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In the wake of nuclear accidents such as Three Mile Island Unit 2 and Fukushima, the nuclear power industry's safety record is scrutinized. Today the main concerns lies with hydrogen production in a nuclear reactor core when the zirconium fuel cladding reacts with the water coolant during an accident, creating combustible hydrogen gas. This concern is being addressed with new technologies for new nuclear power plants and the development of hydrogen mitigation techniques for currently operating plants.

This study looks at a few potential safety upgrades including hydrogen mitigation upgrades such as hydrogen ignitors and Passive Autocatalytic Recombiners (PARs). As well as some radionuclide release mitigation safety upgrades, filtered vents and hardened vents. Using event tree/fault tree analysis to obtain probabilities and consequences of accidents and a newly developed "Economic Safety Factor (ESF)" has been proposed. The ESF is a metric that is aimed at helping utilities and regulators decide whether a particular safety upgrade is beneficial for the cost. While the ESF is not the only mechanism a utility or regulator will consider when evaluating plant changes, it can give a good indication of whether or not a safety upgrade, or another type of upgrade is a good idea. It was found that hydrogen ignitors and PAR should be implemented. The addition of filtered vents is dependent on their cost based on the results of an ESF cost-benefit analysis method. As a response to the accidents at Fukushima, hardened vents have been required on all Mk. I and Mk. II Boiling Water Reactors by the US Nuclear Regulatory Commission.

Cost-Benefit Analysis: Proposed Safety Upgrades to Currently Operating Nuclear Power Plants

by
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I understand that my thesis will become part of the permanent collection of Oregon State University libraries. My signature below authorizes release of my thesis to any reader upon request.

Jonathan T. Jordahl, Author

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LIST OF ACRONYMS

AEC – Atomic Energy Commission

BDBE – Beyond Design Basis Event
BIKE – Bicycle Imminent Kollision Evaluation
BWR – Boiling Water Reactor

CDF – Core Damage Frequency

DoE – Department of Energy

ESF – Economic safety factor
ET – Event Tree

FEMCA – Failure Effects Mode and Cause Analysis
FT – Fault Tree
FT/ET – Fault Tree/Event Tree

HI – Hydrogen Igniter

LOCA – Loss of Coolant Accidents
LWR – Light Water Reactor

NRC – Nuclear Regulatory Commission
NPP – Nuclear Power Plant

PAR – Passive Autocatalytic Recombiner
PDF – Probability Density Function
PDS – Plant Damage State
PRA – Probabilistic Risk Assessment
PWR – Pressurized Water Reactor

RPV – Reactor Pressure Vessel
RSS – Reactor Safety Study

SAPHIRE – System Analysis Programs for Hands-on Integrated Reliability Evaluations
SFRA – Small Final Risk Approximation
SOARCA – State-of-the-Art Reactor Consequence Analyses

TMI-2 – Three Mile Island Unit-2

1. Introduction

The field of Probabilistic Risk Assessment (PRA) has been vital in the safety of nuclear reactors around the world. Thanks to PRA techniques, such as event/fault tree analysis, Failure Effects Mode and Cause Analysis (FEMCA), computer modeling of accident scenarios, and other techniques analyzing the safety of Nuclear Power Plants (NPPs), the understanding of NPP safety has been significantly enhanced throughout the years. This has enabled NPP operators and regulators to make thoughtful decisions based on the conclusions of PRA. Examples of these decisions include whether or not to implement safety upgrades, setting a defining safety limit, and looking at maintenance schedules.

Since the Fukushima Daiichi accident in 2011, the Nuclear Regulatory Commission (NRC) has looked into the issue of hydrogen buildup in containment due to the interaction of water with the zirconium alloy fuel cladding at very high temperatures that are possible during severe reactor accidents. This buildup of hydrogen caused the explosions of the confinement portion of the Fukushima reactors as was seen on media around the world. The NRC and the Department of Energy (DoE) have been applying all the techniques of PRA, drawing conclusions, and making changes to the way NPPs operate to make them safer by mitigating the hydrogen build up problem. Nuclear regulators and utilities around the world are constantly looking for ways to make NPP safer. Reducing the risk that NPP pose to the public to allow NPP to continue to operate.

This thesis looks at four particular upgrades: Hydrogen Igniters (HI), Passive Autocatalytic Recombiner (PAR), hardened vents, and filtered vents. All of these are mitigation safety upgrades meaning that they only reduce the amount of damage done during an accident and do nothing in the way of preventing an accident from occurring in the first place. This being said, these upgrades could mean the difference between an accident like Three Mile Island Unit-2 (TMI-2) compared to Fukushima.

Using Fault Tree/Event Tree (FT/ET) analysis coupled with the novel idea of an "Economic Safety Factor (ESF)" each upgrade has a calculated "ESF". An ESF is a measure of the effectiveness of the upgrade to make a NPP safer by means if mitigating or to help prevent an accident from occurring per unit of money spent on the upgrade. The baseline value for comparison for the ESF is the hardened vents as the NRC has required them on all Mk. I and Mk. II Boiling Water Reactors (BWRs). Using this method it was found that HIs and PAR are worth implementing and filtered vents more often than not, are worth implementing.

Chapter two goes over important concepts to PRA and chapter three goes over the history of PRA. In chapter four the theory of the ESF is described with detailed examples of such calculations and some underlying assumptions. The fifth chapter goes over an example of a bicycle helmet being a safety upgrade for a bicycle rider with a full calculation and discussion on the results as an example to further help the understanding of the theory behind the ESF. The sixth chapter goes into detail about the proposed upgrades: HI, Passive Autocatalytic Recomiber (PAR), filtered vents, and hardened vents. Chapters seven discuss the model NPP, Surry Power Station and Peach Bottom Atomic Power Station, in detail. Chapter eight goes over the final results and discusses them with chapter nine being the final concluding chapter.

2. Background

PRA and risk is a large field of study and is still maturing to this day. With the main goal of quantifying risk there are multiple methods available of doing so. Such as FT/ET analysis, computer simulations like MELCOR/MACCS, FEMCA, and others. The biggest goal of PRA is to quantify risk so that regulators may make regulatory decisions based of the PRA.

2.1 Risk

The most basic definition of risk is the potential to lose something of value but in doing so potentially gaining something of value [1]. Risk, as defined in the nuclear industry, is the consequence of an accident multiplied by the probability of that accident. Probability is the chance that an accident has occurred while the consequence is what occurred. With reactors, consequence is often defined as the number of latent cancer deaths or the dose to the public within fifty miles of the NPP. The risk of a NPP that is of the most concern to any utility or regulator is the chance that there will be another Chernobyl or Fukushima style accident causing massive damage to the surrounding area. This very small risk probability is justified by gaining a large, carbon free power plant. In this thesis, consequence will be measured as the actual or estimated cost of an accident such as Fukushima, TMI-2, or a postulated accident. In the case of most accidents in the nuclear industry that cause core damage, or damage to any reactor core component, the consequence is decommissioning the plant. Thus, the economic consequence is the cost to decommission and cleanup.

2.3 Probabilistic Risk Assessment (PRA)

Modern PRA efforts are divided into three “levels” that correspond to important transition points in the progression of an accident scenario [2]. Level 1 starts with an initiating event and ends at core damage, or at a stable plant condition that is short of core damage. Level 2 starts with the occurrence of core damage and ends with radionuclide release, and Level 3 starts with the release of radionuclides to the environment and examines the consequences resulting from that release. Higher levels of PRA provides more in depth and detailed analysis of the risks and repercussions of accident scenarios than lower level PRA. However, higher levels of PRA cost more time and energy to complete.

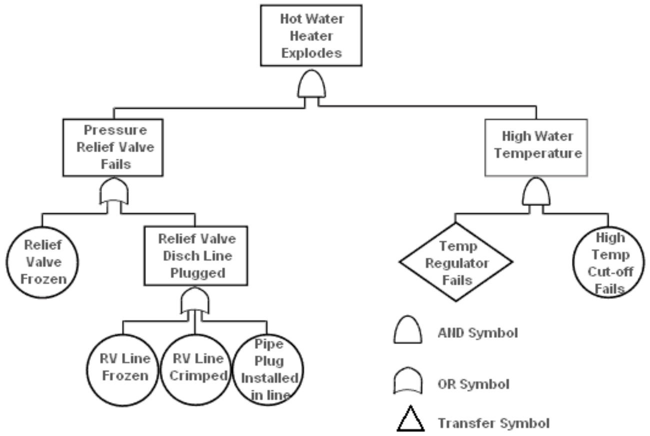

Figure 1: Simple fault tree diagram of a hot water heater exploding [3].

A Level 1 PRA is solely a calculation of the Core Damage Frequency (CDF). It looks at accident progression in terms of accidents that can lead to core damage to estimate the CDF. Typically, this would start from the definition of an initiating event and branch out through safety system success or failure until either core damage is reached or a safe reactor condition is achieved. This is represented graphically thorough Fault Trees (FTs) [4]. Each of these FTs is analyzed to provide a CDF for that particular accident, then all the frequencies are added together to get a total CDF. A simple FT of a hot water heater exploding is given in Figure 1.

A Level 2 PRA begins at the end of a Level 1 PRA by examining the plant's response to the Level 1 events that lead to core damage, and analyzing how the plant responds to this state. Incidents that lead to core damage are typically called severe accidents. Level 2 PRA is the analysis of the plant's severe accident response, and whether or not it is capable of keeping the severe accident consequences sealed within the containment building. This uses further FTs and, rather than primarily looking at safety systems success/failure looks at phenomenological events like "steam generator tube rupture" or "hydrogen explosions". Different severe accident paths lead to different Plant Damage States (PDS) when core damage occurs, severe accident progression analysis is necessary for each PDS, making Level 2 PRA significantly more expensive and lengthy to conduct than a Level 1 PRA.

A Level 3 PRA begins with radionuclide release, where Level 2 analysis ends. The results of a Level 3 PRA provides an estimate of the consequences of a radionuclide release, and when combined with Levels 1 and 2, presents an overall estimate of the effect on the people living near the plant and the potential for the plant to contaminate the surrounding environment with radioactive material. The consequences of the accident, both in terms of the health of the public and the quality of land, depend on multiple factors. For example, population density and evacuation readiness primarily affect only the health of the public while other factors such as weather conditions, geography, and the size of the radionuclide release can affect both the health of the public and quality of the land. A Level 3 PRA estimates the final measure of risk by combining the consequences of the accident and the likelihood of a radionuclide release. However, one is rarely completed because it requires a great deal of data and computational power and is thus tends to be a very expensive exercise. The various paths to radionuclide release in an accident scenario affect the nature of the radionuclide release, and these differences need to be accounted for and sufficiently analyzed.

2.4 Fault Tree/Event Tree Analysis

A FT is a deductive failure analysis, top down method utilizing Boolean logic to find the probability of an undesired event. It does this by mapping out all of the different failure possibilities in the system's components and uses "gates" to tie them all together ending in the undesired event [4]. FT analyses have been a nuclear industry standard PRA technique for some time and continues to be extremely valuable to the reactor safety community because of their simplistic methodology and ease of solution. There are many computer programs out there that can solve a FT such as Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) [5], FT analyzer by ALS [6], and many more. However, SAPHIRE is a developed tool by the US DoE that does both ET and FT analysis and it has been used for many years as an industry standard and is widely available.

SAPHIRE calculates the probability of failure and generates the different paths to a failure, otherwise known as “path sets” [7]. One particular example of the use of SAPHIRE was calculating the probability of failure of a modular helium reactor producing hydrogen for industrial applications [8]. The hydrogen was produced either through high temperature electrolysis or the sulfur iodine thermochemical hydrogen production process. The study produced a total of 27 FTs, 1115 sub-trees, and 263 basic events and found about 1869 pathways to failure, also known as cut sets. With about 27 of those cut sets accounting for about 96.3% of the total probability of failure. That probability of failure for the entire system was 0.241, or 24.1% chance of failure. With so many FTs, sub-trees, basic events, and cut sets to calculate, this case study shows how powerful a tool SAPHIRE is as this FT was a level 1 PRA and was solved on a standard Windows desktop computer. It is because of its ease of use, availability from RSICC, and the codes ability to tie FTs and ETs seamlessly that SAPHIRE was chosen for this study.

Event Trees (ETs) map out various accident scenarios, i.e. the different types of accidents that can happen given a specific initiating event [9].This initiating event is usually identified as a change in the system, most likely due to some kind of transient. From this initiating event the system state changes and the system responds. If the system fails, the path goes down on the ET and if the system succeeds it goes up resulting in different subsequent responses, as depicted in Figure (2). An ET has multiple "end states". An end state is a state the system is in at the end of an ET. The end state of the system can range from a complete and total system failure and loss in which the entire system needs to be decommissioned and/or replaced to just a minor shut down period where some maintenance needs to be done and anywhere in between.

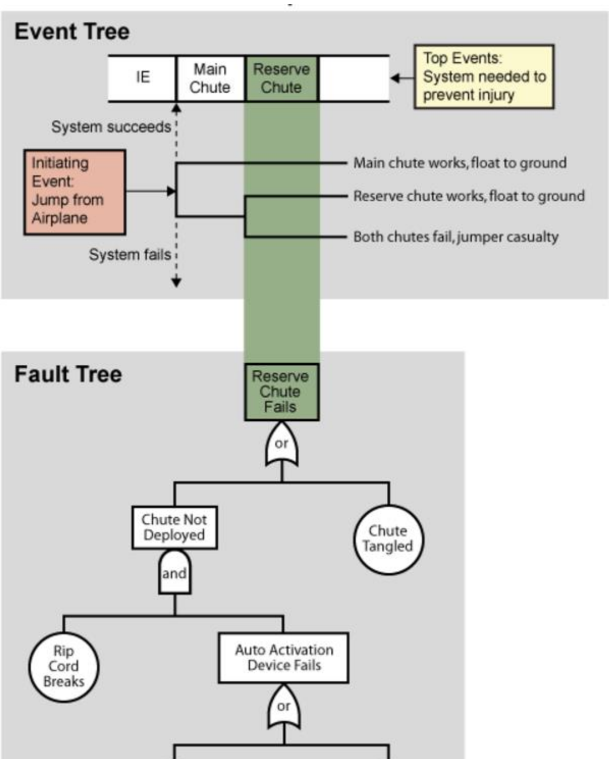
Event Tree
IE
Main Chute
Reserve Chute
Top Events: System needed to prevent injury
System succeeds
Initiating Event: Jump from Airplane
Main chute works, float to ground
Reserve chute works, float to ground
Both chutes fail, jumper casualty
System fails
Fault Tree
Reserve Chute Fails
or
Chute Not Deployed
Chute Tangled
and
Rip Cord Breaks
Auto Activation Device Fails
or

Figure 2: Example of how fault trees and an event tree tie together [10].

FTs and ETs complement one another, the probabilities of an event are found by an ET and are determined by FTs. Every event in an ET, including the initiating event, is easily modeled by a FT. Each FT maps out the different possible ways a system or subsystem can fail along with the probability, or chance, that it will fail. This probability is the probability that the system (or subsystem) will fail and is plugged into the ET's "down" path and the "up" path (system succeeding) is one less than the probability of failure. Each event is modeled like this, with a FT, to get the overall picture. The end state probabilities are then calculated by simply multiplying each probability found on the path to get to a particular end state. Figure (2) is a pictorial representation of how FTs and an ET complement one another. The computer code SAPHIRE 8 can solve both simultaneously, quickly and easily, and on a standard desktop computer.

2.5 The Hydrogen Problem

As demonstrated during the events at Fukushima Daiichi in March 2011, hydrogen build up in containment is a serious and inherent safety issue in currently operating BWRs and Pressurized Water Reactors (PWRs) around the world. Hydrogen is created from the rapid oxidation of the zircaloy cladding in both types of light water reactors (LWRs) during severe accidents. The governing chemical reaction is $H_2 + \frac{1}{2}O_2 \rightarrow H_2O$ with an activation energy of $\Delta H^{\circ} = -238\ KJ/mol$ [11]. This reaction can occur creating hydrogen gas in large quantities and concentrations at temperatures seen inside the containment structure of a NPP during a severe accident causing a significant safety issue. Hydrogen builds up in the containment and can get hot enough that it spontaneously combusts, causing explosions and potentially breaching containment and confinement thus releasing radionuclides to the environment and contaminating the surrounding area.

During the TMI-2 accident there was evidence of hydrogen explosions in the containment structure [12]. Pressure and temperature spikes were recorded and there were scorch marks consistent with a hydrogen burn on equipment inside the containment. The release of radionuclides at TMI-2 was orders of magnitude less than that at Fukushima due in part because of the size and pressure ratings of their containments. Most PWR containments (such as the one at TMI-2) are much larger in volume and are rated to withstand higher pressures (due to the higher operating pressures in PWRs) than BWR containments like those at Fukushima. This is why the hydrogen explosions in TMI-2 where not as problematic as those at Fukushima.

3. Literature Review

The nuclear power industry has one of the best safety records of any type of power generation. This is because of strict safety standards, maintenance schedules, management and regulatory oversight, and advances in safety analysis. With new technology and faster computers, safety analyses can be done more quickly and easily with a FT/ET method or with robust and accurate computer codes such as MELCOR/MACCS [13]. This is the result of over forty years of continuous advancements and improvements of the techniques, computer codes, and computers themselves that are used in these analyses.

3.1 History of Safety Analysis

PRA techniques where first developed as the nuclear industry was getting its start in the 1940's and 1950's. First, with the WASH 740 report in 1957, otherwise known as "The Brook Haven Report" which was primarily a detailed analysis of the potential consequences of NPP accidents. Followed by the WASH 1400 report in 1975 known as "The Reactor Safety Study" which included the analysis of the probability of accidents as well as consequence analysis. In 1990 another report was published called "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" which used more powerful PRA techniques and was the first time a computer code was used for a full PRA analysis of a NPP. Most recently, the "State-of-the-Art Reactor Consequence Analyses

(SOARCA)" report, NUREG 1935, was published in 2012 and is the most modern, up to date, and fully detailed Level 3 PRA analysis of a NPP that has ever been completed. These reports represent the main development of PRA since the birth of the nuclear industry in the 1940s. There have been a number of smaller reports that have added to PRA development [14-18].

3.1.1 WASH-740

WASH-740 [19] was the first study completed covering NPP risk. It is largely a consequence analysis as it assumed the worse possible case scenario for a NPP. The scenario assumed in this study included a reactor core meltdown with no containment building at a large NPP where half of the core inventory was released into the atmosphere as fine grain particles. This study, however, included no consideration of the probability of occurrence of such an accident. The results of the study reported an estimated 45,000 deaths, 100,000 injuries and more than $17 billion in property damage. These results are unrealistic and the authors at the time knew this. However it was the first time any sort of "risk" analysis was done even though there was no probabilistic study involved. WASH 740 set the bar for future safety analysis and drove the community to "best estimate" and "risk-informed" safety analysis. This type of accident is largely unrealistic, not even the worst nuclear accident in history came close to an accident such as this.

3.1.2 WASH 1400 – The Reactor Safety Study

The Reactor Safety Study (RSS), or WASH 1400 [20], was published in 1975 by the NRC and it compared the risk that NPPs pose to the public to other hazards the public is exposed to; such as motor vehicles, fires, lightning, air travel, even tornadoes and hurricanes. The results of the analysis demonstrated that all of these other risks are much greater than the risk posed by NPPs. A risk that has been proven over the decades with the extremely low amount of deaths caused by nuclear power generation in the US and around the world since the industry got its start in the 1940s.

The RSS was one of the first, full PRAs to be completed. With both consequence and probabilistic analysis done. It introduced FT and ET analysis to a large audience for the first time and identified transients (rod withdrawal, loss of flow, etc.) and small break Loss of Coolant Accidents (LOCA) as the main risk contributors to CDF, i.e. the nuclear reactor core melting, in civilian NPPs. These consequences are much more realistic than those in WASH 740 as the probabilities for transients and small break LOCAs are higher than losing the containment building. Which is a high consequence but extremely low probability so the overall risk is lower than a low consequence but “medium” probability. It is the probability analysis combined with the consequence analysis that makes WASH 1400 a PRA report.

One of the biggest things to come out of the RSS was the combined use of FT and ET analysis. The RSS cost some $4 million in the early 1970’s (around $25 million in

2016) and took about four years to complete. It would have taken much longer had ETs not been used because making an overall FT from all the FTs of the safety related systems in a NPP would have been too time consuming. So to remedy this, an ET was used which freed up resources and time [21]. The largest hurdle the RSS team faced was the lack of reliability data. In the 1970's, failure rate data was hard to come by. As a result the engineers and scientists working on the RSS had to make assumptions and make best estimates on failure rate data based on operational experience. As a result, the uncertainty associated with failure rate data was found to be off by a factor of 10 to 100 and in some extreme cases up to 1000. However, even with such large uncertainties, it was found that component reliability data could be off by factors of 100 or even 1000 and have very little effect on the final system reliability.

FT and ET analysis helped to obtain the probability of reactor risk, the other side of the coin, consequences, was also looked at extensively in the RSS. The most surprising fact the RSS produced was that the consequence of a NPP accident was not as massive as everyone at the time had thought. CDF was calculated, dose to the population was calculated and three types of radiation effects on the population were calculated: early fatalities, early illnesses, and long-term health effects. The consequences calculated assumed that medical care would be immediately available and ultimately took into account the property damage associated with the spread of radionuclides [21]. With the consequence analysis and the probabilistic completed, the RSS was completed in 1975.

The completed report had two abstracts, one comparing the risk of NPPs to other risks the public is exposed to and another summarizing the results. Unfortunately, once the RSS was published a lot of debate about it and the nuclear industry subsequently occurred.

3.1.3 Post RSS (WASH 1400)

At about the time the RSS was completed, the Energy Reorganization Act of 1974 created the NRC and the DoE out of the Atomic Energy Commission (AEC) [22]. Shortly thereafter the NRC created a risk assessment review group to assess the quality of the results in the RSS. The findings of the review group are found in the “Risk Assessment Review Group report to the US NRC” otherwise known as the Lewis Committee report [23]. It determined that the methods used in the RSS should be more widely utilized because they are more rational than previous attempts, but warned about the uncertainties in the RSS. The Lewis Committee report also pointed out that the NRC at the time did not take the main risk contributors of transients, small break LOCA, and human error seriously in their regulatory activities. It further noted that the plume dispersion models are plant specific, therefore the risks are plant specific and require refinement and a sensitivity analysis to be applied to a specific NPP. As a result of the Lewis Committee report, the NRC in 1979 withdrew its endorsements about the executive summary to the RSS [24].

After the TMI-2 accident in 1979, there was the need to answer the question of “how safe is safe enough?”. The NRC used NUREG 0880 to answer this question with a few qualitative safety goals and one quantitative safety goal of having no more than one core damage event per 10,000 years of reactor operation [25]. A few years after NUREG 0880 the NRC revised 10CFR50 with 51FR20028 which contains two qualitative safety goals and two quantitative safety goals. The qualitative safety goals are:

> "Individual members of the public should be provided a level of protection from the consequence of nuclear power plant operation such that individuals bear no significant additional risk to life and death"
>
> "Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks."

The two quantitative safety goals are:

> "The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the US population are generally exposed."
>
> "The risk to the population area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes."

These qualitative and quantitative safety goals are the same way of expressing the same thing: NPPs must be safe. Safer than any competing power generating technology and safer than all other risks combined that the public is exposed to. This is what drives the NRC to do what it does and the reason PRA is mandatory on all plants before construction begins, while the plant is under construction and during plant operation.

3.1.4 NUREG-1150 - Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants

In 1988 the NRC came out with a follow up report to the RSS. It was NUREG-1150 “Severe Accident Risks: An Assessment for Five US Nuclear Power Plants” [26]. The goal of this report was to show that plant specific PRA can be done. NUREG-1150 assumed much more realistic conditions than the RSS and involved an extensive FT/ET analysis of five NPPs: Surry Power Station, Peach Bottom Atomic Power Station, Zion Nuclear Power Station, Sequoyah Nuclear Generation Station, and Grand Gulf Nuclear Generating Station. These plants included a verity of different types of plants, such as a three and four loop Westinghouse PWR (PWR, W3 and W4), and four and six loop BWR (BWR4 and BWR6). In contrast the RSS only looked at Surry Power Station and Peach Bottom Atomic Power Station.

The report not only included internal risks but also external threats as well, such as an airplane hitting the plant, earthquakes, tsunamis, and other such events. The modeling was a combination of FT/ET analysis followed by accident progression analysis. The analysis started with determining the probabilities of any given internal or external event using FT/ET analysis, human reliability analysis, systems analysis, dependent and subtle failure analysis, PDS analysis, uncertainty analysis, and expert judgment. Then a combined thermal hydraulic, neutronic, and material mechanics computer code that describes how the plant responds to a given state and how it progresses toward core melt or some other system state was used to analyze the five plants in NUREG-1150. The code modeled all five plants and the radionuclide release to the surrounding area and calculated dose to the public for each plant; fundamentally completing a full Level 3 PRA for all five NPPs.

The next step in the NUREG-1150 method is the Level 2 PRA: accident progression, containment loadings, and structural response. Basically, the question they wanted to answer was "how the plant responds given that a particular accident has occurred?". This was done through ET analysis with an initiating event and modeling subsequent events to obtain an end state of the system. Once an end state of the system is determined, the transport of the radioactive material is then evaluated. NUREG-1150's analysis was completed using accident progression codes such as MELCOR and CONTAIN. The accidents used where those identified in the RSS that were major risk contributors, which are transients and small break LOCA.

Once the Level 2 PRA had been completed, NUREG-1150 looked at offsite consequences and emergency planning, determining the dose to the surrounding population and the extent of the damage caused by the radioactive release. The second to the last step in NUREG-1150's methodology is an uncertainty analysis of the probabilities, dose calculations, and any other consequences. NUREG-1150 is more accurate than the RSS in its uncertainty analysis because NUREG-1150 had a better data base, there was more experience doing PRA and operating plants when the study was being conducted. Uncertainty was estimated by using probability distributions for selected parameters. These distributions were generated by a panel of experts in industry.

Finally, NUREG-1150's methodology ends with risk integration, combining all the data from the previous four pieces to produce well defined probabilities and consequences. The main products of the risk analysis completed in NUREG-1150 is the mean, median, 5th percentile value, and 95th percentile value of a variety of risks for each plant.

The main results of NUREG-1150 were the determination of the probability of an individual early fatality per reactor-year and the latent cancer deaths per reactor-year. The NRC safety goals are $5x10^{-7}$ individual early fatalities per reactor-year and $2x10^{-6}$ individual latent cancer deaths per reactor-year. In NUREG-1150, it found that for a PWR and a BWR the number of individual early fatality per reactor year were found to be $2x10^{-8}$ and $5x10^{-11}$, respectively. The number individual latent cancer deaths per reactor-year where estimated at $2x10^{-9}$ and $4x10^{-10}$ for PWR and BWR, respectively. These numbers are well below the NRC safety goals and prove just how safe NPPs are.

3.1.5 NUREG-1935 – SOARCA

Most recently the NRC published a report titled "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report" in 2012, otherwise known as NUREG-1935 [27]. This report looked at Surry and Peach Bottom, both of which have been looked at extensively in the RSS and NUREG-1150. Using more advanced codes and efficient computers, a more detailed safety analysis has been completed on these two plants, yielding greater accuracy.

The approach used in SOARCA was to utilize the highly detailed MELCOR code, which is an integrated phenomenological modeling code of an accident scenario that combines reactor and containment thermal hydraulics along with radionuclide response. This code was then coupled with the offsite consequences analysis computer code called "MACCS2" or MELCOR Accident Consequence Code System, Version 2 [27]. MACCS2 is used to predict the outcome for the more likely, yet still small, probabilities of core damage events.

SOARCA relied on the increased amount of operating experience and more detailed databases for accident selection. Any accident scenario with a probability of occurring of less than $1x10^{-6}$ was left out unless the consequence was significant in which case that probability threshold was set at $1x10^{-7}$. As a result only four types of accidents were looked at for PWRs and three for BWRs. For Surry (PWRs) the four accidents looked at where long-term Station Black Out (SBO), Short-term SBO, Short-term SBO with Thermally-Induced Steam Generator Tube Rupture (TISGTR), and interfacing systems LOCA. The three accidents considered for the BWR at Peach Bottom were Long-term SBO, Short-term SBO with Reactor Core Isolation Cooling (RCIC) black start, and Short-term SBO without RCIC black start. The total CDF for PWRs and BWRs evaluated in the SORCA report are $2.24x10^{-5}$ and $3.60x10^{-6}$ per reactor-year, respectively.

For comparison, the values for CDF from NUREG-1150 for PWRs and BWRs are $4.0x10^{-5}$ and $4.5x10^{-6}$ per reactor-year.

4. Economic Safety Factor

The ESF is proposed to be useful as a simple metric that is intended to help utilities and regulators decide whether or not a proposed upgrade is a good idea and one that merits implementation. At its core, the ESF is benefit divided by cost, but gets slightly more complicated with the definition of risk used in this thesis and the time dependence of the variables involved. These factors can be simplified with a few basic assumptions. Furthermore, there are a few interesting cases of the ESF such as when a zero or negative ESF is calculated. Lastly, an example of a time dependent ESF is given.

4.1 ESF Theory

Cost in the context of the ESF is defined as the cost of an upgrade itself along with the installation, maintenance, and/or operational costs. These costs can be time dependent over the life of the reactor and are generally added up throughout the years of operation. However, as will be noted for the proposed upgrades, the maintenance and operation costs are negligible compared to initial purchase and installation of the upgrade. The summation of all of these costs is the total cost. Or expressed mathematically

$$C_T = \sum_i C_i \qquad (1)$$

Benefit is usually seen as a positive change in risk, but can also be seen as money saved or increase in profit. Benefit can be seen as a decrease in risk. Thus the initial risk minus the final risk can be seen as a benefit

$$benefit = R_i - R_f \qquad (2)$$

Here, a subscript “i” represents the condition or state before the implementation of an upgrade and a subscript “f” represents the risk after the implementation of the upgrade. The change in risk (i.e. the benefit) of implementing an upgrade is usually positive, and the probability and consequence after the upgrade is usually smaller than before the implementation of the upgrade. There are cases in which benefit can be negative and this scenario will be discussed in a later section. The benefit will be positive the majority of the time when doing a cost-benefit analysis when using the ESF of a safety upgrade.

The nuclear industry’s definition of risk can be inserted into Equation (2). Along with assuming time dependence, Equation (2) yields.

$$\mathrm{b}enefit = \gamma_i(t)\rho_i(t) - \gamma_f(t)\rho_f(t) \qquad (3)$$

Where $\gamma(t)$ is the consequence of an accident as a function of time and $\rho(t)$ is the probability of an accident occurring as a function of time.

The overall benefit (with a capital “B”) is a time integration of the change of risk (benefit with a lower case “b”). Time is integrated from the plant’s current state to its decommissioning, i.e. the operational time left. In equation form this is

$$Benefit = \int_0^{\tau} \big(\gamma_i(t)\rho_i(t) - \gamma_f(t)\rho_f(t)\big)dt \qquad (4)$$

The probabilities and consequences are time dependent because the reliability of any component changes over time due to wear and tear, even under normal operating conditions. When the system or component is brand new its failure probability also changes with time due to “burn in” failures. Burn in failures being failures of a component or system in the first few time increments of operation or use. Consequences can change over time due to a multitude of unforeseen side effects, changing social structure (e.g. Chernobyl and the fall of the Soviet Union), or more predictably the decay of the radioisotopes released to the environment.

The change of social structure within a nation can affect the consequences of a potential radionuclide release by the responsibility of the cleanup effort be placed on a different nation depending on the political situation. For example: Chernobyl is in modern day Ukraine, but the accident happened when the Soviet Union had the territory and when the Soviet Union collapsed, the burden of Chernobyl’s cleanup changed. Ukraine became an independent nation and thus Russia (formally the Soviet Union) saw no need to help clean up the Chernobyl reactor site. Over the years, the relations with Ukraine and Russia have been strained and as a result Russia has failed to meet some of its commitments helping to clean up the Chernobyl reactor site [28].

Equation (4) is the most general form of benefit, combining it with Equation (1) yields the most general form of the ESF, given in Equation (5)

$$ESF = \frac{\int_0^{\tau}\left(\gamma_i(t)\rho_i(t) - \gamma_f(t)\rho_f(t)\right)dt}{\sum_i C_i} \qquad (5)$$

The effects and variations of the ESF will be further expanded in the following sections.

4.2 Time Dependent ESF Example

As an example of a time dependent ESF, i.e. one in which the probability, consequence, and cost of the upgrade change with useful remaining plant life. A proposed upgrade can be analyzed with several different probability distribution functions such as the Weibull distribution, constant and exponential distributions. Along with a "hazard function called a "bathtub curve". This example is to show just how complex the time dependence can make an ESF calculation and the many different ways probability distributions can be calculated. Also taking into account lost profit and inflation in the consequences. This is a generic upgrade that just slightly increases the safety of a generic system.

4.2.1 Probability Density Functions

Probability Density Functions (PDFs) describe how a component or a system fails over time. There are several different PDFs, such as Weibull, constant, and exponential distributions. Additionally a "hazard function" can be translated into a PDF such as a bathtub curve. Each curve has its different advantages and disadvantages. The bathtub curve and the Weibull distribution are both based off the exponential distribution.

The “bathtub” curve is a hazard function that models “hazard rate”. A hazard rate is a ratio of the failure rate, a PDF, over the reliability rate of a component throughout time. There are three main sections of the curve: early failure (burn in), constant random failure, and wear out (burn out) failure. Early failure occurs due to initial manufacturing, construction, and material defects. Random failures can happen at any time and can be caused by a variety of factors. Burn out is when a component has been used for a long enough time and wear and tear takes its toll. The name comes from the shape of the curve as one can clearly see in Figure (3). In order to convert the hazard function, h(t), into a PDF, Equation (6) is used.

$$PDF(t) = \rho(t) = h(t)R(t) = h(t)\int_0^t h(\tau)d\tau \quad (6)$$

A Weibull distribution is a PDF that is commonly used in reliability engineering. It is often used with two parameters, a shape parameter (L), and a scale parameter (k). The shape parameter dictates the shape of the distribution, and the slope of the line, and the scale parameter dictates the order of magnitude of the distribution. Figure (4) shows a simple Weibull distribution with a shape parameter of five and a scale parameter of 1.2.

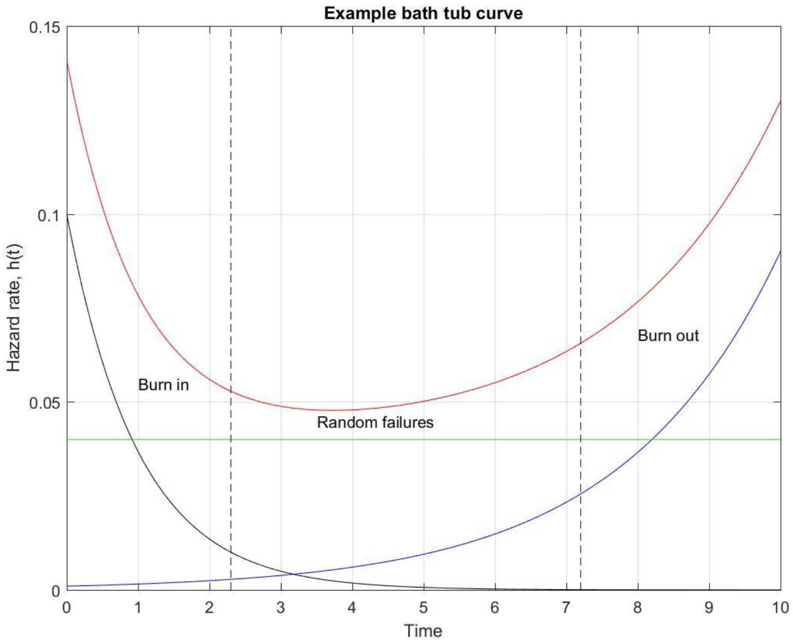

Figure 3: Example bath tub curve

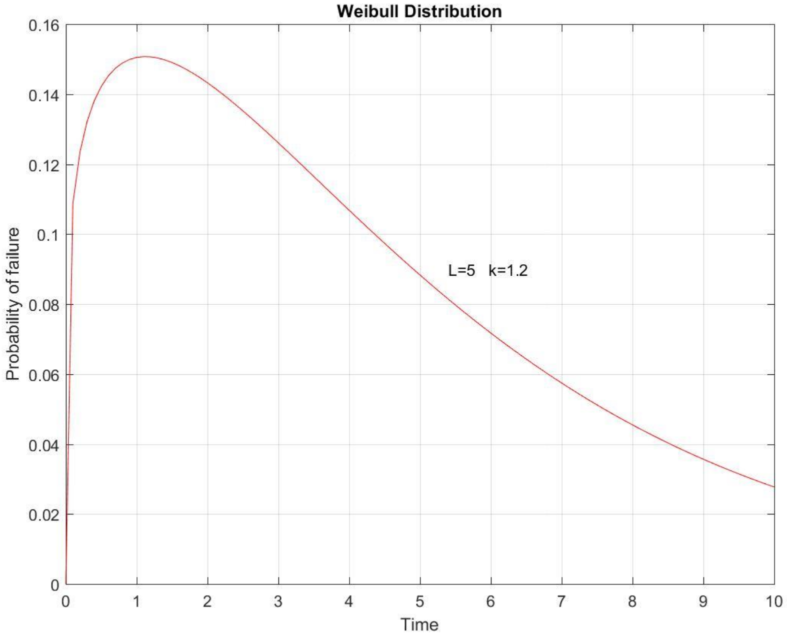

Figure 4: Weibull Distribution with L=5 and k=1.2

The constant distribution is just a measure of randomness; i.e. what is the chance that some random occurrence inside a system or component will cause it to fail? While the exponential distribution models degradation effects. The more you use something the higher the probability of failure will be due to wear and tear. This is why routine maintenance needs to be done on any system, to bring down this chance of failure.

The equations for initial failure and failure after the upgrade are described in Table (1) for each PDF (these equations were generated by the author and are assumed to resemble real time dependent probabilities). In the bathtub curve, the upgrade reduces the burn-out and random failures but the burn-in failures stay the same. With the Weibull distribution, the shape parameter increases and the scale parameter decreases slightly. The constant distribution has a lower chance of failure after the upgrade. In the exponential distribution, the initial failure rate has a lower rate of increase, but higher initial chance of failure than the final failure rate. This would be assuming the upgrade was applied when the system was installed. Table (1) lists the PDFs for each distribution. It should be noted that the bathtub initial and final failures are not PDFs, they are hazard functions and need to be plugged into Equation (6) to yield a PDF.

Table 1: List of initial and final failure probabilities for time dependent ESF example.

Distribution	Initial failure	Final failure
Bathtub	$h_{i,burn-in}(t) = \frac{e^{0.45t}}{10^6}$ $h_{i,random}(t) = 4x10^{-6}$ $h_{i,burn-out}(t) = \frac{e^{-0.45t}}{10^{11}}$	$h_{f,burn-in} = \frac{e^{0.2t}}{10^6}$ $h_{f,random}(t) = 2.8x10^{-6}$ $h_{f,burn-out}(t) = \frac{e^{-0.45t}}{10^{11}}$
Weibull	$\rho_i(t) = \frac{1.2}{1x10^5}\left(\frac{t}{1x10^5}\right)^{0.2} e^{-\left(\frac{t}{1x10^5}\right)^{1.2}}$	$\rho_i(t) = \frac{1.223}{1x10^5}\left(\frac{t}{1.5x10^5}\right)^{0.223} e^{-\left(\frac{t}{1.5x10^5}\right)^{1.2223}}$
Constant	$\rho_i(t) = 4x10^{-6}$	$\rho_i(t) = 2.8x10^{-6}$
Exponential	$\rho_i(t) = 6.2x10^{-6}e^{-0.05t}$	$\rho_i(t) = 64.3x10^{-6}e^{-0.04t}$

The consequences are the same for each PDF: the cost of an accident changes with an annuity rate of 5% and an inflation rate of 2.23% (this is the average inflation rate between 1999 and 2015 taken from the US inflation calculator website [29]) with an initial cost of the accident at $75 billion (which is the direct cleanup costs of the Fukushima Diachi nuclear disaster [30]), this includes decommissioning, cleanup costs, and payouts to refugees. A sensitivity analysis shows that the cost of the accident after the upgrade is negligible up to around 10% of the cost of the accident before the upgrade. Thus it is assumed the cost of the accident is negligible after the upgrade. This assumption is called the “Small Final Risk Approximation (SFRA)” and will be discussed later.

The accident without the upgrade requires decommissioning the plant, but with the upgrade the plant can be repaired and put back into service. Meaning that without the upgrade and if an accident were to occur the plant would suffer lost profit. That is, money not gained from operating the plant. The plant makes a net profit (i.e. income minus operating expenses) of around $1.8 million per day, or $662.9 million per year in profit. This value was obtained by looking at the cost of operating a NPP [30] and looking at the cost of electricity. A consumer power bill is included in Appendix C and is assumed to be the profit on a per kWh basis a NPP makes. The cost of the upgrade itself only changes with the same inflation rate with an initial cost of $45 million. The plant is assumed to be 24 years old with a 60 year total life.

Using Equation (5) and integrating properly on each risk's PDF, the plotted results are given in Figure (5). The calculations were done with a simple MatLab script that is given in Appendix B. Figure (5) demonstrates that the earlier the upgrade is implemented, the ESF will be higher because of the reclaimed costs and reduced risks. From a utility perspective this might not seem like a worthwhile upgrade because not every dollar spent on the upgrade is a dollar saved on an accident; this is what defines an ESF of exactly one. From a regulator perspective, this upgrade could be potentially worthwhile because it mitigates the consequences to a point that the cleanup is swift and easy. A regulator might also want to consider the economic impact that a nuclear accident might incur which can raise the cost of an accident significantly. The economic impact being the cost of milk or honey being increased as well as the loss of business as a result of an area no longer being habitable due to contamination.

All the distributions are approximately linear with the bathtub and exponential distribution deviating further from linear than the constant or Weibull distribution. This tells regulators and utilities that regardless of the probability function used, the earlier the upgrade is implemented the better. Also, depending on the PDF used, a better ESF might be obtained so care must be taken in selecting and defining the PDF used. This is an interesting result because this goes to show that no matter the PDF used, they can all be approximated to be linear in time and thus be pulled out of the integrand in Equation (5). This can also be extended to the consequences in Equation (5). This makes the ESF calculation much easier. The exponential PDF and bathtub function appear to be exponential in shape but can still be approximated as a linear line. Any difference arises in the consequence side of the equation. This result helps prove an assumption made for the analysis of the proposed upgrades in the next section.

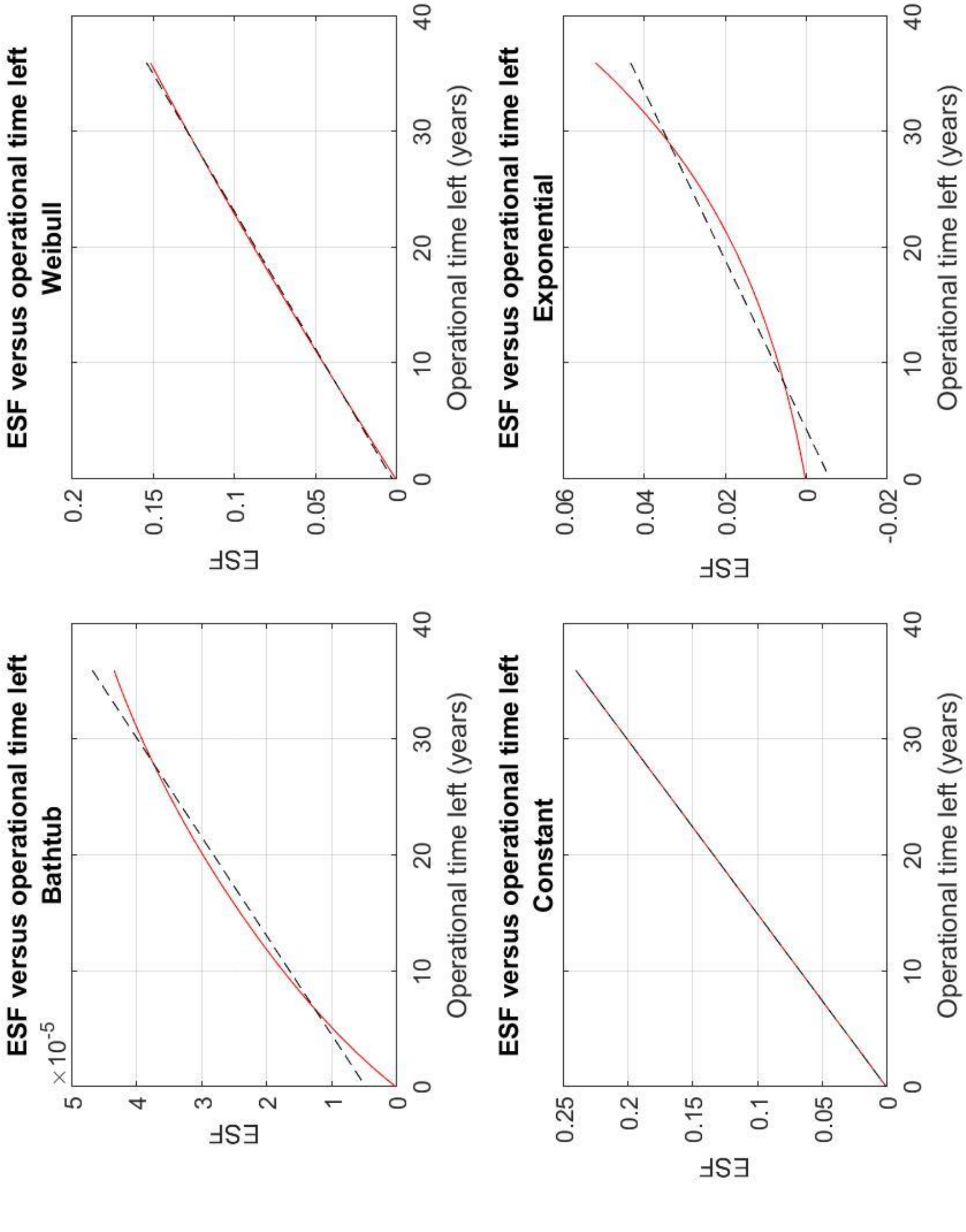

Figure 5: ESF versus operational time left of different PDFs

4.3 Assumptions to the ESF

Equation (5) can be readily integrated if the assumption is made that the probability is constant over the course of the life of a reactor and that consequence is linearly related to the useful life remaining of the reactor. These assumptions can be made because the maintenance performed on a NPP allow for time independent probabilities and the complexities of consequence analyses such as plume modeling, radionuclide concentrations, and weather patterns forces a linear relationship. Compounded by the lack of publically available economic data on nuclear disasters beyond a single, total cost of an accident. Taking this into account the probabilities can be pulled out and the consequences integrated.

$$Benefit = \rho_i \Gamma_i \tau - \rho_f \Gamma_f \tau \qquad (7)$$

Where Γ is the consequence of the accident averaged over the rest of the operational lifetime of the reactor and τ is the operational time left for the reactor.

Putting Equation (1) and Equation (7) together.

$$ESF = \frac{\left(\rho_i \Gamma_i \tau - \rho_f \Gamma_f \tau\right)}{\sum_i C_i} \qquad (8)$$

For this study, Γ is the economic cost of an accident per year of remaining operational life. If the cost of an accident is $10 billion USD and the plant has 20 years left of remaining operation life, then the cost of the accident per remaining operational year would be \$500 million per year. This number also represents the revenue lost due to the plant being shut down for 20 years when it was expected to be generating electricity for those 20 years. Put more simply, the overall consequence, Γ, is inversely related to remaining operational time, τ. As Equation (9) shows.

$$\Gamma = \frac{\gamma}{\tau} \quad (9)$$

Substituting this into Equation (7) yields

$$ESF = \frac{\left(\rho_i \mathrm{\gamma_i} - \rho_f \gamma_f\right)}{\sum_i C_i} = \frac{R_i - R_f}{\sum_i C_i} \qquad (10)$$

Where γ is the overall cost of a given accident, ρ is the probability of that accident occurring and $\sum_i C_i$ is the summation of all the costs associated with implementing a given upgrade.

For an extreme example, consider the case where a PWR power plant will be decommissioned and a new twelve-reactor Small Modular Reactor (SMR) facility will be built in its place. The cost of “implementing” such an upgrade would be the cost of decommissioning the old plant (assumed to be \$5 billion) and construction of the new plant (assumed to be \$3 billion), totaling approximately \$8 billion. The probabilities of failure for the old plant and the new plant are assumed to be 10^{-4} and 10^{-9} respectively and the consequence of an accident, should one happen, would be \$10 billion USD and \$30 million USD respectively. Inserting these assumed values into Equation (9) yields an

ESF value to be 1.25x10^{-4}. The numbers used to calculate this ESF, while somewhat realistic, are not accurate. They are for example purposes only. Since the probability and consequence are decreased, the risk is decreased, thus providing a positive benefit. Then dividing by the cost of decommissioning and construction of the new facility the ESF is obtained.

With such a low ESF, tearing down the old plant and putting up the new one does not appear to be worth the benefit. However, just tearing down the old plant (meaning $R_f = 0$) the ESF is 2x10^{-4}. This value is higher meaning that it is more worth it to decommission the old plant and not build a new one. The problem is that there is no new power to replace what was decommissioned. This is why when looking at an ESF, plant operators and regulators should consider the bigger picture. The other extreme is starting with nothing and just building a new NPP. In this case the ESF would be negative because $R_i$would be zero netting a negative result. This type of negative ESF means that the “upgrade” is not an upgrade at all because it increases the risk, thus reducing the safety. There are many factors to be considered when deciding to building a new NPP or implementing a safety upgrade than just a simple cost-benefit analysis. The complexity of the upgrade and maintenance might be a factor, available materials and technology are others, social changes yet another, and finally environmental reasons might play a large roll.

4.3.1 Event Tree Simplification to ESF

With the use of ET analysis in combination with the ESF there is one simplification to the ESF that can be made. ETs multiply the probability of each event leading to an end state to obtain the probability of the end state. That is, if there are n events each with its own independent chance of occurring and the nth+1 event, $\rho_{upgrade}$ is the failure probability of the upgrade, then the initial probability of failure is $\rho_i = \rho_1\rho_2\rho_3 \dots \rho_n$ and the probability of failure after the upgrade is $\rho_f = \rho_1\rho_2\rho_3 \dots \rho_n\rho_{upgrade}$. Inserting these two expressions into Equation (10) and simplifying yields

$$ESF = \frac{\rho_1\rho_2\rho_3 \dots \rho_n\left(\gamma_i - \rho_{upgrade}\gamma_f\right)}{\Sigma_i C_i} \qquad (11)$$

Equation (11) demonstrates that when using ET analysis to obtain the probabilities needed to calculate the ESF, the higher the probabilities anywhere in an ET will result in a proportionally higher ESF. This allows for easy manipulation of the ESF if one were to go back and add another event in the ET; simply multiply the probability of the new event and the old ESF together to get the new ESF. If a probability in the ET where to change it is just as easy to change, multiply by the new probability and divide by the old probability.

Some upgrades will only mitigate the damage and therefore reduce the consequence of an accident. Examples include filtered vents, hardened vents, hydrogen ignitors, or PARs. These safety systems will only help prevent further damage and/or reduce (or prevent all together) the radionuclide release during an accident after a core damage event. Other safety upgrades (or features) will only reduce the probability of an accident occurring and have no impact on the consequence. The best kind of upgrade is one that both reduces the chance of an accident from happening and helps mitigate the damage(s). The Emergency Core Cooling System (ECCS) is the best example of a safety system that reduces both the consequence and probability of a core damage event. It can be used before and after a core damage event. If used before, it can help keep the core itself cool to prevent the fuel from melting. After a core damage event, it can be used to reduce the temperature and pressure inside the Reactor Pressure Vessel (RPV), thus potentially preventing a radionuclide release all together.

From a plant operator stand point, they would want an upgrade that prevents core damage from occurring in the first place. Once core damage occurs they assume that the core and facility is lost. Mitigation upgrades are not a thing that plant operators would want to implement because they will only cost the operator money and only be used in the event that their facility is already lost. While from a regulator’s perspective, they want both preventative and mitigation upgrades. Regulators are most interested in protecting the public so they will look at anything to lower the risk that NPPs pose to the public.

4.3.2 Small Final Risk Approximation (SFRA)

Looking at Equation (11) it is apparent that the probability of the upgrade failing ($\rho_{upgrade}$) is going to be on the order of one over a thousand or less. Or at the very least much less than one. Combined with the fact that the final consequence (γ_f) is going to be less than the initial consequence (γ_i), Equation (11) can be simplified to the following

$$ESF = \frac{\rho_1 \rho_2 \rho_3 \cdots \rho_n \gamma_i}{\Sigma_i C_i} \qquad (12)$$

Lastly, the assumption can be made that all the probabilities of failure in a system for a NPP is the CDF. Arriving at the final equation

$$ESF = \frac{(CDF)\gamma_i}{\Sigma_i C_i} \qquad (13)$$

The SFRA will be proven in this report using the filtered vents and HI.

4.3.3 Zero ESF

In some cases the ESF will be zero or so small that it is effectively zero. In these cases the risk is the same or nearly the same before and after the upgrade is applied. These “upgrades” are not safety upgrades but might be some other type of upgrade such as a power upgrade. In these cases, the ESF is an inappropriate measure for the cost-benefit analysis. However, it would be a good indicator that the upgrade should not affect the safety of the plant.

4.3.4 Negative ESF

A negative ESF can mean one of two things, either the cost is negative or the benefit is negative. If the cost is negative, that means that whatever the upgrade is doing is saving the plant money while netting a positive benefit. In such a case a utility would most likely go for the upgrade since they would be saving money. In the case of a negative benefit, a regulator would not want utilities to go for the upgrade as it would mean increasing the risk of an accident. Nor would a utility want to go for it because increasing the risk while at the same time costing money does not make financial sense for a utility.

Another special case is when both cost and benefit are negative. This nets a positive result and one should be very careful with this type of ESF results. Saving the company money while at the same time increasing the risk might not be the best idea. Plant operators could increase core power to produce more power thus making more money, i.e. a negative cost. However, the increased temperatures and pressures alone would increase the risk of an accident, a negative benefit. This results in a negative number divided by another negative number equaling a positive ESF, even though the risk has been increased (i.e. the benefit decreased). This is just another example why of the ESF is not a standalone metric for deciding whether or not to implement an upgrade.

5. BIKE Example

As an example study using FT/ET analysis and using the ESF, a study was done to determine whether adding a bicycle helmet while riding a bike is worth the cost. Otherwise known as “Bicycle Imminent Kollision Evaluation” or BIKE for short. The key assumptions included to keep this example simple include: linear risk (probability times the consequence is the risk), human factors remain constant (i.e. only looking at mechanical failures and assuming human factors to be just a probability), and that all probabilities of failure remained constant over time.

5.1 BIKE Data and Analysis

The cost of a bicycle helmet is rather cheap, around $15-$20 from any sporting goods store for a basic, run of the mill bicycle helmet. Calculating the risk one has of having an accident while on a bicycle regardless of helmet or no helmet is an exercise with FT/ET analysis. See Appendix A for all the FTs and ET related to this example. The probability of getting into a bicycle accident is estimated to be around 1%. This means for every one hundred bicycle rides, one will result in an accident of some kind. This 1% chance is analogous to CDF.

The probability of crashing is the beginning of the ET, with each node of the ET being its own FT. Starting with whether or not a crash occurs, the next nodes to follow in order are: “Is the rider aware of they are about to crash?”, “Is the rider wearing a helmet?", "Are they wearing any other safety gear?", and lastly "Is the gear bulky?". Figure (6) demonstrates this and is the ET used for this example.

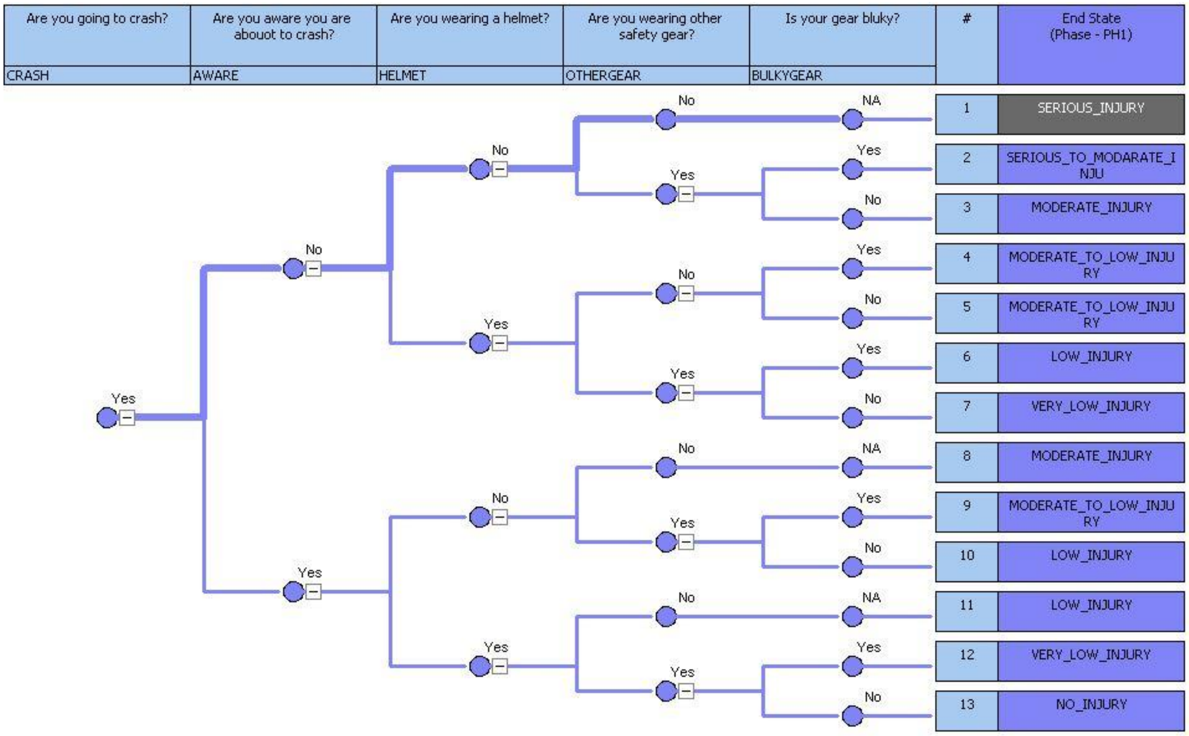

Figure 6: BIKE Example event tree

5.2 BIKE Results and Conclusion

In total, there are thirteen end states described in Table (2). With each end state condition there is an associated consequence, or cost ranging from serious (near death experience with a visit to the emergency room), to virtually no injury or damage to the bike. This range demonstrates the variety of consequences that can occur during a bicycle accident. A "moderate to low" consequence representing a scraped up knee, some scratches, bruising, etc. or possibly minor damage on the bicycle itself. “Low” consequence could represent just a minor cut on the rider or a small scratch on the paint job of the bike, same with a “very low” consequence. The difference is the size of the scratch. With a “serious” consequence, this would be being hit by a motor vehicle and having to take a trip to the emergency room, racking up a large medical bill and/or having a near death experience. The cost of the accident is an approximation for demonstration purposes and is the total integrated cost of the accident assuming the bike itself has a few years of useful life left and the rider has many years left ahead of them.

Table 2: BIKE example results

End state number	End state condition	Cost of accident	Probability
1	Serious	$10,000	2E-5
2	Serious to moderate	$1,000	2.5E-6
3	Moderate	$100	4.75E-5
4	Moderate to low	$10	3E-5
5	Moderate to low	$10	5.7E-4
6	Low	$1	7.5E-6
7	Very low	$0.50	1.425E-4
8	Moderate	$100	0.018
9	Moderate to low	$10	2.25E-4
10	Low	$1	4.275E-3
11	Low	$1	0.054
12	Very low	$0.50	6.75E-4
13	Virtually no injury	$0.01	0.012825

To calculate the ESF for a bicycle helmet (non-bulky helmet), compare end state eleven and eight. Multiplying the cost of the accident by the probability of the accident for end state eleven and eight yield

$$R_i = \gamma_8 \rho_8 = \$100 * 0.0018 = \$0.18$$

$$R_f = \gamma_{11} \rho_{11} = \$1 * 0.054 = \$0.054$$

The benefit is then the difference between these two numbers which is $0.126. To obtain the ESF, divide this number by the cost of the helmet of $15 to get an ESF of 0.0084. This is the ESF for a bike helmet. Calculating an ESF for non-bulky elbow and knee pads (assuming that the rider is wearing a helmet) in the same way yields and ESF of 6.3x10^{-5}. It should be clear to the reader that wearing a bike helmet is certainly worth the cost of one because its ESF is much higher than that of elbow and knee pads which do not offer much protection. Also, a helmet can save your life, even in the smallest of crashes, a factor that the ESF does not encompass in its numerical value for the upgrade.

6. Proposed Safety Upgrades

There are four safety upgrades that are considered in this study. Each potential upgrade helps mitigate the release of radionuclides and thus reduces the overall risk of an accident. They are: HIs, Passive Autocatalytic Recombines (PAR), hardened vents, and filtered vents. Currently the NRC has required all Mk I and Mk II BWRs to implement hardened vents. PAR and filtered vents are available for NPPs to purchase and implement but are not required by the regulating body and HIs come standard on all AP1000 units [32]. There are a number of reactors around the world with the PAR system [33] and filtered vent system.

Both the HIs and PAR work by removing hydrogen gas out of the containment, either by burning it (igniters) or by scrubbing it from the air (PAR). Filtered and hardened vents help to mitigate and control the release of radionuclides to the environment when the containment needs to be vented due to high pressures and temperatures by filtering the radio nuclides out of the air and being able to withstand elevated pressures and temperatures. Filtered vents do as the name suggests, filters the radionuclides out of the gas being vented before being released to the environment by a system of filters consisting of wet and dry filters. Hardened vents simply allow venting to occur at much higher temperatures and pressures than would normally be experienced when venting to out of containment.

6.1 Hydrogen Igniters

A HI is a rather simple device. It sometimes is also known as a glow plug (but can also use a spark plug). Diesel engines use them to start from a cold shutdown state. A glow plug is a resistor heater made of a ceramic material and gets up to 1500 ^{o}C with only 12 VAC [34]. It was found that setting up a number of these glow plugs would ignite hydrogen that built up around them causing small hydrogen explosions. These smaller explosions would prevent a much larger explosion from occurring and thus preventing a breach of containment and/or releasing radionuclides to the environment. Unfortunately, when water vapor is mixed in with the hydrogen gas, as is usually the case with an accident scenario, the igniters do not work as effectively compared to dry air. The moisture works to cool the igniter and reduce the chance of a small hydrogen explosion. Thus resulting in more hydrogen building up before a hydrogen explosion resulting in a larger hydrogen explosion that can potentially damage critical safety systems or breach containment.

Another main issue with HIs is that they are not needed for large dry subatmospheric containment often seen for PWRs. These particular PWR containments are so large and designed to withstand much higher pressures compared to BWRs and PWRs with ICE containments. Enough that all of the fuel cladding in the reactor could react to make hydrogen and that hydrogen could spontaneously combust and not breach the containment [35]. In BWRs and ICE PWRs however, this is not the case. BWR containments are much smaller and are not designed to the same level as most PWR containments. This means that HIs do not mitigate a radionuclide release in a non-ICE condenser PWR but do mitigate in a BWR and ICE condenser PWR containments. HIs will be looked at for both the PWR and BWR case. However, if the PWR is of the non-ICE Condenser containment type, then the results for the HI can be ignored.

As a result of ICE condenser PWRs not being able to handle the hydrogen problem the NRC has required the nine PWRs in the US with ICE condenser containments to install HIs [36]. Furthermore, HIs come standard in all AP1000 plants. In total there are sixteen HIs in an AP1000 [37], four located in the pressurizer compartment, two in the In-Containment Refueling Water Storage Tanks (IRWST), and a total of ten in the upper compartment. A HI design is shown in Figure (7). It is a simple, self-contained, safety system. It is designed to activate itself when a set temperature and pressures are reached. It can also be activated from the control room, should the operators activate the HI early. The version of a HI shown is more specifically a spark igniter. Glow plugs require a continuous power supply which can be a problem during a severe accident such as a SBO. As a result, Simens developed a HI to be reliable in a severe accident condition.

The cost of implementing HIs is rather low. Glow plugs/spark plugs themselves are cheap and available to anyone since they are used extensively in automotive engines. An approximate estimate of a glow plug is around $20 USD a piece [39]. The cost of

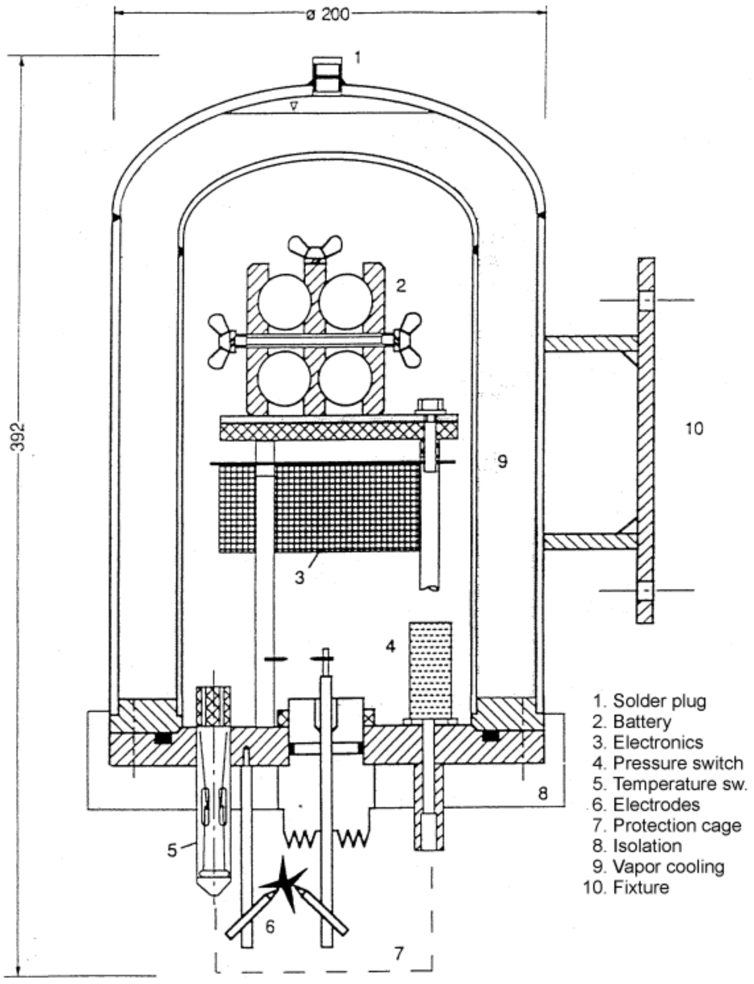

Figure 7: Hydrogen Igniter schematic [38]

developing, procuring, installing, and maintaining a system of a His is around $1 million per unit. This is the best estimate for a HI and is based off of the PAR price per unit. A HI system is slightly more complex than that of a PAR (which has a cost of $750,000), thus an estimated $1 million is assumed per unit

Due to the fact that HIs are cheap, a system can be set up with tens of HIs allowing for high reliability even when an individual HI is not considered “nuclear grade” or very reliable on its own. Assuming each HI is independent of all the rest, meaning that if one HI fails the rest can still operate as normal. Westinghouse’s AP1000 reactor has sixteen HIs set up inside containment at various locations to prevent a hydrogen explosion [37]. Combined with functional tests during each fuel outage and the massive amount of reliability data out there the development of such a system for each plant would be relatively cheap and quick.

6.2 Passive Autocatalytic Recombiner (PAR)

A PAR scrubs the hydrogen out of the surrounding atmosphere and contains it in a metal or ceramic lattice. This is accomplished by utilizing the catalytic properties of particular metals to reduce the exothermic energy of the hydrogen and oxygen reaction to occur. With the PAR the reaction can occur at lower concentrations of hydrogen by passing the containment gasses through the catalytic material. This is accomplished at a lower temperature than is required for a hydrogen explosion to occur. It works based of the Langmuir-Hinchelwood principle [40]; first the reactants diffuse into the catalyst material and then the reaction occurs and the catalyst absorbs the reactants (i.e. the hydrogen).

A PAR system is considered to be “passive” since it does not need power or operator action to operate. It automatically and continually scrubs the air of hydrogen, regardless of the containment temperature and at hydrogen concentrations as low as 1-2%. The actual design of the PAR is simple, the catalyst material is near the bottom of a flow path and as the air heats up, natural convection carries the hydrogen gas, air, and steam mixture over the catalyst material, which then scrubs out the hydrogen before expelling out steam and air at the top [41]. Figure (8) shows the operation of a PAR.

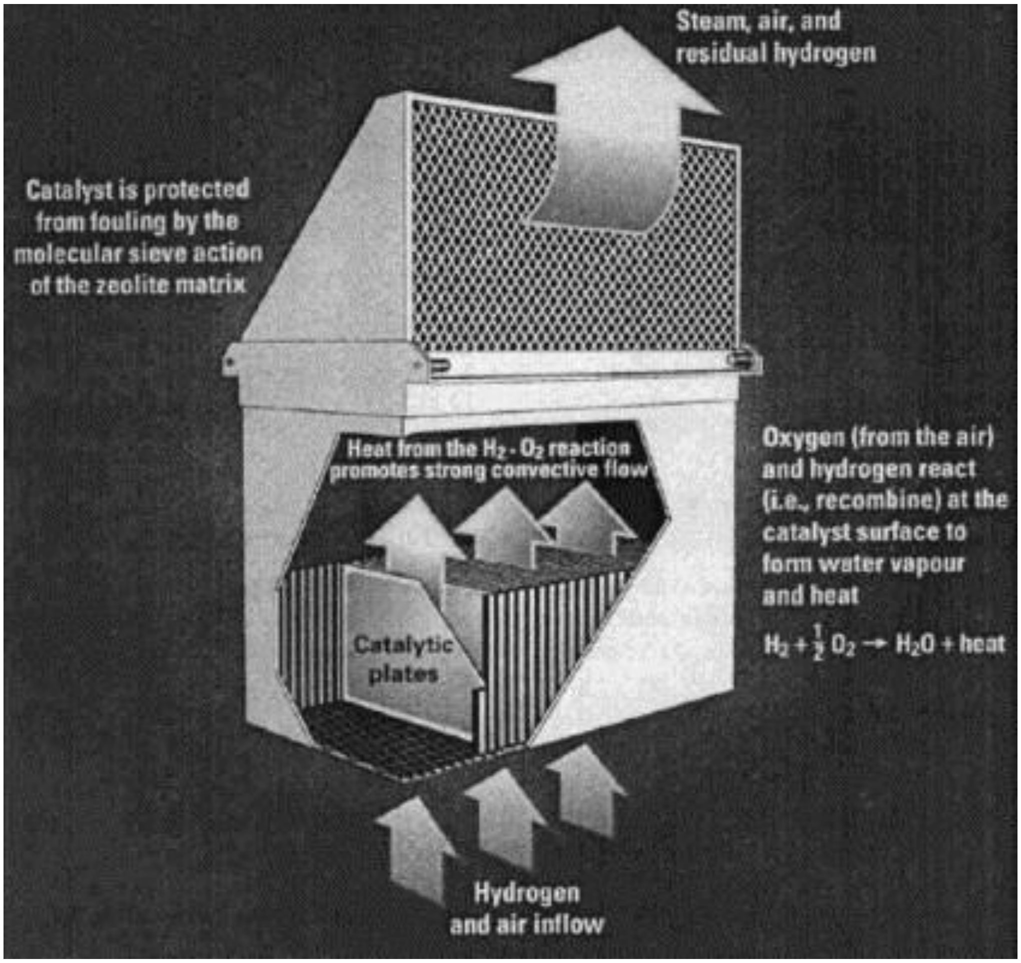
Steam, air, and residual hydrogen
Catalyst is protected from fouling by the molecular sieve action of the zeolite matrix
Heat from the H_2 - O_2 reaction promotes strong convective flow
Oxygen (from the air) and hydrogen react (i.e., recombine) at the catalyst surface to form water vapour and heat
$H_2 + \frac{1}{2} O_2 \rightarrow H_2O$ + heat
Catalytic plates
Hydrogen and air inflow

Figure 8: PAR operation [42]

AREVA sells PARs and has installed them in over 100 NPP around the world [43]. The cost of implementing a PAR system is $750,000 per unit [44]. With 16 PARs strategically placed around the plant, the cost for the system is estimated to be $12 million. Combined with its reliability it is easy to see why some vendors have implemented a PAR system. It can severely mitigate the hydrogen issue during an accident scenario.

6.3 Hardened Vents

Currently, when a BWR has a severe accident and needs to vent from the RPV to the containment, the venting system cannot handle the temperature and pressures of such an action, resulting in failure of the venting system. This is what happened during the Fukushima Dai-ichi accident and as a result the NRC is requiring all Mk. I and Mk. II BWR to install hardened vents [45]. The design requirements for such a venting system are outlined in the NRC document "Order to Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Sever Accident Conditions", EA-13-109 [46]. However, Mk. III BWRs and all PWRs would also benefit from hardened vents. Being able to vent reliably to a safe pressure and temperature can greatly reduce the consequence of a radionuclide release. When a RPVs temperature and pressure increases too much, damage to the RPV and containment start to occur, potentially releasing massive amounts of radioactivity into the environment. With hardened vents, venting can occur more than once and preventing massive damage to the RPV or containment.

The upgraded venting system needs to able to withstand high temperatures and pressures so that the values, pipes, and other associated equipment survives the venting action intact. To do this, all the piping, valves and other equipment must be engineered to withstand severe accident temperature and pressures by requiring better and/or thicker materials [47]. By the end of the upgrade, the venting system can handle Beyond Design

Basis Events (BDBEs) venting without issue. The cost of such a hardened venting system depends on the plant. It can be anywhere between $15 million and $45 million USD. For ESF calculations, several values have been chosen to compare with the other upgrades. The values looked at are $15 million, $16 million, $25 million, and $45 million. The lower and upper end to get the full range, and two values in the middle.

As the NRC is requiring all Mk. I and II BWR NPPs to install hardened vents, this will be a baseline for the evaluation of the ESF. In other words, all other safety upgrades will be compared to hardened vents. If the ESF of one upgrade is higher than that for hardened vents, then that upgrade is cost-beneficial and should be implemented. If the ESF is lower, then that upgrade should not be implemented.

6.4 Filtered Vents

Filtered vents remove radioactive material before being vented to the environment. This upgrade is a mitigation upgrade but one that could reduce the release and spread of radionuclides significantly during a severe accident. AREVA makes a filtered ventilation system that plants can purchase and they advertise that it can scrub 99% aerosolized radionuclides [48]. This reduces the overall radioactive release to the environment by a couple orders of magnitude. As a result, the spread of radionuclides via the plume will be significantly reduced. Meaning the consequence of such a venting event will be significantly reduced. One type of filtered vent works with two ways of scrubbing radionuclides: high speed venturi scrubber and a metal fiber filter. A venturi scrubber is a wet scrubber that works by mixing the gas with a liquid thus causing the gas to be trapped in the liquid. It is composed of three basic sections, a converging section, throat section, and a diverging section as shown in Figure (9). As the gas enters the converging section it is forced to speed up in accordance with the Bernoulli equation. Liquid enters either at the beginning of the converging section or at the throat. When the liquid does enter the filter system, the gas mixes with the liquid making gas/liquid droplets. The gas/liquid droplets then enter the diverging section where it slows down. The droplets can then be collected and stored, and the gas moves onto the next stage of the filtered vent system, the metal fiber filter.

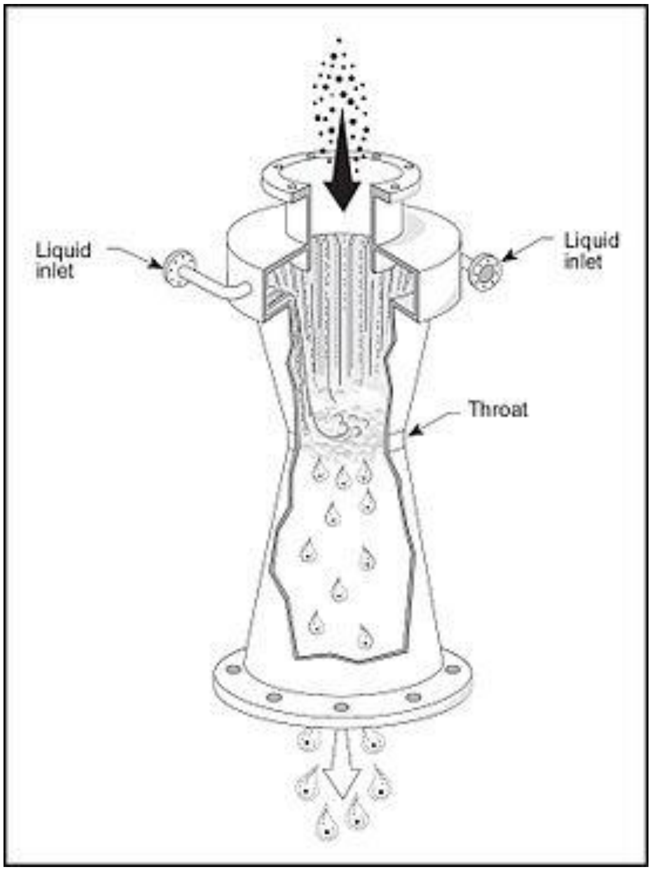

Figure 9: Basic diagram of a venturi scrubber

The metal fiber filter is much like a HEPA filter and scrubs the water droplets out of the gas just before releasing the filtered gas to the environment. Using additives in the water such as caustic soda allows for up to a 99.5% retention of iodine and 99% retention in aerosolized radioactive gases [48]. Metal fiber filters are passive and do not require any power or operator action to activate. It works by a pressure difference, the pressure in the containment building is higher than the environment so the gases are pushed through the filter to the environment. That is, when the containment building reaches a set pressure, venting automatically starts. Combined with hardened vents, the consequences of venting to the environment can be drastically reduced as the amount of iodine and other radionuclides released would be very small in comparison without the filtered vent system.

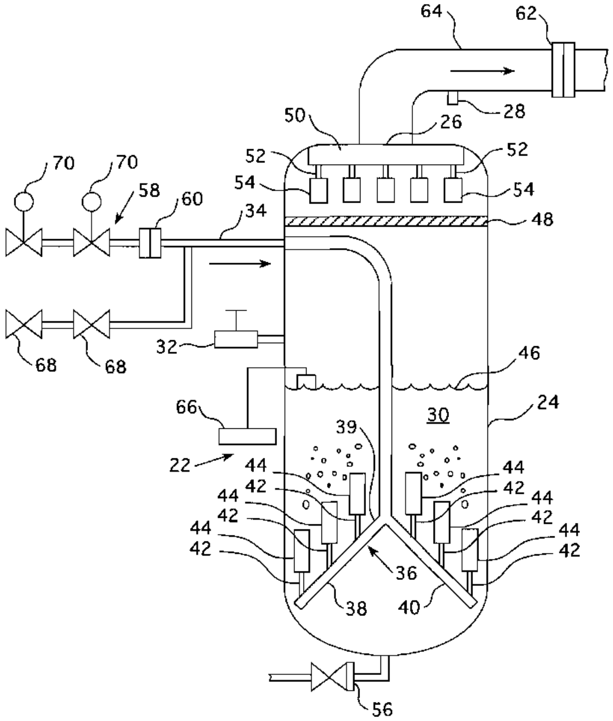

Figure 10: US Patent 2014001030 A1 Filter for nuclear reactor containment ventilation system

Another type of filter that can be applied to a BWR and PWR is described by US patent number 20140010340 A1 [49]. This filter works by filtering gases through a tank of water to scrub radionuclides and energy from the gas stream followed by a metal fiber filter as shown in Figure (10). This particular filter design will be used in the ESF calculations because of its simplicity and similarity to the AREVA filter.

The highest cost of instillation of a filtered ventilation system was in 1988 at a plant in Sweden and was $12.5 million ($25 million in 2016) for a BWRs [50]. This will be the assumed cost in this analysis to give the most conservative estimate for filtered vents. The cost of maintenance, testing, and inspections is insignificant, it is estimated to be between $10,000 and $30,000 per year (over the course of 60 years at $30,000 that is $1.8 million to keep the filter maintained. This is relatively small compared to the overall cost). This technology has been proven to be effective but there are differing reports on whether or not it is cost-beneficial or not. Such a system is required in Sweden and Switzerland by their regulating bodies along with Finland, France, Japan, and South Korea [51].

7. The Model Nuclear Power Plants

The two NPPs to be used in this study are Surry Power Station and Peach Bottom Atomic Power station. These two plants are chosen because of their history and relevance in nuclear PRA along with the fact that these two plants are still in operation today. Surry is the model PWR and Peach Bottom is the model BWR.

7.1 Surry Power Station (PWR)

Surry Power Station is located in Virginia about 17 miles north west of Newport News, VA. It is owned and operated by Virginia Electric & Power Company and Unit 1 began operating in 1972 and obtained a license extension in 2003 so the plant will be licensed to operate until 2032 [52]. Unit 2 started operating a year later and obtained license extension to be able to continue operation until 2033. Both units are licensed to 2587 MWt and are three loop Westinghouse PWRs with dry, “sub-atmospheric” containments [53]. These types of containment have very large free volumes and are kept below atmospheric pressures during normal operation [54]. This helps prevent overpressure and leakage during a LOCA.

Surry Power Station was modeled in the RSS, NUREG-1150, and SOARCA thus it is assumed to be the model PWR for this study. As the Lewis Committee report, NUREG-1150 and SORCA all explain, each plant will have to do its own in depth analysis to get its respective CDF but using Surry will give all other PWR plant operators an approximate idea of what the ESF will be for each particular upgrade.

The CDF listed in NUREG-1150 and SORCA for Surry are $4.0\text{x}10^{-5}$ and $2.2\text{x}10^{-5}$ per reactor-year, respectively. SORCA reports a lower CDF, i.e. a higher factor of safety, because of upgrades made to the plant between the two reports (NUREG-1150 was published in 1990 and SORCA in 2012). SORCA also has more powerful and more accurate estimations on accident frequencies, source terms, and environmental impact compounded with much more powerful super computers to perform the calculations.

7.2 Peach Bottom Atomic Power Station (BWR)

Peach Bottom Atomic Power Station is a three unit site located in Delta Pennsylvania operated by Exelon Generation Co., LLC [55]. Units 2 and 3 are Mk. I GE Type 4 BWRs, while Unit 1 was a gas cooled reactor that is now decommissioned. Both of the BWR units are rated at 3,951 MWt with Unit 2 starting operation on July 5, 1974 [56] and Unit 3 starting operation on December 23rd of that same year. Both units have received a license extension, Unit 2 received its renewal on August 8, 2003 and Unit 3 acquired its on July 7, 2003. They will continue to operate until 2033 and 2034 respectively.

As is the case for Surry, Peach Bottom is the representative BWR for all BWRs. This is because of its history of being used in PRA. NUREG-1150 reports Peach Bottom’s CDF as $4.5\text{x}10^{-6}$ while SORCA calculates a CDF of $3.3\text{x}10^{-6}$. These results are close to one another and smaller than that of Surry, meaning BWRs appear to be safer than PWRs. As a result, in Equation (11), the CDF will be higher in the PWR case meaning any upgrade will have a higher ESF in a PWR than a BWR. At least in this simplified analysis.

8. Results and Discussion

Table (3) lists the data for each upgrade for both PWR and BWR from the utility perspective and the regulator perspective. The difference is in the cost of the accident, the cost of the accident for a utility is about $10 billion as anything above that is picked up by the government according to The Energy Policy Act of 2005 [57]. The regulators cost is modeled from the Chernobyl accident in Ukraine. The cost is around $235 billion. [58] From the discussion on the SOARCA report, the PWR and BWR CDF used in the ESF calculations are $2.24x10^{-5}$ and $3.60x10^{-6}$ respectively. The costs of the upgrades, are: $16 million for HI, PAR are about $12 million, filter vents were found to be around $25 million, and a range from $15 million to $45 million for the hardened vents.

Table 3: Final ESF results

Upgrade	BWR Regulator	BWR utility	PWR Regulator	PWR Utility
Hydrogen Igniter	0.052875	0.00225	0.329	0.014
PAR	0.0705	0.003	0.43867	0.018667
Filtered Vents	0.03384	0.00144	0.21056	0.00896
Hardened Vents ($15m)	0.0564	0.0024	0.35093	0.014933
Hardened Vents ($16m)	0.052875	0.00225	0.329	0.014
Hardened Vents ($25m)	0.03384	0.00144	0.21056	0.00896
Hardened Vents ($45m)	0.0188	0.0008	0.11698	0.0049778

To prove the SFRA, two FTs where developed, the HI and filtered venting system. The resulting ESF with and without the approximation were compared and found to add an extremely small error of around 1x10^{-9} in the filtered vents. The FT for filtered vents is shown in Figure (11) and the HI FT are shown in Figure (12) with the basic event probabilities shown in Table (4).

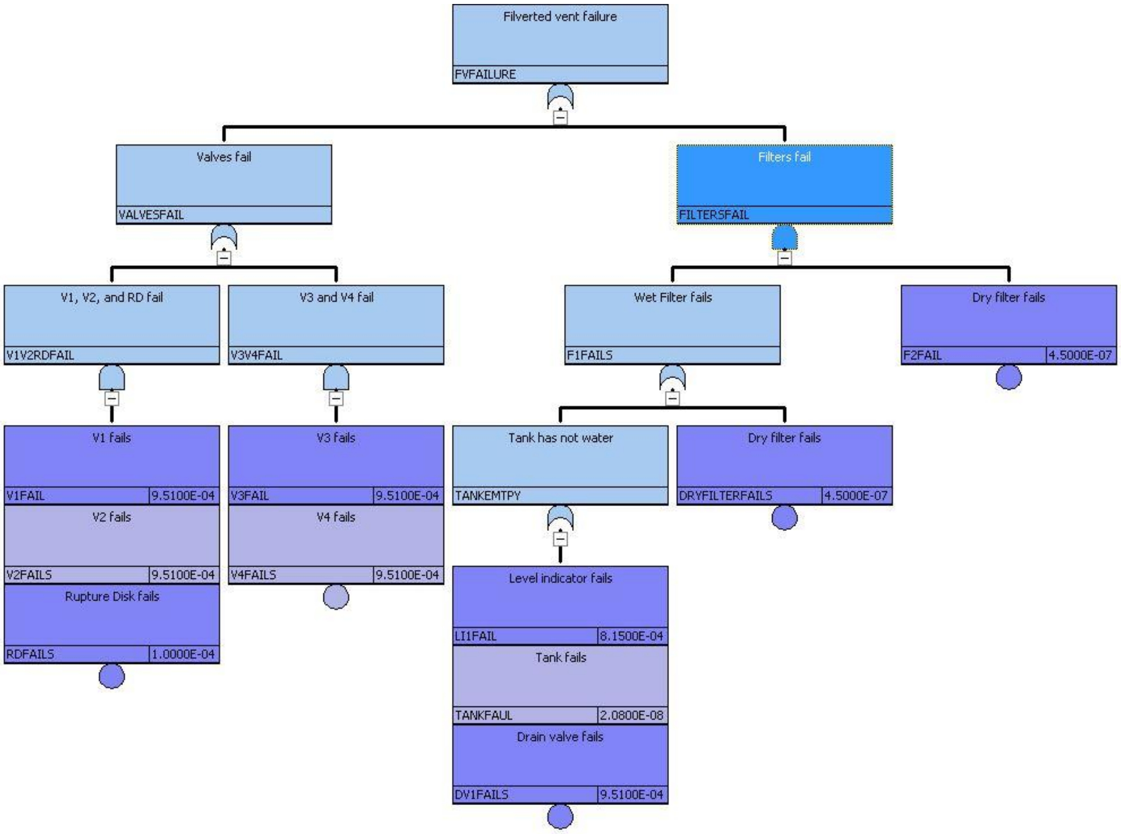

Figure 11: Filtered vent FT

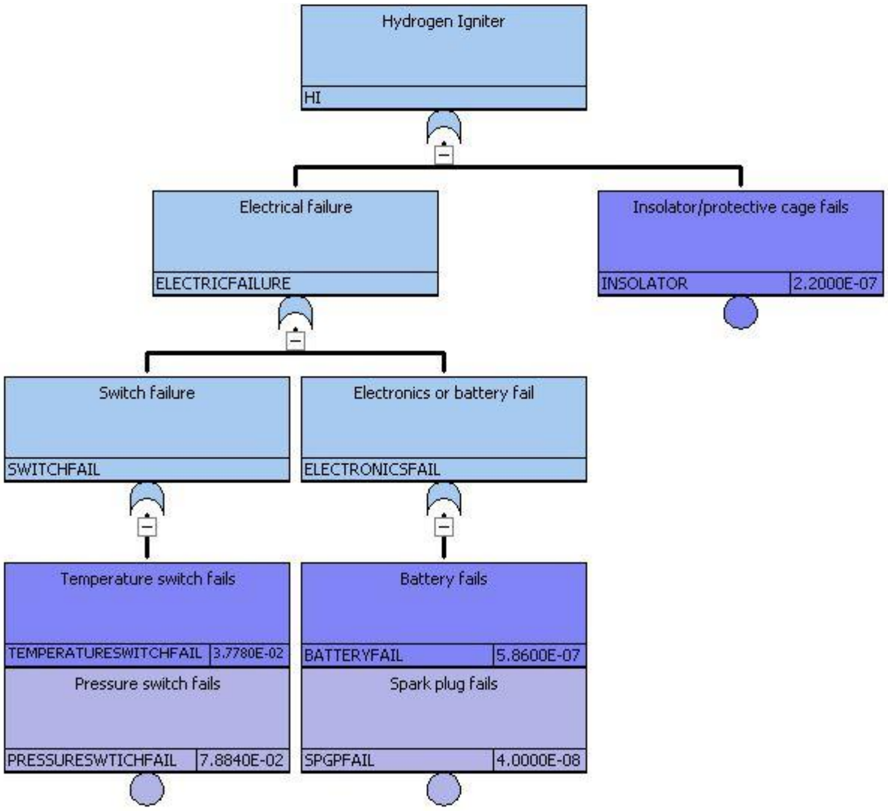

Figure 12: HI FT

Table 4: Basic event information for HI and filtered vent FTs.

Basic Event	Probability of failure	Source	Notes
Valve failure	$9.51x10^{-4}$	[59]	V1, V2, V3, V4 and drain valve
Rupture disk fails	$1x10^{-4}$	[62]	
Level indicator fails	$8.15x10^{-4}$	[59]	
Tank fails	$2.08x10^{-8}$	[59]	
Dry filter fail	$4.5x10^{-7}$	[59]	
Temp switch fails	$3.778x10^{-2}$	[61]	Average value from source
Pressure switch fail	$7.884x10^{2}$	[60]	
Battery fail	$5.86x10^{-7}$	[59]	
Spark plug fail	$4x10^{-8}$	N/A	Assumed to be ~0

The resulting probabilities of failure for the HI and filtered vent where then multiplied by the cost of a mitigated accident which is on the order of magnitude of decommissioning and the cost of a new plant which is assumed to be $10 billion. The difference between the unmitigated risk and the mitigated risk proved to be extremely small.

Figure (13) and (14) graph the PWR and BWR ESF, respectively, from the regulator and utility perspective on the same graph.

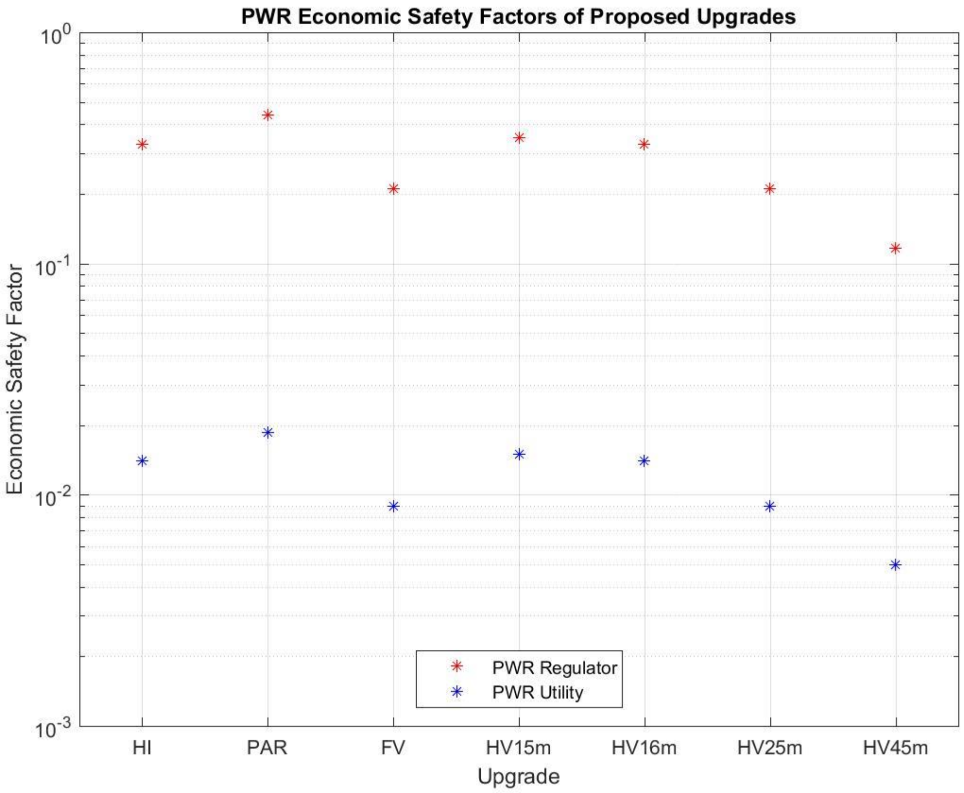

Figure 13: PWR ESF versus upgrade

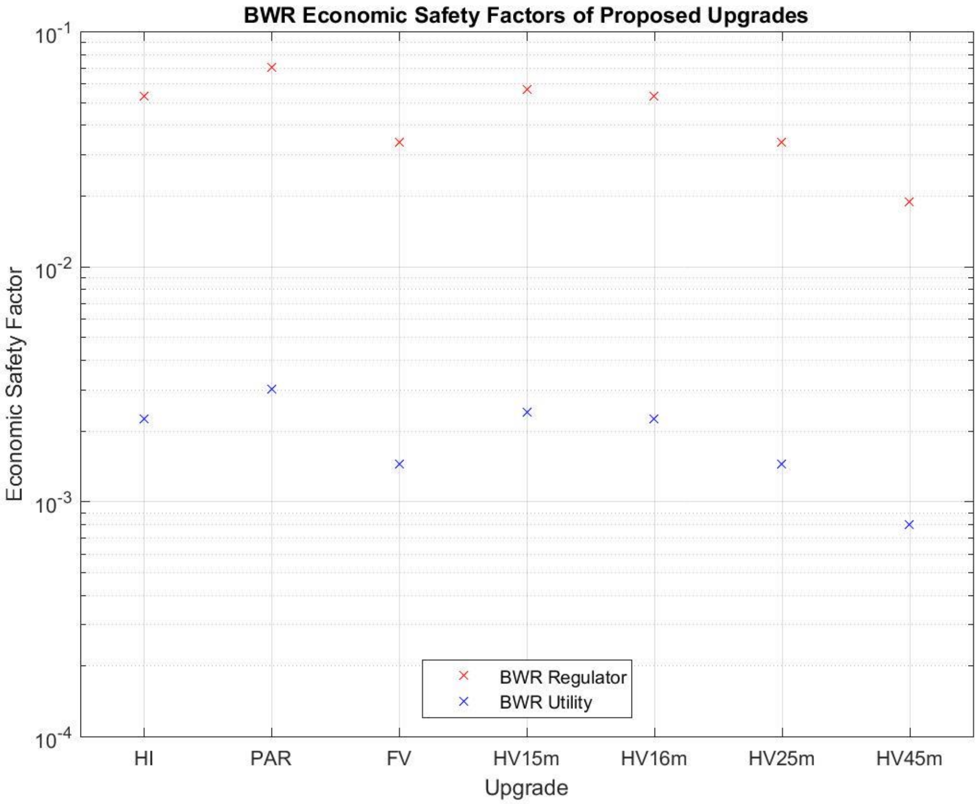

Figure 14: BWR ESF versus upgrade

From these graphs it is clear that the PWR can gain a better benefit than the BWR and that the cost of the hardened vents are dependent on the cost. The cost of a hardened vent system versus ESF is graphed in Figure (15) and has an inverse cost relationship as to be expected from Equation (13). This means as the cost approaches zero, the ESF approaches the same result.

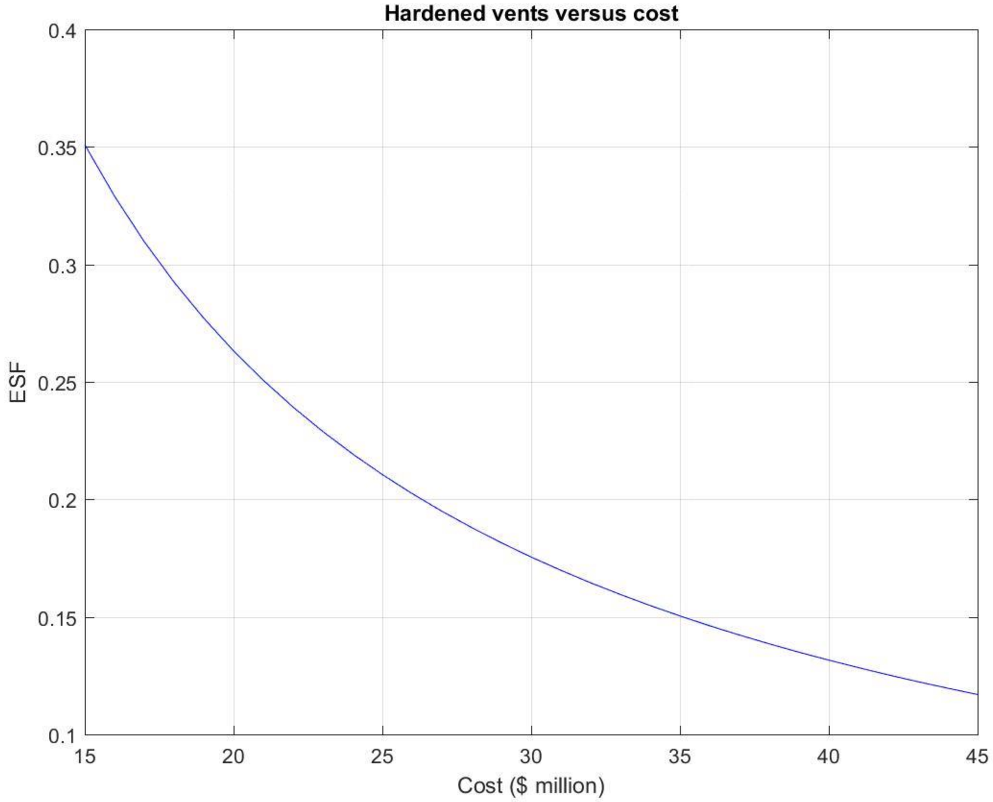

Figure 15: Hardened vent ESF versus cost of system

8.1 ESF of Proposed Upgrades

The NRC has required all Mk I. and Mk II. BWRs to implement hardened vents, as a result this is the baseline for the ESF calculations. According to this baseline, the HI and PAR ought to be implemented at all plants. However, filtered vents depends on the cost of the system. Any safety system, upgrade, or change to a NPP is specific to the plant, meaning the cost of any upgrade, safety related or not, is different between each plant. From the resulting ESF calculations, if the hardened vents are cheaper than the filtered vents then filtered vents should be implemented. If a filtered vent system costs more than a hardened venting system then the ESF will be lower resulting in a recommended decision to not implement filtered vents.

PAR and HI are a cheap and effective upgrade and are worth implementing from both a utilities and regulators perspective. This is because both upgrades have a higher ESF than a hardened vent. Even though a HI has a slightly lower ESF than the cheapest hardened vent system it is still worth implementing because of how close these two ESF are to one another that the uncertainty associated with each ESF puts them at approximately the same value.

PAR and HI are not necessarily worth the cost of implementing in large, dry PWR containments because of their large free volume. However, the NRC has required HI for ICE condenser containments because these types of containments are smaller than the other types of PWR containments, a factor that the ESF does not account for in this analysis. Another factor not taken into this analysis is the water vapor concentration versus hydrogen concentration with the HIs. Higher water vapor concentrations reduce the chance that a HI will work as is discussed in NUREG-2486, the main report on the HI experiment the NRC conducted in 1982.

8.2 Error and Uncertainty

The SFRA only adds an error of at most $1x10^{-9}$ in the final results for the filtered vents and does not change the final result nor the final decision. Thus, the SFRA holds true. This means that there are only three pieces needed to calculate the ESF of a particular upgrade given the assumptions used: the CDF, cost of the accident, and the cost of the upgrade. So long as the upgrade is proven to be reliable to within a few percent chance of failure and that the final consequence is lower than the initial consequence. If neither one of these facts holds true, the SFRA should not be used and a fuller analysis should be done. An example of when the SFRA should not be utilized would be in the analysis of high probability, small consequence accidents. Such as when a fuel rod bows due to the forces of the coolant flowing over the rod and coming into contact with another fuel rod. A safety upgrade to mitigate such an event would be to add another bracket to the fuel rod bundle.

The uncertainty associated with all of these numbers is high because of the rather limited available cost data on accidents and the upgrades themselves. The hardened vents alone have been found to have a wide range of costs. Another source of uncertainty for HIs and PARs comes from having multiple units for a single reactor, resulting in a cost that depends on the number of units and each unit adds more uncertainty to the final ESF calculation. Uncertainties such as the ones described is one of the issues in PRA [21]. The uncertainties in the SOARCA are not listed because the uncertainty analysis has yet to be done. SOARCA states that the uncertainty quantification will be published in a later report that has yet to be published.

The biggest thing to note in the HI FT analysis is the lack of the actual spark plug/glow plug failure. This is for two reasons: first these devices are extremely reliable. An assumed failure probability of $4x10^{-8}$ for the spark plugs was chosen as it is approximately an order of magnitude lower than the smallest probability of all the basic events. Spark plugs typically operating for years or decades without failure in an automotive engine. Secondly, there is no reliability data available for such devices that can be easily found. The later reason is also why the fact of the HI not working in certain steam-hydrogen concentrations was not looked at. This data is lacking and is something that can be experimented upon.

8.3 Further Discussion

One interesting thing to note is that the ESF for PWR are all much higher than that of BWRs. Take the hardened vent for example, the ESF for a BWR from the regulator perspective is at its highest is 0.0564 while a PWR regulator ESF is 0.329. This difference arises from the CDF. This is because of the lower CDF BWRs have over PWR, resulting in a higher ESF for PWRs. Meaning that each upgrade is more cost-beneficial for a PWR to implement than a BWR. Such is the case with Westinghouse's AP1000 reactor design in which they have decided to install HI even though the NRC study in 1982, NUREG-2486, showed that HI are not always affective.

8.4 Future Work

There are a number of factors not taken into account in this analysis, such as the HI's water vapor concentration, PAR placement, containment integrity analysis, and more detailed filtered vent analysis. Such things can and should be looked at when doing a full PRA on the respective upgrades. These factors were left out of the analysis in this thesis to help demonstrate the different levels of complexity one can take when using the ESF.

A HI reliability study needs to be done to fully describe the systems probability of failure and to help further decide if HI should be implemented from either the utility or regulator's perspective. The last study done on HI ability to function in a vapor/air/hydrogen environment was the NRC study in 1982. Since then, HI technology has advanced and have been more developed (in particular by Westinghouse).

A more rigorous economic analysis on the cost of accident scenarios and the cost of the upgrades themselves can also be conducted. Limited availability of such data causes large uncertainties in the ESF. Studying the costs of and how upgrades change the nature of the accident in greater detail will help reduce and define the uncertainties.

9. Conclusion

Overall, the ESF is a simple and useful tool to help a regulator, or utility decide if a particular upgrade is cost-beneficial to implement. This novel approach allows for the analysis to be simple such as the case presented in this thesis or as complicated as the theory section describes. The real challenge is finding data on reliability for components and systems, and the cost of accidents and upgrade.

Assuming that hardened vents are a baseline for whether or not an upgrade should be implemented and depending on the price of hardened vents, filtered vents can be cost beneficial. Regardless of the price of hardened vents, HI and PAR ESF is well above even the highest ESF for hardened vents so they ought to be implemented. Even though this is what the ESF concludes, there are other factors that a utility or regulator might want to consider when deciding about a potential upgrade such as social benefit and/or down time to install the upgrade that the ESF might not take into account.

Despite any short comings the ESF has, it can be a powerful and useful tool to regulators and utilities alike by offering a quick and easy metric by which to compare upgrades. By looking at multiple upgrades one can gain an idea of which upgrade offers a higher benefit.

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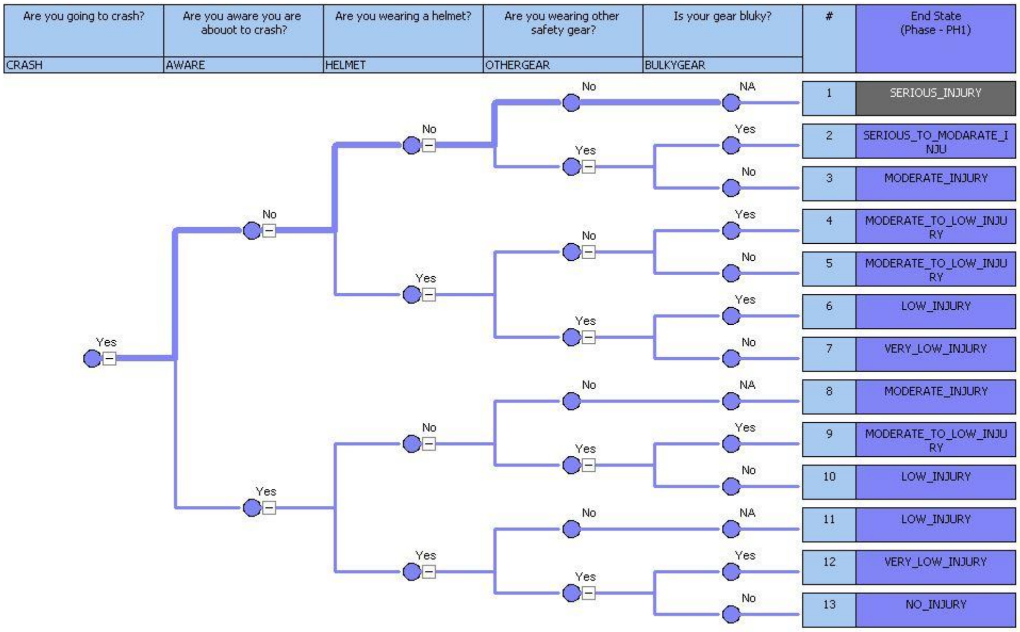
Appendices

Appendix A: BIKE Event tree and fault trees

Figure 11A is the BIKE example ET starting with the probability of crashing. Followed by multiple other events such as “Is the biker aware that they are going to crash?” and “Are they wearing a helmet?”. The resulting end states are shown. Figure 12A is the FT of whether or not a bicyclist gets into an accident of some sort. Human error and mechanical failure are the two main reasons a crash occurs. Finally, Figure 13A depicts the FT for human error, i.e. how the bicyclist fails.



#	End State (Phase - PH1)
1	SERIOUS_INJURY
2	SERIOUS_TO_MODARATE_INJU
3	MODERATE_INJURY
4	MODERATE_TO_LOW_INJURY
5	MODERATE_TO_LOW_INJURY
6	LOW_INJURY
7	VERY_LOW_INJURY
8	MODERATE_INJURY
9	MODERATE_TO_LOW_INJURY
10	LOW_INJURY
11	LOW_INJURY
12	VERY_LOW_INJURY
13	NO_INJURY

Figure A16: BIKE Event Tree

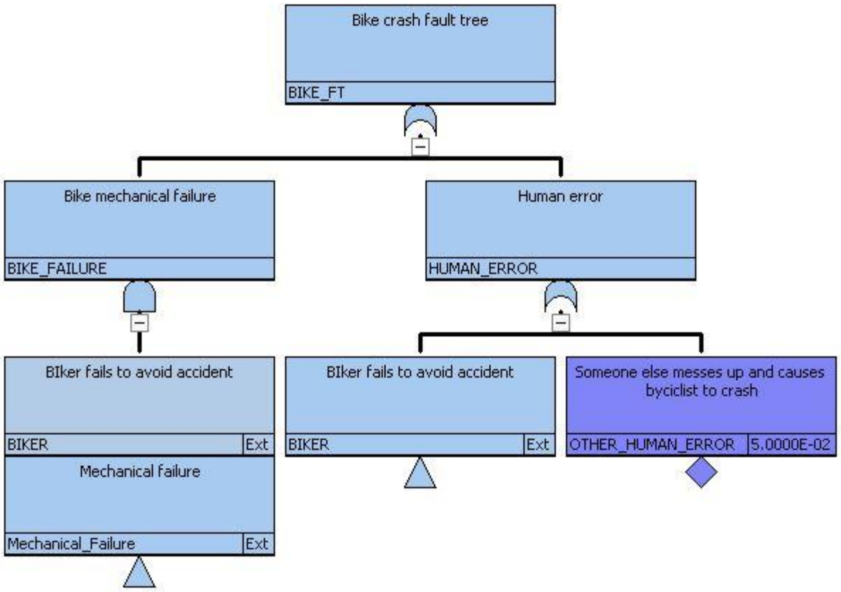


Figure A17: BIKE crash fault tree

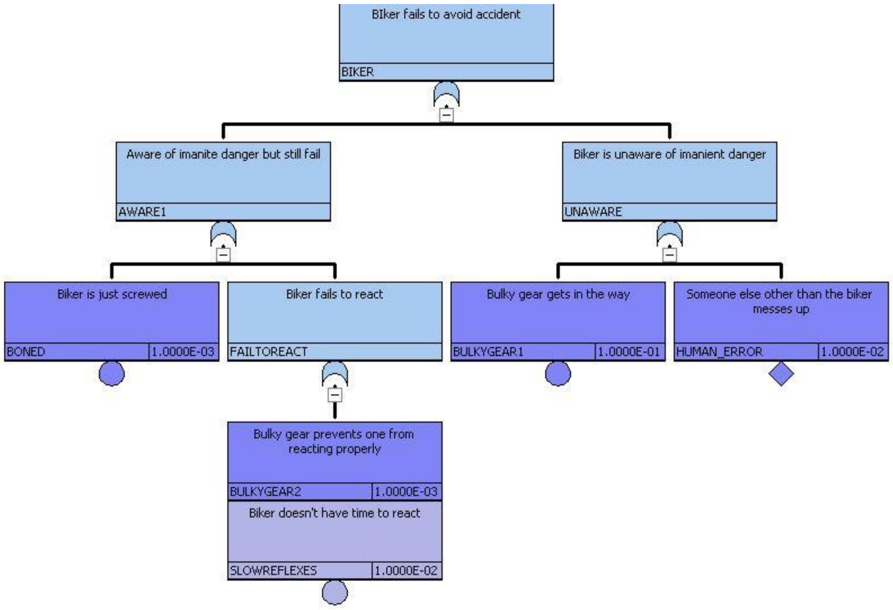

BIker fails to avoid accident
BIKER
Aware of imanite danger but still fail
AWARE1
Biker is unaware of imanient danger
UNAWARE
Biker is just screwed
BONED
1.0000E-03
Biker fails to react
FAILTOREACT
Bulky gear gets in the way
BULKYGEAR1
1.0000E-01
Someone else other than the biker messes up
HUMAN_ERROR
1.0000E-02
Bulky gear prevents one from reacting properly
BULKYGEAR2
1.0000E-03
Biker doesn't have time to react
SLOWREFLEXES
1.0000E-02

Figure A18: Biker failure fault tree

Appendix B: MatLab script for time dependent ESF

The following MatLab code was used to obtain the results in Table 3 and associated graphs.

```
clc, clear all, close all
tic
%Time dependent ESF
%Upgrade decreases burn in and constant (random) failures
%Setting up matracies and general problem stuff

tmin=24;
dt=0.001;
tmax=60;
t=tmin:dt:tmax;

%Time dependent probabilities: rho_i and rho_f

%Bath tub curve for initial
Ware_i=exp(0.2.*t)./1E11;
constant_i=ones(1,numel(t)).*4E-6;
Early_i=exp(-0.45*t)./1E6;

%Bath tub curve for final
Ware_f=Ware_i;
constant_f=ones(1,numel(t)).*2.8E-6;
Early_f=exp(-0.4.*t)./1E6;

%Probabilities of failure through time for i and f.
h_i(1,:)=Early_i+constant_i+Ware_i;
h_f(1,:)=Early_f+constant_f+Ware_f;
R_i(1,:)=((0.2/1E11).*(1-exp(-
0.2.*t)))+constant_i(1,:).*t+((0.45/1E6).*(1-exp(-0.45.*t)));
R_f(1,:)=((0.2/1E11).*(1-exp(-
0.2.*t)))+constant_f(1,:).*t+((0.4/1E6).*(1-exp(-0.4.*t)));
rho_i(1,:)=h_i.*R_i;
rho_f(1,:)=h_f.*R_f;
%Weibull distribution
l1=1E5;    %lambda of initial
k1=1.2;    %k of initial
l2=1.5E5;   %upgrade Weibull parameters
k2=1.223;
rho_i(2,:)=wblpdf(t,l1,k1);
rho_f(2,:)=wblpdf(t,l2,k2);
%Constant failure
```

```
rho_i(3,:)=ones(1,numel(t)).*4E-6;
rho_f(3,:)=ones(1,numel(t)).*2.8E-6;
%Exponential
rho_i(4,:)=6.2E-6.*exp(-0.05.*t);
rho_f(4,:)=4.3E-6.*exp(-0.04.*t);


% Time dependent consequences: gamma_i and gamma_f
%Lost productivity
interest_a=0.05; %intereste rate

% Inflation data taken directly from
http://www.usinflationcalculator.com/inflation/current-inflation-rates/
A=[2015 -0.1    0.0 -0.1    -0.2    0.0 0.1 0.2 0.2 0.0 0.2 0.5 0.7 0.1
2014    1.6 1.1 1.5 2.0 2.1 2.1 2.0 1.7 1.7 1.7 1.3 0.8 1.6
2013    1.6 2.0 1.5 1.1 1.4 1.8 2.0 1.5 1.2 1.0 1.2 1.5 1.5
2012    2.9 2.9 2.7 2.3 1.7 1.7 1.4 1.7 2.0 2.2 1.8 1.7 2.1
2011    1.6 2.1 2.7 3.2 3.6 3.6 3.6 3.8 3.9 3.5 3.4 3.0 3.2
2010    2.6 2.1 2.3 2.2 2.0 1.1 1.2 1.1 1.1 1.2 1.1 1.5 1.6
2009    0 0.2   -0.4 -0.7 -1.3 -1.4 -2.1 -1.5 -1.3 -0.2 1.8 2.7 -0.4
2008    4.3 4   4   3.9 4.2 5.0 5.6 5.4 4.9 3.7 1.1 0.1 3.8
2007    2.1 2.4 2.8 2.6 2.7 2.7 2.4 2   2.8 3.5 4.3 4.1 2.8
2006    4   3.6 3.4 3.5 4.2 4.3 4.1 3.8 2.1 1.3 2   2.5 3.2
2005    3   3   3.1 3.5 2.8 2.5 3.2 3.6 4.7 4.3 3.5 3.4 3.4
2004    1.9 1.7 1.7 2.3 3.1 3.3 3   2.7 2.5 3.2 3.5 3.3 2.7
2003    2.6 3   3   2.2 2.1 2.1 2.1 2.2 2.3 2   1.8 1.9 2.3
2002    1.1 1.1 1.5 1.6 1.2 1.1 1.5 1.8 1.5 2   2.2 2.4 1.6
2001    3.7 3.5 2.9 3.3 3.6 3.2 2.7 2.7 2.6 2.1 1.9 1.6 2.8
2000    2.7 3.2 3.8 3.1 3.2 3.7 3.7 3.4 3.5 3.4 3.4 3.4 3.4
1999    1.7 1.6 1.7 2.3 2.1 2   2.1 2.3 2.6 2.6 2.6 2.7 2.2];

interest_i=sum(A(:,end))/numel(A(:,end))/100; %inflation rate
profit=1.8E6;   %Profit from a nuclear plant per year
annuity=zeros(1,numel(t)); inflation=annuity;
MoneyLost=inflation;

for i=1:numel(t)
    if rem(t(i),1)==0; % then its an integer
        annuity(i)=1/((1+interest_a)^(t(i)-tmin+1));
        inflation(i)=1/((1+interest_i)^(t(i)-tmin+1));
        MoneyLost(i)=profit*sum(annuity)*sum(inflation);
    end
end

%Cost of decomission and accident itself
AccidentCost_i=7.5E10*sum(inflation);
AccidentCost_f=5E7*sum(inflation);
```

```
gamma_i=MoneyLost+AccidentCost_i;
gamma_f=AccidentCost_f;

% Risk_i and Risk_f

for i=1:numel(rho_i(:,2))
    Risk_i(i,:)=rho_i(i,:).*gamma_i;
    Risk_f(i,:)=rho_f(i,:).*gamma_f;
end

UpgradeCost=4.5E7*sum(inflation);

% Time integral from t=tmin to t=tmax years

for i=1:numel(Risk_i(:,1))
    for j=1:numel(Risk_i(i,:))
        Benefit(i,j)=(sum(Risk_i(i,end:-1:j))-sum(Risk_f(i,j:end)))*dt;
    end
    % Economic Safety Factor!
    ESF(i,:)=Benefit(i,:)./UpgradeCost;
end

%% Curve fit and error estimation

for i=1:numel(Risk_i(:,1))
    [x(i,:),s(i)]=polyfit(t,ESF(i,:),1);
    y1(i,:)=x(i,1).*t+x(i,2);

%     ste(i,:) = sqrt(diag(inv(s.R)*inv(s.R'))./s.normr.^2./s.df);
%     Error(i)=mean(ste(i,:))*100;
%     Error2(i)=max(abs(y1(i,:)-ESF(i,:)));
end

%% Plotting for debugging purposes

close all
te=(tmax-tmin):-dt:0;

figure (1)
subplot(2,2,1), plot(te,ESF(1,:),'r-'), grid on, hold on,
plot(te,y1(1,:),'k--')
title({'ESF versus operational time left', 'Bathtub'})
xlabel('Operational time left (years)'), ylabel('ESF')
subplot(2,2,2), plot(te,ESF(2,:),'r-'), grid on, hold on,
plot(te,y1(2,:),'k--')
title({'ESF versus operational time left', 'Weibull'})
```

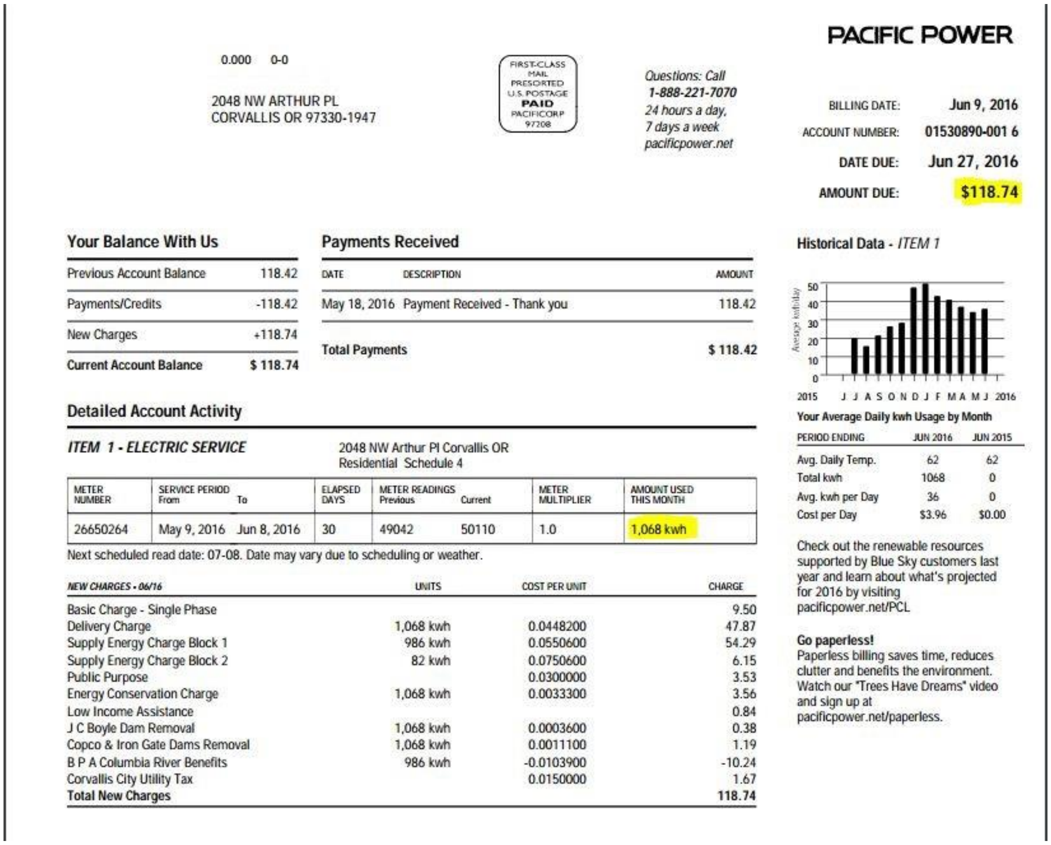
```
xlabel('Operational time left (years)'), ylabel('ESF')
subplot(2,2,3), plot(te,ESF(3,:),'r-'), grid on, hold on,
plot(te,y1(3,:),'k--')
title({'ESF versus operational time left', 'Constant'})
xlabel('Operational time left (years)'), ylabel('ESF')
subplot(2,2,4), plot(te,ESF(4,:),'r-'), grid on, hold on,
plot(te,y1(4,:),'k--')
title({'ESF versus operational time left', 'Exponential'})
xlabel('Operational time left (years)'), ylabel('ESF')
saveas(1,'ESF versus operational time left ALL','jpeg')


figure (2)
plot(t,rho_i(2,:),'b'), hold on, plot(t,rho_f(2,:),'k'), grid on
xlabel('Operational time left (t)'), ylabel('Probability of failure'),
title({'Probability V Useful life left', 'Weibull'})


toc
```

Appendix C: Consumer Power Bill

A standard consumer power bill is shown below to help calculate the net profit of a NPP. The price per kWh is $0.11 per kWh. Running a 1000 MWt NPP for one year at a 96% capacity factor would result in a gross profit of $935.0 million per year. The cost to operate a NPP is about $311.0 million per year. A difference of about $622.9 million per year or $1.8 million per day.



PACIFIC POWER

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CORVALLIS OR 97330-1947

FIRST-CLASS
MAIL
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PACIFICORP
97208

Questions: Call
1-888-221-7070
24 hours a day,
7 days a week
pacificpower.net

BILLING DATE:	Jun 9, 2016
ACCOUNT NUMBER:	01530890-001 6
DATE DUE:	Jun 27, 2016
AMOUNT DUE:	$118.74

Your Balance With Us

Previous Account Balance	118.42
Payments/Credits	-118.42
New Charges	+118.74
Current Account Balance	**$ 118.74**

Payments Received

DATE	DESCRIPTION	AMOUNT
May 18, 2016	Payment Received - Thank you	118.42
Total Payments		**$ 118.42**

Historical Data - ***ITEM 1***

Your Average Daily kwh Usage by Month

PERIOD ENDING	JUN 2016	JUN 2015
Avg. Daily Temp.	62	62
Total kwh	1068	0
Avg. kwh per Day	36	0
Cost per Day	$3.96	$0.00

Check out the renewable resources supported by Blue Sky customers last year and learn about what's projected for 2016 by visiting pacificpower.net/PCL

Go paperless!
Paperless billing saves time, reduces clutter and benefits the environment. Watch our "Trees Have Dreams" video and sign up at pacificpower.net/paperless.

Detailed Account Activity

ITEM 1 - ELECTRIC SERVICE

2048 NW Arthur Pl Corvallis OR
Residential Schedule 4

METER NUMBER	SERVICE PERIOD From	To	ELAPSED DAYS	METER READINGS Previous	Current	METER MULTIPLIER	AMOUNT USED THIS MONTH
26650264	May 9, 2016	Jun 8, 2016	30	49042	50110	1.0	1,068 kwh

Next scheduled read date: 07-08. Date may vary due to scheduling or weather.

NEW CHARGES - 06/16	UNITS	COST PER UNIT	CHARGE
Basic Charge - Single Phase			9.50
Delivery Charge	1,068 kwh	0.0448200	47.87
Supply Energy Charge Block 1	986 kwh	0.0550600	54.29
Supply Energy Charge Block 2	82 kwh	0.0750600	6.15
Public Purpose		0.0300000	3.53
Energy Conservation Charge	1,068 kwh	0.0033300	3.56
Low Income Assistance			0.84
J C Boyle Dam Removal	1,068 kwh	0.0003600	0.38
Copco & Iron Gate Dams Removal	1,068 kwh	0.0011100	1.19
B P A Columbia River Benefits	986 kwh	-0.0103900	-10.24
Corvallis City Utility Tax		0.0150000	1.67
Total New Charges			**118.74**

Figure C19: Consumer power bill