	40.1 %	40.3 %
Efficiency at 530°C	41.1 %	40.7 %
% Increase	+1.0 %	+0.4 %

Taking into account both the effects of capital cost reductions and efficiency gains with the Supercritical CO_2 cycle, there is the potential for a significant reduction in electricity cost. In Table 4.11, the comparison of the two cycles is shown with a core outlet temperature of 550°C. The decrease in LUEC is nearly 2% only from this single change.

Table 4.11 - Effect of S-CO2 Cycle on Capital Cost and LUEC ($2007)

	Rankine Cycle at 550°C	S-CO_2 Cycle at 550°C	Change
Direct Capital Cost (2007$)	1906.95 M	1683.37 M	- $223.58 M
LUEC (Mills/kw-hr)	89.91	84.50	-5.41 (-6.0%)

4.3.3 Summary

By raising the temperature and using the Supercritical CO_2 cycle, the overall efficiency of the Sodium Fast Reactor can be increased. The benefits of these changes are summarized below in Table 4.12 showing that a savings on capital cost of $223.58 million (11.9%) yields a total cost of electricity savings of 8.5%, if both higher core outlet temperatures and SCO_2 cycles are used for SFR applications.

Table 4.12 - Maximum Potential Savings through Efficiency Gains ($2007)

	Direct Capital Cost	LUEC (Mills/kw-hr)	Savings
Rankine Cycle at 510°C	1906.95 M	92.34	–
Rankine Cycle at 550°C	1906.95 M	89.91	2.43 (2.6%)
S-CO2 Cycle at 550°C	1683.37 M	84.50	7.84 (8.5%)

4.4 Case Study 3 - Manufacturing the Steam Generator as Non-Safety Grade

The steam generator is the largest individual direct expense in the construction of the ALMR, comprising over 10% of the total direct capital cost. Therefore, it may be a target for possible large-scale savings. Due to the magnitude of its contribution to overall cost, any reduction in cost of the component will have a large impact on the overall cost.

4.4.1 Risk-Informed Special Treatment Requirements for Reactors

Recent changes have been proposed by the US Nuclear Regulatory Commission regarding the special treatment requirements within the scope of structures, systems and components (SSCs). These changes would make use of a risk-informed process to evaluate the safety significance of each SSC, and allow for a more focused determination of which will require special treatment requirements under 10 CFR 50.69. These special treatment requirements determine the cost of the components largely due to the demands of various safety classifications.

10 CFR 50.69 describes four classes of SSCs: RISC 1, 2, 3 and 4 [10CFR50.69].

Risk-Informed Safety Class (RISC) -1	Safety-related SSCs that perform safety significant functions
Risk-Informed Safety Class (RISC) -2	Nonsafety-related SSCs that perform safety significant functions
Risk-Informed Safety Class (RISC) -3	Safety-related SSCs that perform low safety significant functions
Risk-Informed Safety Class (RISC) -4	Nonsafety-related SSCs that perform low safety significant functions

SSCs within RISC Class 1 are currently subject to special treatment requirements and would remain subject to these requirements. SSCs that would fall into this category would be those that are concluded to be significant contributors to plant safety through the risk-informed categorization process. An example of a Class 1 component would be the components within the scram system, such as the electrical equipment required to send the signal to scram.

SSCs within RISC Class 2 are not presently subject to special treatment requirements, as they are items that were previously categorized as non-safety related. Examples of these components could include emergency diesel generators or startup feedwater pumps. These SSCs might need to have requirements to maintain reliability and availability as determined through the plant PRA.

SSCs within RISC Class 3 are currently subject to special treatment requirements, but are items that can be shown through the risk-informed categorization process to not be significant contributors to plant safety. If a component was deemed to fall within this class, it would still need to meet functional requirements, but with a reduced level of quality assurance. For these items, the level of assurance provided by the commercial grade programs shall be sufficient.

SSCs within RISC Class 4 are not subject to special treatment requirements, and will continue to be beyond the scope of special treatment requirements [McKenna and Reed, 2001].

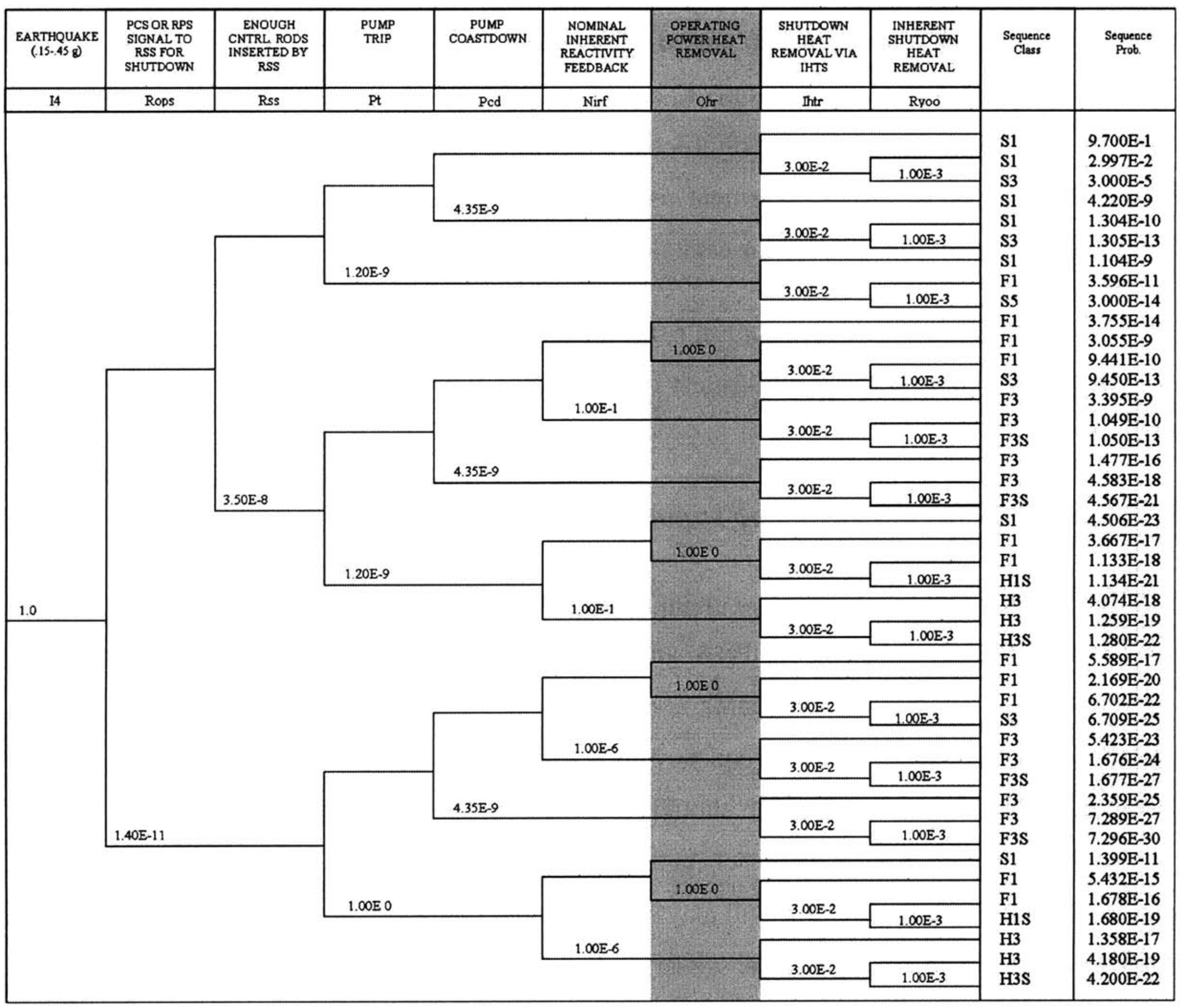
The major change lies within RISC Class 3. According to the CFR, the categorization process must "consider results and insights form the plant-specific PRA." This change allows the licensee to use the results of PRA analysis to remove items from the special treatment requirements. These special treatment requirements vary by individual component, but they include additional design considerations, qualification of materials, documentation, reporting, maintenance, testing, surveillance and quality assurance above and beyond the industry-established requirements classified only as commercial grade [McKenna and Reed, 2001]. These additional requirements may add a significant cost to these SSCs. Therefore, the opportunity to remove some of these special treatment requirements by classifying an item as RISC Class 3 presents a significant opportunity for capital cost reduction.

Therefore, if it can be shown through a risk-informed process that the steam generator is not a significant contributor to the overall safety of the plant, it may be removed from the safety grade category.

4.4.2 Use of PRA to Confirm Safety Standards

As described previously, the effect of manufacturing the steam generator as non-safety grade will be reflected through changes the PRA analysis. According to the TNF, if an item is not manufactured to safety grade standards, it cannot be accounted for in the LBE determination. Therefore, when determining the frequency of any event that involves the steam generator, the probability of failure for its function should be set to 1.0, simulating that it will always fail. In order to determine the impact of this change, the PRA event trees must be analyzed to determine what function the steam generator will play in an accident scenario, and how it will act to prevent or mitigate any possible core damage scenario. For this analysis, the fault trees from the PRISM PRA will be used to initially determine what role the steam generator will have in all accident scenarios. An example of one fault tree from the PRA is shown in Figure 4.4.

Figure 4.4 - Sample Event Tree from PRISM



S1, S3, S5	Loss of Shutdown Heat Removal
F1, F3, F3S	Loss of Flow Accident
H3, H1S, H3S	Unprotected Loss of Heat Sink Accident

*Accident severity increasing with higher numbers

As can be seen from the event tree, the steam generator is one of several means of heat removal during an accident scenario. In fact, this is the only contribution towards safety that the steam generator provides during any accident in this type of reactor. According to the PRA, the steam generator can provide emergency heat removal as an auxiliary heat removal system through steam venting while water is available [Hackford, 1986].

Referring to the discussion on RAW value from Chapter 2, the RAW value for the Steam Generator represents its importance in the safety case to mitigate any potential accidents. As seen in Table 2.1, the RAW value for the steam generator is 1.01. This shows that this component does not contribute much to the mitigation of any dominant sequences. Based on the results from the PRA, the steam generator can be removed from the safety grade category.

4.4.3 Economic Impact

Nuclear safety-grade products such as valves and piping have been shown to have a premium of 2 to 6 times the cost of a commercial grade product, due to the bolstered requirements in quality assurance for a safety grade item [Coords, 2008]. If the same reductions were feasible with the steam generator, it is possible that over 50% savings on this item could be realized. Given that in the ALMR model the steam generator makes up about 10% of the total direct capital cost, these savings could translate into large overall savings.

On first look, the main drawback to removing the steam generator from the safety-grade category is with the issue of availability. A steam generator manufactured under commercial guidelines would have a greater probability of failure, most likely in the form of a sodium-water or sodium-steam leak. While it has been shown that this is not a major issue for the release of radiation as covered in the PRA, a tube leak within the steam generation equipment would result in a reduced availability. Therefore, the non-safety grade steam generator must be manufactured to certain performance standards to ensure proper operation.

As stated previously, changes have been proposed to 10 CFR Part 50 that may allow for this reduction in safety grade. In the Regulatory Guide 1.26, the design criteria laid out within 10 CFR Part 50 are summarized, presenting four different quality groups that SSCs must fall into to satisfy the general design criteria, specific to the "Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants"[10 CFR Appendix A]. These are given as Quality Groups A through D. Group A corresponds only to components of the reactor coolant boundary that must be designed, fabricated, erected and tested to the most stringent standards. Therefore, the remainder of the components falls within Quality Groups B, C or D. Quality Group B standards should be applied to safety related components within the reactor coolant boundary. Quality Group C standards should be applied to safety related components that are not part of the reactor coolant boundary. Quality Group D standards should be applied to all other water or steam containing components that are not included in Quality Groups B or C. The standards of these Quality Groups are described in Table 4.13 [U.S. NRC, Regulatory Guide 1.26].

Table 4.13 - Quality Standards from Regulatory Guide 1.26

Component	**Quality Group B**	**Quality Group C**	**Quality Group D**
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," Class 2	ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," Class 3	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1, "Rules for Construction of Pressure Vessels"
Piping	As above	As above	ASME B31.1
Pumps	As above	As above	Manufacturers' standards
Valves	As above	As above	ASME B31.1

Using the ALMR model as a reference, the steam generator was originally designed to be manufactured under their High Grade Industrial Standards (HGIS) in the 1993 ALMR design [Oda, 1993] before being upgraded to safety-grade in the 1994 ALMR Design [Gokcek et al, 1995]. Therefore, despite the proposal to remove the steam generator from the safety grade category, it must still perform at the standards originally established in the 1993 design. Under this design, the steam generator will be designed, fabricated, installed and inspected as a Group D component. However, according to the design description, there are many areas where higher standards are used to assure that the owners' investment is adequately protected. As shown in Table 4.14, the majority of the requirements for the HGIS include using the ASME Section III standards rather than the Section VIII and B31.1 Codes.

Table 4.14 - ALMR High Grade Industrial Standards

Requirement	**HGIS Requirement**
Overall QA Program	ASME NQA-2
Design and Analysis	ASME Section III
Material Supplier QA and Program	ASME Section III
Qualification of Weld Procedures and Personnel	ANSI B31.1
Control of Weld Rod Chemistry	ASME Section III
Radiograph Welds	ASME Section III
Pressure Test	ASME Section III
Helium Leak Test	ASME Section III
ISI	ASME Section III
Certification of NDE Personnel	ANSI B31.1
Lifetime Records Retention	ANSI B31.1

These High Grade Industrial Standards are almost as stringent as those that would be required of a Safety Grade, Quality Group C component. By requiring the use of the same codes from ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," the steam generator will be nearly manufactured to Safety Grade standards in order to meet performance and warranty requirements. These results seem to indicate that, contrary to the possible savings that were initially suggested, these savings will not be significantly large. The only areas where there may be savings are in the QA Program and the retention of fabrication records. According to Melita Osborne of Westinghouse Nuclear Engineering, there may be possible savings of 5-10% with these simplifications [Osborne, 2009]. This would result in a decrease of total direct capital cost by around 0.5%. Rather than changing from safety grade to non-safety grade, Westinghouse suggests that the best way to achieve cost reductions in the steam generator is to reduce the amount of hardware required, by utilizing fewer valves and shorter pipe runs.

4.5 Case Study 4 - Control Rods

4.5.1 Use of PRA to Identify

The ALMR reactor design includes six control rods per module that would be used to scram the reactor if faced with any of the initiators requiring control rods for shutdown. This is a similar layout as presented in the PRISM PRA, which can be used as an example for this purpose. Referring to Chapter 2, the SCRAM LEF is 38, showing some room for flexibility in design. In addition, if changes can be made without modifying any probabilities, there would be no effect on the PRA under the TNF approach.

Within Appendix A of the PRISM PRA, fault trees are presented for a "Single Control Rod Insertion" as well as for the "Reactor Shutdown System for Initiators which Require one Control Rod for Shutdown"[Hackford, 1986]. See Figures 4.5 and Figure 4.6. The first fault tree shows the possible failures and probabilities of these failures for each control rod, with the final failure probability of a single rod at 5.78×10^{-6}. The second fault tree begins with each of the six control rods and shows the probability of all six failing randomly along with all six failing due to a common cause failure (CCF). In this sequence, the probability of all six failing randomly is almost zero, and the CCF frequency tends to drive the overall probability of failure of the Reactor Shutdown System (RSS). It is due to this reasoning that a proposal can be made to reduce the number of control rods.

The listed frequency for CCF of three or more rods is 5.78×10^{-9}, which is three orders of magnitude lower than the single rod failure probability. With two rods, the random failure frequency of all control rods drops to 3.34×10^{-11}, and with three this frequency falls even further to 1.9×10^{-16}. Therefore even with only two rods, the CCF frequency would determine the failure frequency of the RSS based on the fault tree in Figure 4.6 [Hackford, 1986]. Using the PRA, it can be shown that the frequency of scram failure will not increase by reducing the number of control rods. Accounting for the requirement for redundancy in safety systems, three control rods rather than six will be sufficient to meet the safety requirements of the scram system [Johnson, 2009].

Figure 4.5 - Fault Tree for a Single Control Rod Insertion

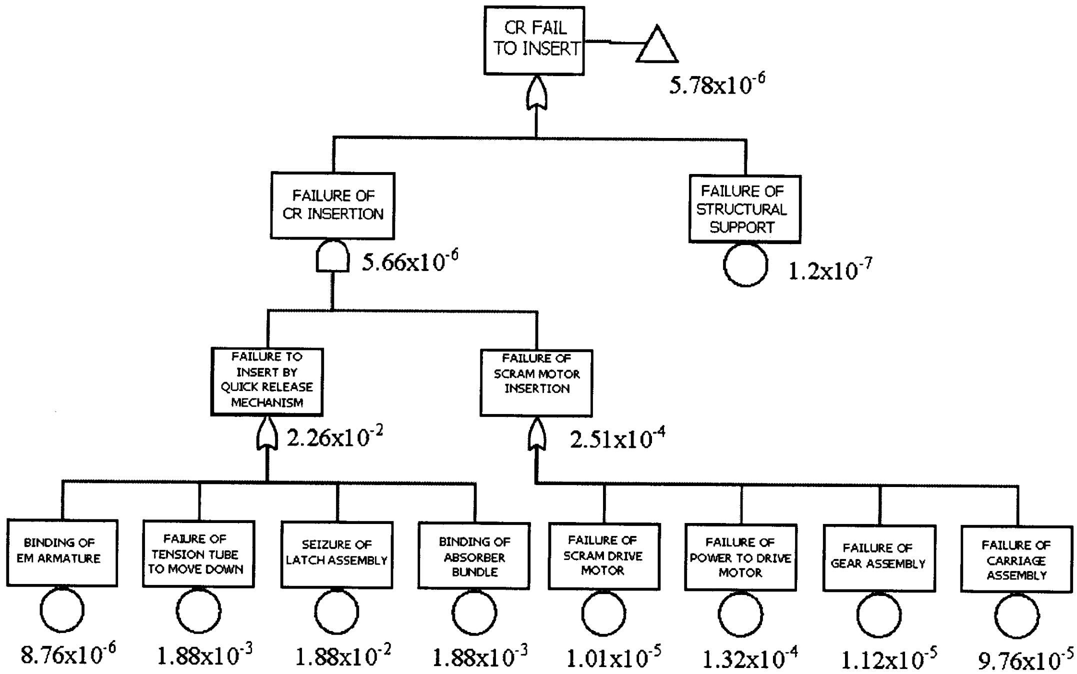
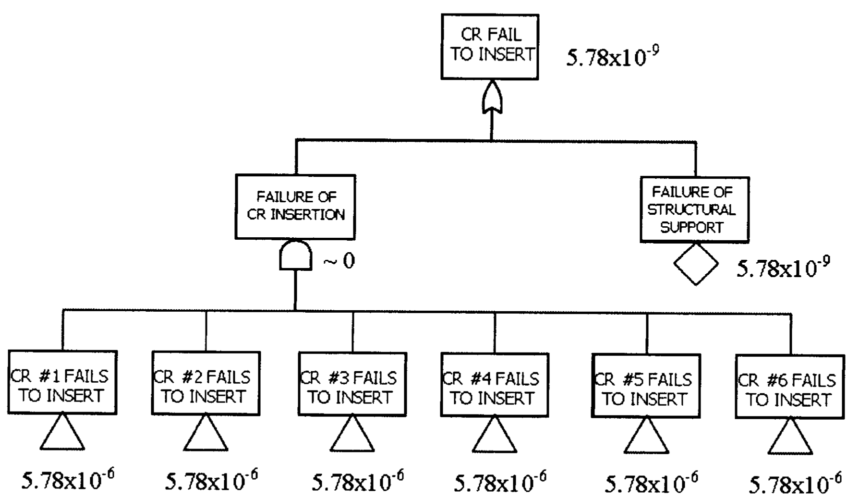


Figure 4.6 - Fault Tree for RSS for Initiators which Require 1 Control Rod for Shutdown



4.5.2 Economic Impact

From the 1994 ALMR cost estimate presented in Table 2.3, the control rods and control rod drives make up 1.33% of the total direct capital cost of the reactor. This cost is $13.79 million per reactor block [Gokcek et al, 1995]. Since the control rod drives are each independent units, this total cost should be very close to the individual cost multiplied by the number of units [El-Sheikh, 1994]. With a reduction from six to three control rods, this figure will be halved. Therefore, there is potential for cost savings of about 0.65% from the direct capital cost due to this change, about $6.9 million per reactor block.

Chapter 5 - Summary and Future Work

5.1 Summary

These four sample cases showed the potential of the cost reduction methodology by the application of the Technology Neutral Framework and applying alternative design options to improve thermal efficiency for sodium cooled fast reactors. This list is not exhaustive but provides a basis for further design optimization without affecting overall safety performance in terms of meeting safety standards. These four case studies showed the following economic results:

Case Study 1 – Removal of ECDA from LBEs

- Provides justification for the use of the smaller containment and the design without redundant safety-grade decay heat removal loops
    - Potential Benefit – savings up to 5% of direct capital cost
- Allows for the operation at higher temperatures, up to cladding temperature limits of potentially 550°C
    - Potential Benefit – up to 1% increase in plant efficiency with traditional PCS

Total capital cost savings: *$98.4 million (5.4% of direct capital cost)*

Estimated cost of electricity: *89.91 mills/kw-hr (– 4.2% of original estimate)*

Case Study 2 – Increasing Efficiency

- With Supercritical CO_2 PCS, smaller turbomachinery may lead to large savings in capital cost
    - Potential Benefit – savings up to 12% of direct capital cost
- At higher temperatures, the S-CO_2 cycle operates at higher efficiencies than the Rankine cycle
    - Potential Benefit – a further 0.75% increase in efficiency, 1.75% increase in efficiency from using Rankine cycle at 510°C

Total capital cost savings: *$223.6 (11.9% of direct capital cost)*

Estimated cost of electricity: *84.50 mills/kw-hr (– 8.5% of original estimate)*

Case Study 3 – Manufacturing Steam Generator as Non-Safety Grade

- Does not result in significant savings due to performance and reliability standards
    - Potential Benefit – Negligible cost savings, may help with licensing

Case Study 4 – Reduce Number of Control Rods

- Due to the common cause failure dominating sequences when the design has more than three control rods, six are not necessary
    - Potential Benefit – savings up to 0.65% direct capital cost

Total capital cost savings: *$13.8 million (0.7% of direct capital cost)*

Estimated cost of electricity: *92.10 mills/kw-hr (–0.3% of original estimate)*

If all of these design alternatives were implemented in the design of the pool-type SFR, the total economic benefit could be as seen in Table 5.1. The capital cost reduction over the standard pool type sodium cooled fast reactor would be about $335.8 million – almost a 17% reduction. The new LUEC could be as low as 84.28 mills/kw-hr, representing a reduction of over 10%. This reduction cuts in half the estimated difference in electricity cost between SFRs and LWRs. With further changes and more detailed information, greater reductions may be possible to bring these numbers even closer.

Table 5.1 - Total Economic Benefits from Potential Design Alternatives (2007$)

	SFR w/o design alternatives	**SFR with all suggested changes**	**Reduction in cost**	**% Change**
Direct Capital Cost	$2005.33	$1669.55 M	$335.8 M	-16.7%
LUEC (Mills/kw -hr)	93.89	84.28	9.61	-10.3%
Relative to LWR at 75.56 mills/kw-hr	+24.3%	+13.7%		

5.2 Future Work

The work in this project has helped to demonstrate the risk and performance-based methodology to improve the economic feasibility of the Sodium Fast Reactor. However, the scope of this research has been limited by the availability of detailed cost information to analyze more broad design alternatives. For example, if reliable and detailed cost information could be obtained for a loop-type SFR, it would be an interesting study to compare the benefits and drawbacks of the loop versus pool type.

In addition, the economic model could be bolstered by further research into the areas of the fuel cycle and the operations and maintenance costs. These numbers have only been assumed from the ALMR reports for the sake of this research, but it would be important to gain a better understanding of these inputs in order to create a complete economic model. With more insight into the fuel cycle and its associated costs, the design alternative of metal versus oxide fuel could be investigated more thoroughly. Furthermore, the benefits of on-site reprocessing versus a central reprocessing center and other economic variables in closing the fuel cycle could be studied which could affect the fuel cycle choice.

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Appendix A – G4-ECONS Data for ALMR

Table A-1 shows the data inputs used for the G4-ECONS model to develop the economic model for the ALMR. Table A-2 shows the outputs that were given by the model using this data.

Table A.1 - ALMR Input Values for G4-ECONS

	Description	
	Model Start Year (not presently used)	0
	Case Description	ALMR 2-Block scaled to 2007$ with some LWR values
	Reactor Plant Description	0
	Year Adjust	1
	Hours in a Day	24
	Days in a Year	365
	Site Size (Acres)	n/a
	Site Size (Hectares)	n/a
	Reactor Net Electrical Capacity	1243
	Reactor Average Capacity Factor over Life	85%
	Thermodynamic Efficiency (net)	37%
	Plant Economic and Operational Life	40
	Years to Construct (up to 10 years allowed)	6
	Cost per Acre for land (for Greenfield sites only) [cost per hectare is calculated by model]	n/a
	Average craft labor rate (for detailed capital cost breakdowns)	n/a
	Financial environment defining discount rate	0
	Cost of Capital for Interest during Construction & Amortization	11%
	Estimated D&D cost for Reactor at end-of-life (use 33% of direct cap cost [total for 20 series of accts] if no estimate available)	2550
Non-Fuel Operational Annual Recurring Costs for Reactor		
	On-site Staffing cost	22.80
	Pensions and Benefits	6.80
	Consumables	19.30

	Repair Costs	0.00
	Charges on working Capital	6.40
	Purchased services & subcontracts	6.80
	Insurance premiums & taxes	3.00
	Regulatory fees	0.00
	Radioactive waste management	7.30
	Other General and Administrative (G&A)	0.00
	Capital replacements as a % of direct capital (for LWR typically << 0.5% of direct [account 20 series] cost per year)	0.0%
	Contingency on non-fuel O&M cost	38.00
Fuel Material and Refuelling Data		
	Fuel Assembly Description	PuO2 MOX
	Heavy metal mass of fuel assembly	0
	Fuel assemblies in Full Core	220
	Fuel assemblies per Reload	55
	Average time between refuelings	2.00
Fuel Assembly Information		
	Fuel cycle code	3
	UOX fuel if open cycle	n/a
	MOX fuel if open cycle (future use)	n/a
	Switch for activation of enrichment plant tails assay optimization algorithm	n/a
	Tails assay for virgin U fed and reprocessed U fed enrichment plants (if tails assay opt switch = "OFF")	n/a
	Enrichment level of feed to enrichment plant for virgin EU (uranium)	n/a
	Enrichment level of feed to enrichment plant for reprocessed EU (uranium)	n/a
	Required U-enrichment level for virgin EU reactor fuel (initial [first] core average)	n/a
	Required U-enrichment level for virgin EU reactor fuel (reload average)	n/a
	Required U-enrichment level for reprocessed EU reactor fuel (reload average)	n/a
	U-235 content of DU diluent for higher actinide-bearing fuels, e.g. MOX reloads or FR reloads	0.250%
	Pu fraction in HM for higher actinide bearing fuels (initial cores), e.g. MOX	0
	Pu fraction in HM for higher actinide bearing fuels (reloads), e.g. MOX	0
	Non-Pu higher actinide fraction (Am,Np,etc) for higher act.-bearing fuels (transmutation fuels)	0
Reprocessing Material Balance Data		
	Percentage of Spent Fuel which is U	1
	Percentage of Spent Fuel which is Pu	0
	Percentage of Spent Fuel which is FP (fission products)	0
	Percentage of SF which is non-Pu minor actinides (Np, Am,Cm)	0
	Percentage of fission products which are segregated for targets or special disposition/transmutation	0

Nuclear material Source Unit Costs		
	Source Depleted U Ops: Storage, Conversion, Makeup-REPU Blend (for fast reactor feed: FC=3)	10
	Source Pu Storage & Treatment (for fast reactor feed if not from dedicated reprocessing) FC=3	0
	Uranium Ore (Mining and Milling U3O8) [model converts $/lbU3O8 to $/kgU] FC=1 or 2	n/a
Intermediate Fuel Cycle Step Unit Costs		
	Oxide to UF6 conversion (natural or virgin EU)	n/a
	Reprocess. U chemical form (such as UNH or metal) to UF6 conversion (REPU) [for Fuel Assy path: FA]	n/a
	Enrichment for non-REPU (Virgin) UF6	n/a
	Enrichment for REPU UF6 [Fuel Assembly path: FA]	n/a
	Fabrication of virgin EU fuel	n/a
	Fabrication of reprocessed U fuel (REPU) [Fuel Assembly path: FA]	n/a
	DUO2-diluent conversion for MOX (DUF6 or DU3O8 to spec DUO2 powder for MOX fabrication)	n/a
	Purchase or Fabrication of mixed actinide fuels incl MOX & FR variants (drivers + blankets)	2500
	Fabrication of special transmutation targets (higher actinides or certain fission products)	0
	Outside reactor bldg spent fuel storage (before repository transport/emplacement or reprocessing)	60
	Cost of spent fuel reprocessing (head end and separations component)	10600
End States for Major Materials (Unit Costs)		
	DUF6 conv/storage/geologic disposal as impure U3O8 (enrichment plant DUF6 tails)	n/a
	Excess PuO2 & other higher actinide oxide storage (from reprocessing) FC=3	2000
	Path switch for REPU: Credited fuel assys = FA (fuel assembly path) Storage only = STO Treatment, packaging, & geologic disp of REPU form = GEO	n/a
	Conversion, Pkg, & Permanent Geologic Disposal of reprocessed U (GEO)	n/a
	Storage only of Excess REPU from reprocessing (STO)	n/a
	Treatment, Pkg, & Geol Disp of separated HLW waste from reprocessing	600
	Treatment, Pkg, & Geol Disp of TRU & other non-HLW waste from reprocessing	100
	Treatment, Pkg, & Geol Disp or Stg of special separated fission products from reprocessing	0
	Geological Repository. disposition of spent fuel (waste fee in mills/Kwh or cost in $/kgHM) open cycle only	0.0
	Geo. Repository disposition of spent fuel: switch indicating cost unit above (ENERGY=mills/kwh MTHM=$/kgHM)	n/a
	Contingency on overall fuel cycle cost	0.00
Capitalised Pre-Construction Costs		
	Land and land rights	9.14

	Site permits	0.00
	Plant licensing	0.00
	Plant permits	0.00
	Plant studies	0.00
	Plant reports	0.00
	Reserved for other activity as needed	0.00
	Reserved for other activity as needed	0.00
	Contingency on 11-18 above	0.00
Capitalised Direct Costs		
	Buildings, Structures, & Improvements on Site	358.80
	Reactor Plant equipment	908.69
	Turbine/Generator Plant equipment	335.37
	Electrical equipment	130.28
	Water intake and heat rejection plant	91.39
	Miscellaneous plant equipment	82.42
	Special materials	0.00
	Simulator	0.00
	Contingency on 21-28 above	0.00
Capitalised Support Services		
	Design Services at A/E Offices (home office)	99.10
	PM/CM Services at A/E Offices (home office)	0.00
	Design services at plant site (field office)	121.32
	PM/CM services at plant site (field office)	0.00
	Construction supervision at plant site (field supervision)	218.78
	Field indirect costs (rentals, temp facilities, etc)	0.00
	Plant commissioning services	0.00
	Plant operation-demonstration run	0.00
	Contingency on 31-38 above	0.00
Capitalised Operation Costs		
	Staff recruitment and training	0.00
	Staff housing facilities	0.00
	Staff salary-related costs	0.00
	Reserved	0.00
	Reserved	0.00
	Other Owners' capital investment costs	318.69
	Reserved	0.00
	Reserved	0.00
	Contingency on 41-48 above	0.00
Capitalised Supplementary Costs		
	Shipping & transportation costs	0.00
	Spare parts and supplies	0.00

	Taxes	0.00
	Insurance	0.00
	Reserved	0.00
	Reserved	0.00
	Reserved	0.00
	Contingency on 51-58 above	0.00
Method by Which Overall Capital Contingency is Handled (Summed Sub-Cont, Percentage, or Entered Value)		
	Method designator Zero, Percentage or Value (ZER, PCT, or VAL)	VAL
	Percentage used if PCT option is selected and line 124,134,144,154,and 163 are zeroed	n/a
	Contingency value in $M to be entered if lines 124, 134, 144, 154, and 163 are zeroed	1091.50
Other Financials		
	Real Escalation (beyond general inflation)	0.00
	Fees/Royalties	0.00
	Contingency on 61-68 (reflects fin/schedule uncertainty)	0.00
Other		
	Capacity Factor Reduction (A contingency on reactor performance; may be calculated in future)	0.0%
	Code 64 (for possible future use)	0.00
	Code 65 (for possible future use)	0.00
	Code 66 (for possible future use)	0.00
	Code 67 (for possible future use)	0.00
	Code 68 (for possible future use)	0.00

Table A.2 - ALMR outputs from G4-ECONS model

RESULTS: LUEC Components for most-recently calculated strategy #		
	8	
Capital	64.79	mills/kwh
O&M	11.93	mills/kwh
Fuel Cycle	15.15	mills/kwh
D&D	0.47	mills/kwh
TOTAL	**92.34**	mills/kwh
Spec TCIC	**4304**	$/kw(e)