Mitigating Consequences of Flooding in a Typical Boiling Water Reactor Similar to Fukushima Daiichi Plant in Japan

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ABSTRACT

KOMBIZ SALEHI. Mitigating Consequences of Flooding in a Typical Boiling Water Reactor Similar to Fukushima Daiichi Plant in Japan. (Under the direction of Dr. Jy Wu and Dr. Zia Salami).

The Fukushima Daiichi nuclear power plant experienced a major accident on March 11, 2011, which was caused by a seismic event of 9.0 magnitude on the Richter scale followed by a tsunami wave of about 15 meters. This combination resulted in the loss of all emergency core cooling systems, including the essential diesel generators and the station batteries. The consequence of such an event was an “accident;” that resulted in multiple degraded core conditions, including hydrogen detonation causing the breach of three containments.

There is a need for a clear approach with careful attention to detail to conduct research on how to mitigate the consequences of an accident similar to the Fukushima Daiichi Nuclear Power Plant for the U.S. nuclear power plants at the coastal states. This research identified the causal factors related to flooding during the 2011 Fukushima Daiichi accident, and evaluated how the damaging consequences of such an event could be significantly reduced for a typical boiling water reactor. To achieve this purpose, the research addressed the impact of flooding that was a major cause of the Fukushima Daiichi nuclear power plant accident.

By using an analytical approach that is common in the nuclear power industry, Probabilistic Risk Assessment, the research identified safety-related systems and key components of such systems that have the potential for failure

during a flooding event. The impact of the failure of key components was examined, and how such failures could affect core damage was investigated. This research determined that a single component failure will not result in an increase in core damage. However, the research identified that if there is a loss of all AC and DC power for longer than a day, core damage may occur, even for the U.S. nuclear power plants.

The research made recommendations for both newer and older existing nuclear power plants, especially at coastal sites where Fukushima-type events, but not with the same intensity, are more likely to occur. Implementing one or more of these recommendations will significantly reduce the likelihood of a degraded core condition, even in the presence of seismic events and floodings. This research used an analytical approach to identify what went wrong at Fukushima (causes), and specify what can be done to mitigate damaging consequences (recommendations).

An essential validation of the effectiveness of the recommendations was achieved by performing a repeat analysis after implementing one or more recommendations. When the results of the analyses exhibited a larger margin in the values of core damage frequency, the verification process was completed; that is, implementing a recommendation would reduce the probability of a core damage. This process may be continued beyond the current research and into research on other systems and components after implementing more recommendations.

DEDICATION

The researcher wishes to dedicate this effort to his parents who fostered in him the value of education, setting goals, maintaining perseverance, and achieving established goals, regardless of obstacles, age, and other factors.

DISCLAIMER AND ACKNOWLEDGEMENTS

Disclaimer:

Unless otherwise indicated, opinions expressed herein are mine and do not represent the individual views of my committee members or the corporate views of the University of North Carolina at Charlotte. Kombiz Salehi, Charlotte, NC November 14, 2018

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GLOSSARY

Critical Terms

a. Two terms are used in reference to a nuclear power plant: station and plant. They are synonymous, most utilities use station and others use plant. For example, Japan uses ‘Fukushima Daiichi Nuclear Power Station’, whereas, Salem uses ‘Salem Nuclear Power Plant. They are often referenced as NPP. Within each plant or station, there is one or multiple reactors, referred to as units;, hence, Dresden Unit 1 and Dresden Unit 2. At the Fukushima Daiichi Nuclear Power Station, there are six units, Units 1-6. Further, most journals, books, and articles abbreviate ‘nuclear power plants’ as NPP, avoiding repeated reference to nuclear power plant.

b. There are two nuclear power plants with Fukushima in their names, Fukushima Daiichi and Fukushima Daini. The accident took place at the Fukushima Daiichi nuclear power station. There was no notable accident at Fukushima Daini nuclear power station, located six miles north of Fukushima Daiichi. In this manuscript, unless specified otherwise, the reference is to the Fukushima Daiichi Station.

c. In the nuclear power industry, four terms are used in order of increasing severity: ‘abnormal events’, ‘transients’, ‘incidents’, and ‘accidents’. Most abnormal events are manageable, whereas a transient is a sudden change from a normal operating condition. Examples of transients are pump trips, generator trips or unanticipated reactor shutdowns (referred to as reactor scrams). An incident is more severe than a transient and can be associated with some degree of radiation leakage, mostly within the plant itself. Accidents are the most serious, such as ‘loss of coolant accident’, or a ‘degraded core condition or accident’.

d. In general an ‘event’ refers to any occurrence that is noticeable by plant operating personnel. It can be used in reference to an accident, an incident, a transient, earthquakes or seismic activities, and floods.

e. Core damage frequency (CDF) refers to the occurrence of core damage based on some frequency (time cycle) signifying, at minimum, damage to the nuclear fuel. The term is expressed in units of a reactor core damage per reactor year.

f. Core geometry is a reference to the portion of the core structures containing the fuel assemblies, nuclear instrumentation and control rods. In an accident condition when the fuel assemblies become extremely hot due to inadequate core cooling, the neighboring core support structures can deform, thereby changing the core geometry. When core geometry changes, water flow to the neighboring fuel assemblies can be restricted, thus causing more fuel damage and more changes in core geometry.

g. Damages to core geometry occurred during the three worst ‘commercial’ nuclear power plant accidents: Three Mile Island, Chernobyl and at three of the six Fukushima Daiichi reactors. “Commercial’ distinguishes fully operational nuclear power plants from test reactors, where core geometry accidents have also occurred (SL-1, for example).

h. Emergency Core Cooling Systems

Emergency core cooling systems (ECCS) comprise numerous systems that are designed to provide core cooling in order to prevent degraded core condition. The breakdown of various systems within the ECCS are as follows:

i. High-pressure core spray system (HPCS)
ii. High-pressure coolant injection system (HPCI)
iii. Low-pressure core spray system (LPCS)
iv. Low-pressure coolant injection system (LPCI)
v. Shutdown cooling system (SDC)
vi. Residual heat removal system provides LPCI, SDC, & containment cooling
vii. Reactor core isolation cooling system (RCIC)
viii. Automatic depressurization system (ADS) has the safety relief valves (SRV)
ix. Diesel generators. The units provide AC power to the systems within ECCS.

Other Common Terms

10 CFR:

The segment of the U.S. Code of Federal Regulation (CFR) that pertains to Energy (10). It establishes rules and regulations for any facility that uses special nuclear material (SNM).

BOILING WATER REACTOR:

In the boiling water reactor (BWR), water passes over the reactor core to slow down the neutrons and acts as a coolant and is also the steam source for the turbine, which in turn powers the generator to produce electricity.

COLD SHUTDOWN:

A reactor coolant system at atmospheric pressure and at temperatures below 200 degrees Fahrenheit following a reactor cool down.

CONTAINMENT:

A gas-tight shell or other enclosure around a nuclear reactor to confine fission products that otherwise might be released into the atmosphere during an accident. Such enclosures are usually dome-shaped and made of steel-reinforced concrete. In boiling water reactors there are two types of containments, primary and secondary. The reactor pressure vessel is inside the primary containment, whereas, the secondary containment houses the reactor building.

CORE:

The central portion of a nuclear reactor, which contains the fuel assemblies, moderator, neutron absorbers or control rods, and support structures holding the fuel assemblies and nuclear power instrumentation. The fission process takes place inside the core.

FUEL ROD AND FUEL ASSEMBLIES:

A long, slender, zirconium metal tube containing pellets of fissionable material, normally ceramic uranium dioxide, which provide fuel for the nuclear reactors. Fuel assemblies in a boiling water reactor consist of fuel rods placed in an 8X8 or 9X9 (earlier 7X7) arrangement held together by spacers made with zirconium alloy. Various sizes of reactors have different numbers of fuel assemblies, usually more than several hundred assemblies in a single reactor.

FUKUSHIMA DAIICHI ACCIDENT:

Explosions, fire, and failed cooling systems caused by a seismic magnitude 9.0 earthquake and a 14-meter tsunami wave in this Japanese nuclear power station on March 11, 2011, resulted in several core meltdowns and a containment breach, releasing radioactivity directly into the atmosphere and the sea.

HALF-LIFE:

The time required for half of the amount of a substance disintegrate. Cobalt 56 isotope has a half-life of 6.6 days. Plutonium 239, produced in a reactor, has a half-life of 24,110 years.

INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA):

The center of worldwide cooperation in the nuclear field, through member countries and multiple international partners. The United Nations established the IAEA in 1957 as part of the "Atoms for Peace" program.

MELTDOWN:

The melting of a significant portion of a nuclear-reactor core due to inadequate cooling of the fuel elements, a condition that could lead to the escape of radiation.

MIXED OXIDE:

Fuel assemblies that contain fuel rods made with uranium oxide pellets and plutonium oxide pellets in a special configuration in an assembly. Because fuel assemblies contain fuel pellets made with plutonium, they are specially treated and handled, from the time they are manufactured to the time they are inserted into a reactor pressure vessel.

PLUTONIUM:

A heavy, radioactive, man-made metallic element with atomic number 94. Its most important isotope is fissile plutonium-239, which is produced by neutron irradiation of uranium-238 (making >96% of the uranium fuel) in reactors.

PRESSURIZED WATER REACTOR:

In the pressurized water reactor (PWR), the water which passes over the reactor core to act as moderator and coolant does not flow to the turbine, and it is contained in a pressurized primary loop. The steam, which drives the turbine is produced in a separate vessel, a secondary loop, is called steam generator. The steam generators are often as large as the reactor pressure vessel.

PRA:

It stands for ‘Probabilistic Risk Assessment’, it uses risk analysis to determine a quantifiable value for a specific event to determine the probability of a specific event, including a seismic event, a loss of cooling accident, or a any adverse event within a nuclear power plant.

RADIATION:

Can consist of alpha particles, beta particles, gamma rays, x-rays, neutrons, high-speed electrons, high-speed protons, and other particles.

REACTOR PRESSURE VESSEL:

RPV is the vessel containing the reactor and other material that produces energy in a boiling water or pressurized water reactor.

RESEARCHER

The author of this manuscript who conducted this research and created this manuscript.

SAPHIRE:

It stands for System Analysis Program for Hands-on Integrated Evaluation, and is a software program used to calculate probabilistic risk assessments.

SPENT FUEL POOL:

An underwater storage and cooling facility for spent (depleted) fuel assemblies that have been removed from a reactor.

STANDBY:

The status of a system or a reactor that is ready to start or run. Run condition is a reference to the status of a reactor in power mode. In a boiling water reactor it is a synonymous term called ‘startup. The possible modes are shut down or refueling mode, startup or standby mode, and run or power mode.

URANIUM:

A radioactive element with the atomic number 92 and, as found in natural ores, has an atomic weight of approximately 238. A particular kind of uranium, U-235, is used as fuel in nuclear power plants for fission because its atoms are easily split apart.

Acronyms and Abbreviations

In addition to the terms defined above, the nuclear power industry uses many acronyms and abbreviations which are also used in this manuscript.

AC	Alternating current
AEC	Atomic Energy Commission (former name for the U.S. NRC)
AND gate	An AND gate used in FTA
BE	Basic Event (used in PRA)
BWR	Boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CAFTA	Computer Aided Fault Tree Analysis
CDF	Core Damage Frequency
CFR	Code of Federal Regulation
CST	Condenser Storage Tank
DBA/E	Design Basis Accident/Event
DC	Direct Current
DG or D/G	Diesel Generators
DID	Defense-in-depth
DOE	U.S. Department of Energy
DS	Automatic depressurization system
D/W	Drywell
EPG	Emergency Procedures Guidelines
EOP	Emergency Operating Procedures
ETA	Event Tree Analysis
ERDA	Energy Research and Development Association (former DOE)
FTA	Fault Tree Analysis

GE	General Electric Company
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
IC	Isolation Condenser
IE	Initiating Event (used in PRA)
I&C	Instrumentation and Control
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LUHS	Loss of Ultimate Heat Sink
MAAP	Modular Accident Analysis Program developed by EPRI
MITI	Ministry of International Trade Industry in Japan
NISA	Nuclear and Industrial Safety Agency in Japan
NPP	Nuclear power plant
NPS	Nuclear power station
NRA	Nuclear Regulation Agency in Japan
NRC/U.S. NRC	Nuclear Regulatory Commission
OR gate	An OR gate (used in PRA)
PCV	Primary containment vessel
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
R/B	Reactor building
RCIC	Reactor core isolation cooling system
RHR	Residual heat removal system
RO	Reactor Operator license
RPS	Reactor Protection System
RPV	Reactor pressure vessel
SBLC	Standby liquid control system
SBO	Station Blackout (Loss of all AC power)
S/C	Suppression chamber
SDC	Shut down cooling system
SGTS	Standby Gas Treatment System
S/P	Suppression Pool or Taurus
SPDS	Safety Parameter Display System
SRM	Source Range Monitor

SRV	Safety relief valve
TE	Top Event (used in PRA)
Taurus	Torroidal shaped suppression pool in BWRs
W/W	Wet well (suppression pool)

CHAPTER ONE – INTRODUCTION

1.1. Background

The accident that occurred in Japan at the Fukushima Daiichi plant on March 11, 2011, was one of the three most severe accidents in the history of commercial nuclear power plant operations. Unlike two earlier accidents (Three Mile Island and Chernobyl), the Fukushima Daiichi accident was initiated by unstoppable natural processes, i.e., the Great East Japan earthquake with 9.0 magnitude and associated tsunami waves. This 2011 accident has received tremendous attention, research, and coverage, and as a consequence, this effort needed to be different from those performed by others. The initial effort was to compile and review pertinent written documents encompassing other Ph.D. dissertations, journal articles, newspaper articles, official reports, videos, and books on this subject. After a comprehensive literature review, knowledge gaps were identified and were used to define the research goals. The researcher examined safety-related systems and within each system identified key components for analysis. Next, the researcher undertook an analytic approach using an established software tool, System Analysis Program for Hands-on Integrated Evaluation (SAPHIRE), to determine failure frequency of components for each system. This software then determined the frequency of core damage as a consequence of a failure of key components. If the failure of a component was found to reduce the margin of safety, the researcher recommended changes to mitigate the reduction in safety margin when a component fails.

The research focused its effort on three principal parts of the Fukushima Daiichinuclear power station accident, which are described in detail in the "Objectives" section of this paper. The first part was identifying the general causes of the accident. The second part was the role of human error by the reactor supplier, the utility and the regulators. The third part was the specific effects of flooding that caused the most damage and severe consequences for the plant. After examining the three principal parts of the accident, a probabilistic risk assessment was completed, which led to the recommendations in a series of alternative approaches. These recommendations, if implemented, would help mitigate the adverse consequences of a possible similar accident for at a typical coastal U.S. nuclear power plant. To verify the effectiveness of a recommendation, the research analyzes results before and after a recommendation is implemented and compares the results on the computed probability of failure. These results reveal whether or not a recommendation will improve the safety margin. Such recommendations have not been previously addressed by other researchers

1.2. Objectives

To support the overall purpose to improve nuclear power plant operations by reducing the risk of damages from flooding, this research accomplished five distinct objectives:

1. Identify the general causal factors that contributed to the accident and extensive damage to the Fukushima Daiichi nuclear power station.

2. Identify the human errors that took place during the design, operation, construction, supply, operation, and regulation of the Fukushima Daiichi nuclear power station.

3. Identify the specific impacts of flooding on a typical boiling water reactor at a coastal location of the U.S.

4. Assess the risk of damages from such an event (flooding) and perform probabilistic risk assessment using fault tree and event tree analyses for various key components of 11 safety-related systems, including emergency core cooling systems.

5. Develop a set of recommendations to mitigate the severity of the adverse damages from of a similar accident, at U.S. coastal nuclear power plants when failures of a system or component may result in a higher probability of core damage frequency.

1.3. Problem statement

There is a potential for U.S. nuclear power stations located at coastal sites to suffer similar consequences as those experienced by the Fukushima Daiichi nuclear power station in Japan. That event caused over $630,000,000,000[1] in damages [1] and destroyed all six units at that plant. There have been three major accidents at commercial nuclear power plants around the world: Three Mile Island in Pennsylvania in 1979, Chernobyl in the Ukraine in 1986, and triggered by a natural event, an earthquake of magnitude 9.0 on the Richter scale Fukushima Daiichi in Japan in 2011.

[1] There are numerous estimates given for the cost of the accident ranging from $100 billions to $670 Billion. The variation is largely a function of the time of the estimates and the underlying assumptions.

All three accidents were exacerbated by human errors, but the last one was in a nearby subduction zone (the convergence boundary between two tectonic plates) deep below the Pacific Ocean. The conditions were further aggravated by the resulting tsunami waves that reached the shore. In this instance, the tsunami waves hitting that part of Japan were as high as 14 meters and occurred within 50 minutes of the earthquake.[2]

The Fukushima Daiichi nuclear power station accident is relevant to the U.S. nuclear power plants since there are currently 15 coastal nuclear power plants operating in the U.S.[3] Thus, there is an urgent need to ascertain whether these plants are subject to major floods that would result in similar outcomes as occurred in Japan. This research will investigate how to mitigate the adverse consequences of such events in the U.S.

1.4. Unique aspects of this research

There have been many analyses and research reports on the Fukushima Daiichi nuclear power station discussing needed initiatives to mitigate or preclude events with similar consequences. This research is different than other efforts in three ways as described below:

A) This research uses an analytical approach involving a probabilistic risk assessment and statistical analysis to determine risks associated with operating nuclear power plants in the U.S. under adverse conditions similar to those experienced by the Fukushima Daiichi nuclear power station. In the probabilistic risk assessment (PRA) analysis, the risks of failure are investigated to determine the risk of failure when a primary system or key standby component fails to perform its intended function. Backup systems and components are designed to function in cases of primary failure. All nuclear power plants are designed such that a single failure will not invalidate the functionality and operability of a system. These facts are considered within a PRA. The use of probabilistic risk assessment is widespread in the nuclear industry, and it is universally used during the design phase in calculations of risks due to a seismic event. However, the use of probabilistic risk assessment for key components and of safety-related systems as a forensics tool after a severe accident is the key element that distinguishes this research from others.

B) This research analyzes the failure of the Fukushima Daiichi nuclear power station on a component-by-component basis for critical components of selected safety-related systems. Of all the analyses performed to date to investigate this accident, only one performed investigations on a component-by-component basis, and that was performed by the Japanese government for the Japanese plants. Their analysis did not apply to representative nuclear power plants in the U.S. This research will apply the results from the Fukushima to a representative U.S. boiling water reactor located at coastal sites.

C) The researcher is able to exercise his domain knowledge for the benefit of this study in enabling component level analysis. Current recommendations established by the U.S. Nuclear Regulatory Commission (NRC) provide generic requirements and allow the licensees to evaluate and determine if their plants are vulnerable to core damage. In this research, the evaluation of failure is by an independent individual, a researcher, who brings a unique perspective to the project. He has a Senior Reactor Operator (SRO) license on an identical nuclear power plant identical to the Fukushima Daiichi nuclear power station, the Quad Cities Nuclear Power Station in the State of Illinois. Both the Quad Cities nuclear power station and two of the units in the Fukushima Daiichi nuclear power station were designed and manufactured around the same time by the General Electric Company. The remaining four units at the Fukushima Nuclear power station were designed by Toshiba and Hitachi, where both of these firms replicated the General Electric design. The researcher will interject his own professional experiences and interpretations into the research effort, when appropriate.

D) After obtaining the results of the failure analysis of this research for the Fukushima Daiichi station, and identifying risks, the next phase was to create a new set of recommendations. The current recommendations established by the U.S. Nuclear Regulatory Commission (NRC) provide generic requirements and allow the licensees to evaluate and determine if their plants are vulnerable to core damage. In this research, the evaluation of failure is by an independent individual, one who has an operating license on an identical nuclear power plant as the Fukushima Daiichi nuclear power station, and who is not a staff member of a licensee.

The research analyzed various safety-related systems and their key components. The process was to initially collect data and technical information followed by reviews of relevant data from prior written work. The research analyzed the sequence of events in the Fukushima Daiichi station and identified the causal factors affecting the failures of various key components of safety-related systems for that station.

1.5. Scope of work of conducting this research

The accident that occurred in Japan at the Fukushima Daiichi plant on March 11, 2011, was one of the three most severe accidents in the history of commercial nuclear power plant operations around the world. Unlike the earlier two accidents (Three Mile Island and Chernobyl), the Fukushima Daiichi accident was initiated by unstoppable natural processes, i.e., the Great East Japan earthquake with 9.0 magnitude and associated tsunami waves. The researcher was compelled to contribute to finding a solution to the problem of the Fukushima Daiichi accident by dedicating this research to this cause. This 2011 accident has received tremendous attention, research, and coverage, and as a consequence, this effort needed to be different than those performed by others. The initial effort was to evaluate and compile pertinent written documents encompassing other Ph.D. dissertations, journal articles, newspaper articles, official reports, videos, and books on this subject. After a comprehensive literature review, knowledge gaps were identified that were identified, which were used to define the research goals. The researcher examined safety-related systems and within each system identified key components for analysis. Next, the researcher undertook an analytic approach using an established software tool, System Analysis Program for Hands-on Integrated Evaluation (SAPHIRE), to determine failure frequency of components for each system. This software determined the frequency of core damage as a consequence of a failure of key components. If the failure of a component was found to reduce the margin of safety, the researcher recommended changes that could be made to mitigate the reduction in safety margin when a component fails. The research focused its effort on three principal parts of the Fukushima Daiichinuclear power station accident. The first part was identifying the causes of the accident. Without identifying a cause, a problem never gets solved. The second part was to identify the human error by entities involved in the design, construction, engineering, the reactor supplier, as well as the operating utility and the regulators. The third part was the flooding that created the most damage and caused severe consequences to the plant. After examining the three principal parts of the accident, this dissertation recommends a series of alternatives to help mitigate the adverse consequences of a similar accident for a typical coastal U.S. nuclear power plant. These recommendations are additional measures that have not been previously addressed by other researchers.

To verify the effectiveness of the recommendation, the research analyzes results before and after a recommendation and compares the results of the computed probability of failure, assuming a recommendation was implemented. The results from the repeated analyses reveals whether the safety margin remains the same or is improved.

The Fukushima Daiichi nuclear power station accident is relevant to the U.S. nuclear power plants as there are important lessons learned from this accident that are applicable to the U.S. nuclear power plants. Since there are currently 15 coastal nuclear power plants operating in the US, [3] there is an urgent need to ascertain whether these plants are subject to major floods that would result in the same or similar outcomes as occurred in Japan. This is the problem investigated here, and this research will investigate how the consequences can be mitigated.

The research analyzed various safety-related systems and their key components. The process was to initially collect data and technical information followed by reviews of relevant data from prior written work. The research analyzed the sequence of events in the Fukushima Daiichi station and identified the causal factors affecting the failures of various key components of safety-related systems for that station.

The objectives of this research will be accomplished by completing the work tasks described below.

A) Prior to the actual performance of the research, the first objective of identifying the general causal factors of this accident must be clearly was accomplished. Fully understanding the causes of the accident at the Fukushima Daiichi station would require several books, given the large scope and the multiple causes of the damage. Already there have been many articles written by various authors around the globe discussing the Fukushima Daiichi accident. This manuscript focuses on the causal factors associated with human activities and by the tsunami induced flooding. It is worth noting that the flooding at this plant was caused by multiple tsunami waves that carried millions of gallons of water into the plant, thereby, submerging parts of the buildings and many pieces of equipment. There will be more on this subject in Chapter 5 of this dissertation.

The categories of general causes that severely and catastrophically damaged the Fukushima Daiichi nuclear power station are listed below as follows:

- The flood's dynamic force destroyed the electrical grid.
- The flood caused critical emergency diesel generators to fail to function.
- The flood also caused the station batteries and battery chargers to fail.
- Dynamic force of flood waters damaged equipment causing major equipment to remain non-functional, including safety-related systems that are critical parts of the emergency core cooling systems.
- Without power and emergency core cooling systems, 4 of 6 reactor cores were damaged.
- The resulting core damage released hydrogen that detonated.
- Hydrogen detonation caused several containments to be breached.
- The breach of the containments allowed radioactive material releases.
- Electric power was not restored until a week later.
- Total normal AC power was not fully restored for about six months.
- It appears the plant will remain closed for a long time, and is closed as of this writing.

B) After identifying the general causal factors of this the accident, then the research began by collecting data, focusing on key causes factors that had not been previously addressed. Chapter 2 details the Fukushima Daiichi nuclear power station and compares to a typical U.S. boiling water reactor. Chapter 3 is allocated to literature review.

C) The second objective of investigating the effects of human factors was accomplished as described in Chapter 4.

D) The third objective of investigating the specific effects of flooding was accomplished as described in Chapter 5.

E) The fourth objective of performing probabilistic risk analyses was accomplished as described in Chapter 6.

F) The fifth objective of making recommendations and verifying the effectiveness of each recommendation was accomplished as described in Chapters 7 and 8, pursuing the necessary steps of the research.

Note: This research does not examine the earthquake that initiated the tsunami. The plant was designed, as required, to sustain such and specified earthquake, and had it not been for the seismically induced tsunami waves, the plant would not have suffered as such. In fact, numerous claims have been made that the plant did not suffer major consequence as a result of the earthquake. Further, there is no realistic way to circumvent an earthquake that exceeds the reasonably interpreted historical record, and earthquakes are often experienced in that part of the world, referred to as the 'Ring of Fire'. However, this manuscript focuses on the flooding as that constitutes the main cause of the accident that occurred at the Fukushima Daiichi nuclear power station. The reason this research did not focus on the earthquake is because: a) earthquakes are unpreventable, and plant designers must take this factor into consideration when they locate the plant; and b) they must design the systems and components of the systems at least to withstand the most credible seismic event that has happened to that location during the existing historical record.

1.6 Restrictions and limitations of the research

This section lists the limitations and restrictions relating to this research. There are two categories of limitations and restrictions that apply to the Fukushima and U.S. nuclear power plants.

1.6.1 Plant systems

There are restrictions to the data used in this research relating to the systems and components of the nuclear power plants that were evaluated. Clearly, it is unnecessary to evaluate all systems' responses and their failure mechanisms under the accident conditions in order to meet the research objectives of this dissertataion. In a typical boiling water reactor type plant, there are thousands [4] of valves, thousands of pumps, and motors. Thus,It is unnecessary to evaluate the entire set of components of this type of a plant. This research only focused on those components of the safety-related systems that were essential for providing core cooling and containment integrity. For this purpose, there were over 100 components that were considered in this research.

1.6.2 Duration of the accident

The onset of the accident began with the seismic event. The focus of the research was to evaluate the sequence of events for the first four weeks. It took about three weeks before power was restored to the plant through an unusual process, and about six months before outside power using the electrical grid was reconnected. The consequences of the accident are still on-going, eight years after the accident. The latest challenge is a question about the effectiveness of the freezing wall around the reactor buildings of the four damaged reactors. As late as January 1, 2017, Japan's Nuclear Regulatory Authority questioned the effectiveness of the frozen wall and it is pushing more towards pumping the radioactive and contaminated water from the bottom of the affected reactor buildings.[5] Tokyo Electric Power Company (TEPCO) stated that they would double their pumping capacity of the contaminated water by the end of 2017.[5]

CHAPTER TWO – BACKGROUND INFORMATION ON THE FUKUSHIMA DAIICHI NUCLEAR POWER STATION AND NUCLEAR POWER PLANTS

This chapter provides an overview of the essential features of the Fukushima Daiichi nuclear power station and selected U.S. boiling water reactor located at coastal sites. This chapter also presents other pertinent information related to the hydrogen issue and data collection.

2.1. Fukushima Daiichi nuclear power station

The Fukushima Daiichi station is located in the Futaba District of Ottozawa in the Fukushima prefecture.[6,7] as shown in Figure 1.

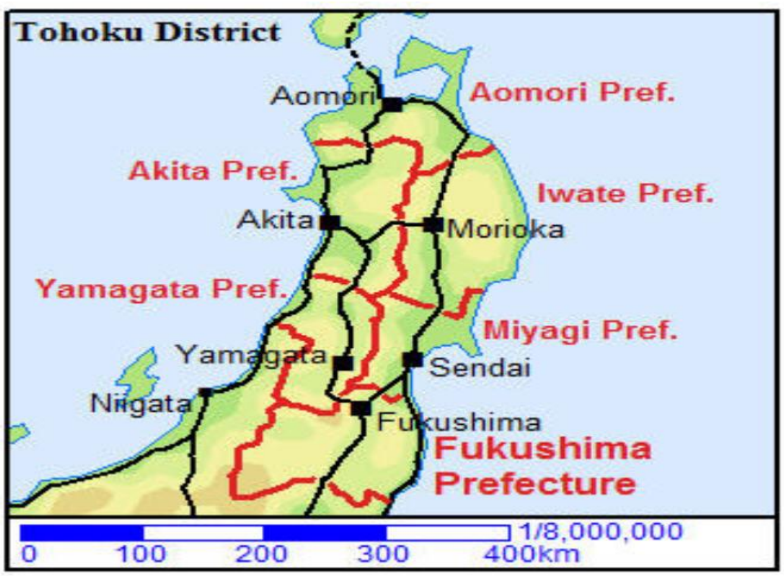

Figure 1. The location of the Fukushima and neighboring prefectures [7]

There are six nuclear reactors, or units, in the Fukushima Daiichi nuclear power station, as shown in (Figure 2), [7] which identifies the on-site locations of reactors and the epicenter of the seismic event on March 11, 2011. The major components of a safety-related system for any nuclear power plant, includes pumps, valves, motors, piping, power supply, instrumentation, water supply containers, and system logic diagrams. A system may have more than one component, in order to provide a backup and redundancy capability for that system. Some systems remain in standby when a reactor is in operation.

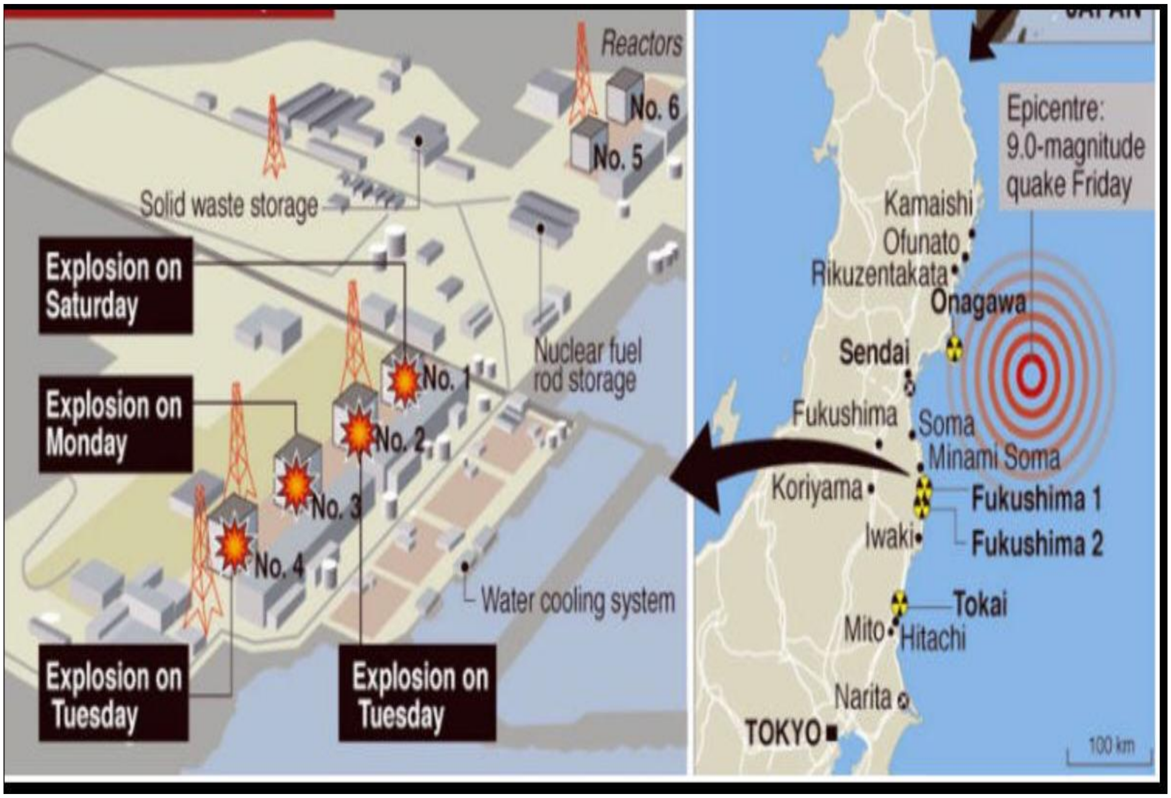

Figure 2. The location of the 6 units and the epicenter of the earthquake [6]

Table 1 provides a summary [1], [2] of the essential features for the six units at the Fukushima Daiichi plant.[6]

[1]. The table makes a reference to NSSS. It stands for Nuclear Steam Supply System, which is a reference to the supplier of the reactor, and key components. For the Boiling Water Reactor (BWR), that is a reference to GE. For the Pressurized Water Reactors (PWR), it could be the reference to Westinghouse, Babcock and Wilcox, Combustion Engineering to name a few.

[2] AE is a term meaning the architect engineering firms. Most utilities recruit an AE firm to design their plant. Aside from selecting the type of the reactor, the first major effort by a utility is to select an AE firm. This is because the site selection and generation of the Preliminary Safety Analysis Report is normally within the scope of work of an AE firm. The most prominent AE firms are Bechtel Power, Seargent and Lundy, and Brown and Root. Ebasco was a major AE firm in the 1970s and early 80s. However, in 1993 it was sold to Raytheon.

Table 1. Various parameters for the Fukushima Daiichi units [6]

Unit	Type (Containment)	Net Power IN MW	Start of Construction	First Criticality	Start of Commercial Operation	Date of Shutdown	NSSS[2]	AE[3]
1	BWR-3 (MARK 1)	439	7/25/67	10/10/70	3/26/71	5/19/2011	GE	Ebasco
2	BWR-4 (MARK 1)	760	6/9/69	5/10/73	7/18/74	5/19/2011	GE	Ebasco
3	BWR-4 (MARK 1)	760	12/28/70	9/6/74	3/27/76	5/19/2011	Toshiba	Toshiba
4	BWR-4 (MARK 1)	760	2/12/73	1/28/78	10/12/78	5/19/2011	Hitachi	Hitachi
5	BWR-4 (MARK 1)	760	5/22/72	8/26/77	4/18/78	12/17/2013	Toshiba	Toshiba
6	BWR-5 (MARK 1)	1067	10/26/73	3/9/79	10/24/79	12/17/2013	GE	Ebasco

2.1.1. Characterization of the Fukushima Daiichi Accident

The accident that occurred began with a seismic event On 11 March 2011, the tsunami waves generated by the Great East Japan Earthquake off the coast of Japan, overwhelming the plant's tsunami barriers. The resulting flood water flooded the safety-related systems and components of the six units at the site. This event compounded the off-site power loss that was created as a consequence of the earthquake damage to the electrical transmission system. The flooding caused the loss of critical emergency diesel generators designed to provide electrical AC power for the safety-related and emergency core cooling systems. Units 1–5 of the Fukushima Daiichi nuclear power station experienced extended station blackout (SBO), which exceeded nine days in Units 1 and 2, and 14 days in Units 3 and 4. [8]. The reactors were not designed to function without power, AC and DC and as a consequence suffered a great deal of damage to the reactor cores for unit 1, 2, and 3 resulting in the breach of the reactor pressure vessel containing these reactors. Without cooling, the fuel rods and fuel assemblies failed releasing hydrogen into containment and becoming accumulated. Without establishing hydrogen mitigation system, as required for the boiling water reactors, the accumulation of hydrogen in the containments created a blowout of their containments. This evolution created a path for the release of radioactive fission products from the several barriers. The first barrier is the fuel rod cladding. The second barrier is the water in the reactor pressure vessel, which was not present or very little. The third barrier is the reactor pressure vessel, which was breached. The fourth barrier is the drywell, and the fifth being the reactor building containment engulfing the drywell and then the reactor pressure vessel.

In the absence of these barriers, the radioactive nuclides were released into the environment, contaminating the nearby areas causing massive evacuation of the people close by. Appendix A provides a detailed sequence of events.

2.1.2. Fukushima Daiichi station safety-related systems

At the Fukushima Daiichi nuclear power station, some are classified as safety-related and others are non-safety-related. All systems support the operation of a reactor and support other systems necessary to maintain operability of the units. Safety-related systems support the reactor core, reactor pressure vessel, and primary and secondary containments. The components within the systems include mechanical, electrical, auxiliary and instrumentation to control and provide adequate information to the plant operating personnel. For example, the diesel generators are safety-related systems, including the generator, diesel oil pumps, and fuel tanks. The Q-List at a station identifies all structures, systems and components covered by the plant's quality assurance (QA) program.

The Q-list contains safety-related equipment and components that are necessary to ensure:

- the integrity of the reactor coolant pressure boundary,
- the capability to shut down the reactor and maintain it in a cold shutdown condition, and
- the capability to prevent or mitigate the consequences of an accident, which could result in potential off-site exposures comparable to those specified in 10 CFR 100.[3]

If a component is not on the Q-list, it is not considered a safety-related item. The design, procurement, installation, testing, replacement, and changes in safety-related systems and components must fall within the plant's quality assurance (QA) program. For instance, if a component on the Q-list fails, there must be a formal analysis to identify the root cause of the failure, prior to its replacement. The replacement part must come from a vendor or a supplier that is qualified and certified as a nuclear grade component supplier. If a nuclear grade component cannot be obtained, a generic or a commercial component must be evaluated to meet the quality and performance level of a nuclear grade

[3] 10 CFR 100 is part of the Code of Federal Regulation (CFR) pertaining to siting criteria for nuclear power plants. Subpart A relates to factors for nuclear power plant site applications prior to January 10, 1997 and for test reactors. Subpart B is for site applications after that date.

component. These and many other clauses and conditions are invoked in 10 CFR 50[4] and in the plants' quality assurance program. Not meeting this criterion causes a licensee to be in non-compliance with the requirements. Such non-compliances will lead to findings by the NRC inspectors, subject to issuance of notice of violations (NOV), which can carry fines and other compensatory measures. There are provisions within the NRC rules and regulations that in cases of willful, gross negligence, and fraudulent exercises, the NRC can impose incarceration for the person committing such an act. The NRC has seldom imposed incarceration as all the licensees' senior management and executives are keenly aware of the severity of the consequences for any of their employees and staff committing an act necessitating incarceration. The researcher has conducted numerous inspections and has written more than 50 notices of violations. However, there was no issue requiring such a penalty for an employee of a licensee. There have been a few cases where the NRC has recommended removing a senior member of a licensee from being involved in the decision making process regarding a nuclear power plant. This scenario is highly political and takes place behind closed doors at the highest level. The safety and security of a nuclear power plant override all other considerations, including financial concerns. This point is being made to emphasize that, in the opinion of the researcher, such power and provisions were not influential in the Japanese regulatory process.

[4] 10 CFR 50 is part of the Code of Federal Regulation pertaining to domestic licensing and production and utilization of facilities.

2.1.3. Reactor protection system

One of the key safety related systems that protects the reactor and the plant is the reactor protection system. This system causes an automatic insertion of the control rods into the reactor core, shutting down the fission process. The concurrent insertion of the control rods, into the reactor core takes place at levels ranging from zero power to rated power in a few seconds. Most of the reactor trips take place automatically. However, the reactor shut down, or reactor scram, as it is called, can also take place manually by an operator or a senior reactor operator. The operating personnel are keenly aware of the process after the reactor shutdown and closely monitor reactor conditions to ascertain that the reactor meets the expected thermal hydraulic parameters after a reactor shutdown. A reactor scram is not at all a routine event and all operating staff pay special attention to monitor plant systems for the reactor response. Noncritical personnel are asked to leave the control room by the shift supervisor on duty, as it is important that such an event be monitored diligently and carefully. There are many alarms and annunciations on the horizontal and vertical panels of the main control room. To understand the types of signals generated by the reactor protection system of the Fukushima Daiichi nuclear power station, the following subsections identify and discuss various safety-related types of signals producing a reactor scram.

After a reactor shutdown, there is decay heat generated in a reactor as various fission products decay by gamma and neutron emissions, providing energy to the fuel and the fuel cladding. Right after reactor shutdown, 6.5% of thermal power continues to emanate from the fuel assemblies.[9] This source of heat gradually reduces to nearly zero, about 7-10 days after the reactor shutdown (Figure 3). This period of decay heat had a pivotal impact on the Fukushima Daiichi nuclear power station accident and on the recommendations from this research effort.

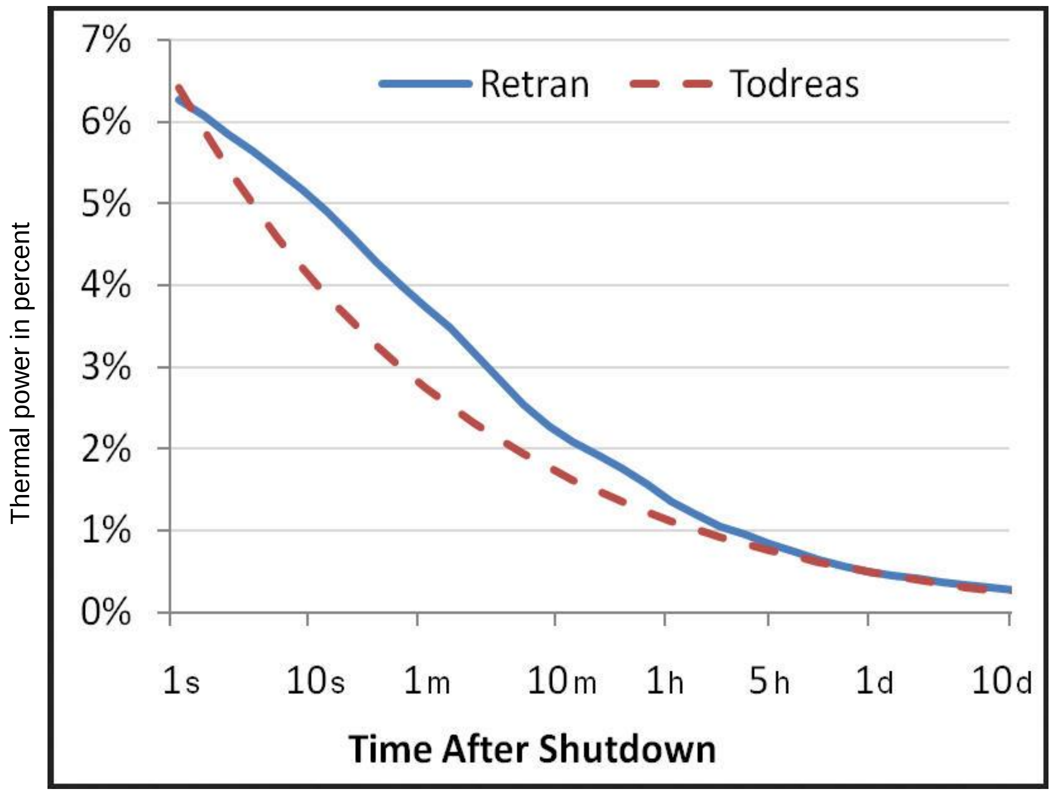

Figure 3 Decay heat in power versus time [10]

Figure 3 displays a typical curve for decay heat plotted using two models, the Retran model and Todreas-Kazimi model [5].[10] During the seven days, there must be cooling water added to the reactor pressure vessel to remove the decay heat. The lack of heat removal causes the fuel temperature to rise. In the

[5] Two models are displayed, Retran, which uses an 11 group exponential decay, the other being Todreas and Kazimi from published work in 1990, adapted from previous empirical correlations. The Retran model inputs no assumptions of operating history, and the Todreas model requires the operating time before shutdown.

absence of cooling, the heat can expand the cladding, potentially leading to its rupture, thereby, allowing fission products to escape to the reactor coolant and ultimately to the plant and the outside environment. During the design and manufacturing of the fuel assemblies, all steps and design features are incorporated to preclude such an outcome. For this reason, there exist multiple redundant systems, pumps, valves, reactor pressure vessel entry points, water sources, and electrical power sources to assure coolant injection into the reactor pressure vessel under high and low pressure conditions.

Reactor scram refers to a sudden shut down of the reactor by either the reactor protection system or by manual activation. When a reactor scrams, all control rods containing poisons, mostly boron or other type material, which absorb thermal neutrons, are inserted into the core and that stops the chain reaction. There are many conditions that initiate a scram, most of which result in automatic shutdown beyond the control of the reactor operator. Examples are: low water level in the reactor pressure vessel, a turbine trip or generator trip, a drywell sump level increase, (which signifies water leakage from the reactor pressure vessel), under-voltage on the electrical grid, increases in the containment temperature or pressure, and sensing a low voltage by the electrical buses on the reactor protection system. All of these causes of scrams are controlled by the reactor protection system. The Fukushima Daiichi reactors have one more protective measure that most US plants do not. For the Japanese nuclear power plants, if a seismic monitor gets tripped due to a seismic event, the Japanese plants scram. On March 11, 2011, all nuclear power plants in Japan, including the Fukushima plants were shut down because of sensing a seismic signal. In addition to the automatic activation of the reactor protection system shutting down the reactor, an operator or a shift supervisor in the control room can manually shut down a reactor by pressing two scram buttons simultaneously. This occurs when the operator senses the reactor is approaching an unsafe condition. There are over 100 independent control rods that are inserted by hydraulic (for boiling water reactors) and gravity (for pressurized water reactors) into the core shutting down the reactor. In the Fukushima Daiichi nuclear power station there were 97 control rods in unit 1, 137 in units 2 through 5, and 185 in unit 6 [11] that moved into the reactor core by the hydraulic pressure. In a boiling water reactor, control rods are inserted from the bottom of the reactor pressure vessel, whereas, in pressurized water reactors, the control rods are driven from the top of the reactor pressure vessel.

There is a backup to all the control rods in both boiling water reactors and pressurized water reactors, called the standby liquid control system. If the control rods do not insert into the core, thus shutting down the reactor, the standby liquid control system consisting of a tank containing sodium pentaborate solution can be injected into the reactor pressure vessel to shut down the nuclear fission process. There are two explosive-activated valves in this system that will actuate and open, allowing the system pump to inject this solution into the reactor pressure vessel from its own entry point into the vessel. Since, this is a highly undesirable event, it is almost never used. The system is tested monthly by cycling the solution to its solution tank in a recycling mode. The neutron economy[6] is important in the operation of a nuclear reactor, and the injection of the solution will profoundly affect the neutron economy, even after its cleanup. There are many signals that cause the reactor protection system to scram a reactor, both in boiling water reactors and pressurized water reactors. For the Fukushima Daiichi nuclear power station, the list of the scram signals is introduced (and numbered sequentially) below:

2.1.3.1. Closure of turbine stop or control valves

1. If the turbine protection system detects a large deviation signal, it stops the main steam flow by closing the main turbine stop valves. Because of the closure of the steam valves, the water level drops, and in anticipation of dropping the water level, the reactor scrams. Not only does the water level drop, but this allows reactivity to increase due to the collapse of the voids, or steam voids. To prevent such a transient, the reactor scrams first. This will halt the admission of steam and water to the turbines.

2. A similar transient can take place when a generator load rejection[7] occurs for the generator and the electrical connections. This transient will cause the closure of turbine control valves and trips the reactor protection system causing a scram.

[6] Neutron economy is a term that implies neutron accountability is crucial to maintain a sustained chained reaction in a reactor. If too many neutrons are lost or leak from a reactor, the reactor can lose chain reaction and require more fuel to be used to sustain the same thermal power oputput.
[7] Generator load rejection is a reference signifying the loss of electrical power availability when the output of the main generator cannot be loaded into the grid.

2.1.3.2. Loss of off-site power

Under normal conditions, the reactor protection system is powered by off-site power. Loss of off-site power opens all relays in the reactor protection system, which trips the reactor causing a reactor scram. It must be noted that loss of off-site power also closes all the main steam isolation valves, thus stopping the steam flow. Closure of one or more of the main steam isolation valves can signify the potential for a break in the main steam lines. To prevent loss of steam inventory from the reactor pressure vessel, the main steam isolation valves close.

2.1.3.3. Power monitoring trips

A reactor operates in several modes or conditions. These modes are cold shutdown or refueling, standby or startup, and power or run. The reactor cores and the reactor pressure vessel in these modes are at different power levels. The reactor starts from a cold condition and ascends to standby or startup condition, and then to power condition. There are different power levels associated with each of these modes and reactor protection is assured in each of these modes. The reactor protection system receives signals for shutdown at each of these reactor conditions with different set points. The starting point is in the source range with its own protection set points. The next is startup, and finally the power or run condition. In power mode, power increases from a few percent to 100% of rated power. The purpose of these trips is to ensure the power level in the reactor is within an acceptable range in various reactor operating modes. The operating modes are a function of reactor power levels in the reactor's ascension to full power. In power range, the trips in the source range and intermediate range are bypassed as they do not apply.

2.1.3.4. Low reactor water level

1. Loss of coolant accident conditions

2. A loss of feedwater caused by a trip on feedwater pumps signifies interruption of the normal make up of water to the reactor through the reactor feedwater pumps. Feedwater pumps provide cooling to the reactor when the steam is being removed. A trip of the feedwater pump would allow the water level in the reactor pressure vessel to drop, and to prevent a substantial drop of water level, the reactor protection system provides the protection and scrams the reactor. Normally, there are three feed pumps for a single reactor, two in operation and one in standby. In case one pump trips, the one in standby comes on line and injects water into the reactor pressure vessel.

2.1.3.5. High reactor water level (in boiling water reactor /6 plants)

1. This feature is not provided in earlier boiling water reactors, including the Fukushima plant. It is covered because, in some boiling water reactors, this trip exists and needs to be covered. For boiling water reactor /6 plants, it prevents flooding of the main steam lines protecting the plants' main steam turbines.

2. Limits the rate of cold water addition to the vessel, thus limiting reactor power increase during over-feed of coolant to the reactor pressure vessel transients.

2.1.3.6. High drywell (primary containment) pressure

1. This signal, if genuine, could be an indication of a potential loss of coolant accident in the primary containment.

2. This signal also begins the emergency core cooling systems to prepare spraying the primary containment after the spray permissive is cleared.

2.1.3.7. Main steam isolation valve closure

1. Protects from a pressure transient in the core, causing a reactivity transient.
2. Only triggers for each channel when the valve is greater than 8% closed.
3. One valve may be closed without causing a scram.

2.1.3.8. High-pressure in the reactor pressure vessel

1. Indicative of main steam isolation valve closure.
2. Decreases reactivity to make up for void collapse due to high pressure.
3. Prevents pressure relief valves from opening.
4. Serves as a backup for several other trips; e.g., turbine trip.

2.1.3.9. Low-pressure in the reactor pressure vessel

1. Indicative of a potential line break in the steam tunnel or other location, which does not trigger high drywell pressure.

2. Bypassed when the reactor is not in the run mode to allow for pressurization and cool down without a scram signal.

2.1.3.10. Seismic event

1. Generally, only plants in high seismic areas have this trip enabled.

2. All Japanese plants have this scram enabled, and on March 11, 2011, all nuclear power plants in Japan scrammed.

2.1.3.11. High discharge volume

In case the scram hydraulic discharge volume begins to fill up, this will scram the reactor prior to the discharge volume becoming totally full. This prevents hydraulic lock, which could prevent the control rods from inserting into the reactor pressure vessel and into the reactor core.

2.1.4. Safety-related systems at the Fukushima Daiichi nuclear power station

In this section, there is a reference to 11 safety-related systems that must remain functional at the Fukushima Daiichi nuclear power station, in order to prevent core damage. Within these 11 systems, are the diesel generators, the station batteries and battery chargers, the reactor protection system which trips the reactor automatically under specific conditions, and the safety relief valves. It is relevant and important to discuss just the eight different systems that make up the emergency core cooling systems showing their system diagram. However, for the full analysis, all 11 systems have been considered, including the diesel generators, the batteries and the safety relief valves. Several of these emergency core cooling systems are backups to another system in the emergency core cooling systems. For example, to reduce pressure in the reactor pressure vessel, there is a high-pressure core injection system and its backup is the reactor core isolation cooling system. Further, there are safety relief valves designed to reduce pressure in the reactor pressure vessel. Once the reactor pressure is reduced, there are low pressure core cooling systems, such as low-pressure core spray system, low-pressure coolant injection system, and shutdown cooling system to inject water into the reactor pressure vessel to cool the fuel assemblies. The shutdown cooling system, is part of the residual heat removal system, and can assist the reactor core isolation cooling system to handle the decay heat after the reactor is shut down.

The systems are listed and discussed sequentially beginning with the most important collective systems grouped together and referred to as emergency core cooling systems (ECCS) (Figure 4) [12]. Most authors and personnel in the nuclear industry refer to them as ECCS, a combination of systems that are used in an emergency condition, designed with the sole purpose of protecting the safety of the reactor under those emergency conditions.

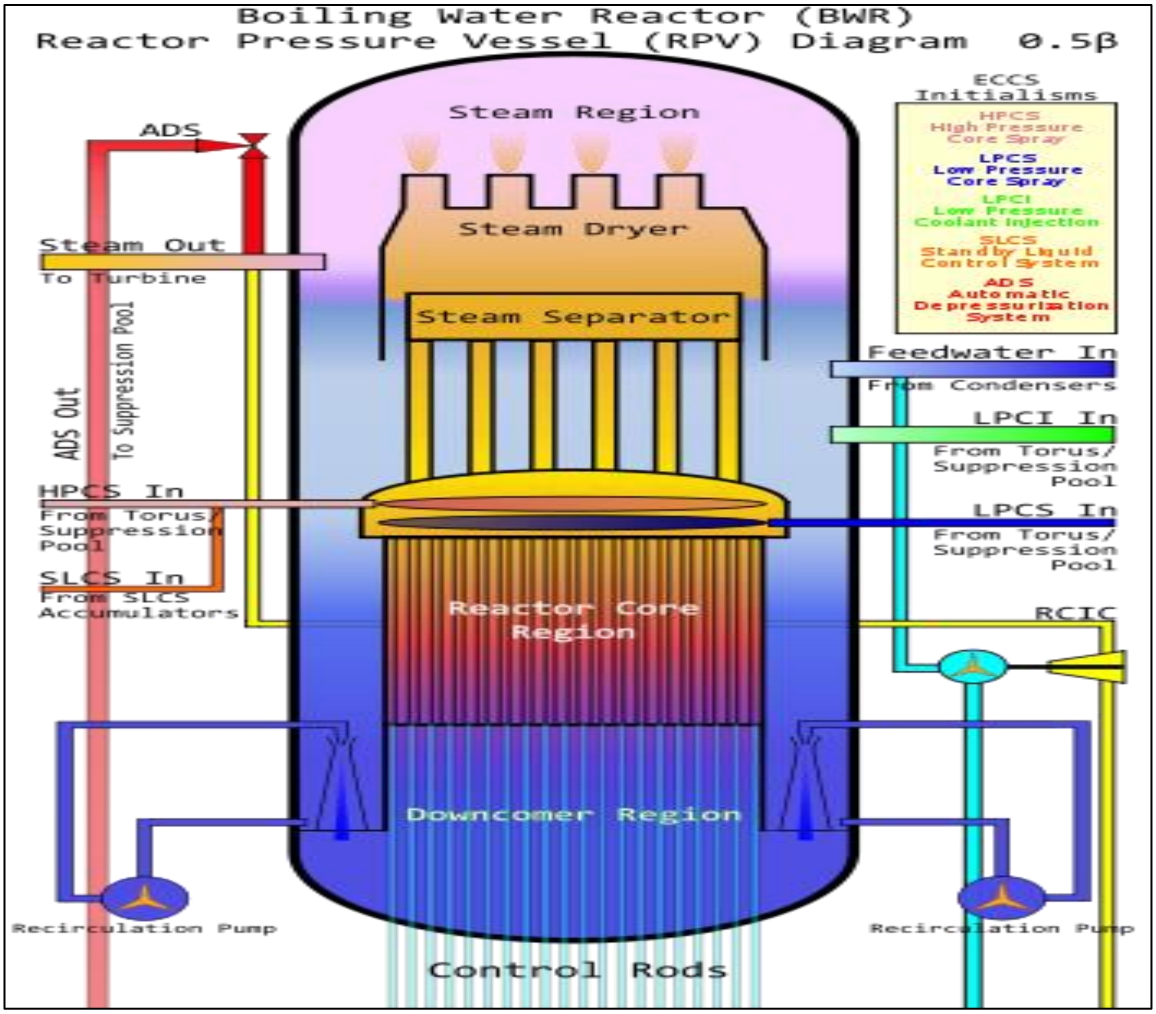

Figure 4. Safety-related systems in a boiling water reactor [12]

2.1.4.1. Emergency core cooling systems

The emergency core cooling systems consist of numerous high and low pressure core cooling systems. The overall purpose of the emergency core cooling systems is to provide core cooling during an emergency and accident conditions to prevent core damage. Depending on the type of a reactor, there are different systems. The high-pressure systems include high-pressure coolant injection, high-pressure core spray, and an automatic depressurization system. Low-pressure systems include low-pressure cooling injection and low-pressure core spray. There are also reactor core isolation cooling system, isolation condenser, and residual heat removal system, which remove the decay heat after a reactor shutdown. Reactor core isolation cooling system helps reduce reactor pressure from full power, and during brief interval when there is no AC power to activate the remaining systems within the emergency core cooling systems. The shutdown cooling system is part of the residual heat removal that supports handling the decay heat, but requires AC power. Grouped with these systems, are the diesel generators providing AC power and the station batteries and battery chargers that provide DC power for the high-pressure coolant injection and reactor core isolation cooling systems.

Figure 5 illustrates[13] various systems within the emergency core cooling systems and their key components.[13] It also depicts three divisions and each division represents its own logic and electric power supply in order to maintain independence. The drawing is for a BWR/4 plant (Shoreham nuclear power station to be specific), and there are differences between this boiling water reactor and the Fukushima Daiichi reactor. Most boiling water reactors maintain two divisions, and the Shoreham nuclear power station opted to have three different divisions. This nuclear power plant has only one unit. The figure provides the electrical distribution with three different diesel generators. The electrical distribution is set up such that in an emergency condition, the power is obtained from their normal station services transformer (NSST), followed by a reserve station services transformer (RSST) and when both were unavailable, the next is an automatic switch to a divisional diesel generator. These terms and the setup are unique to the Shoreham plant only. SW is a reference to service water system, and CRD is a reference to the control rod drive system. HX or HTX is a reference to a heat exchanger. MO is a reference to motor-operated valve, whereas, AO is a reference to air-operated valve. The air-operated valve requires air supply from the plant's air compressors, which require AC power.

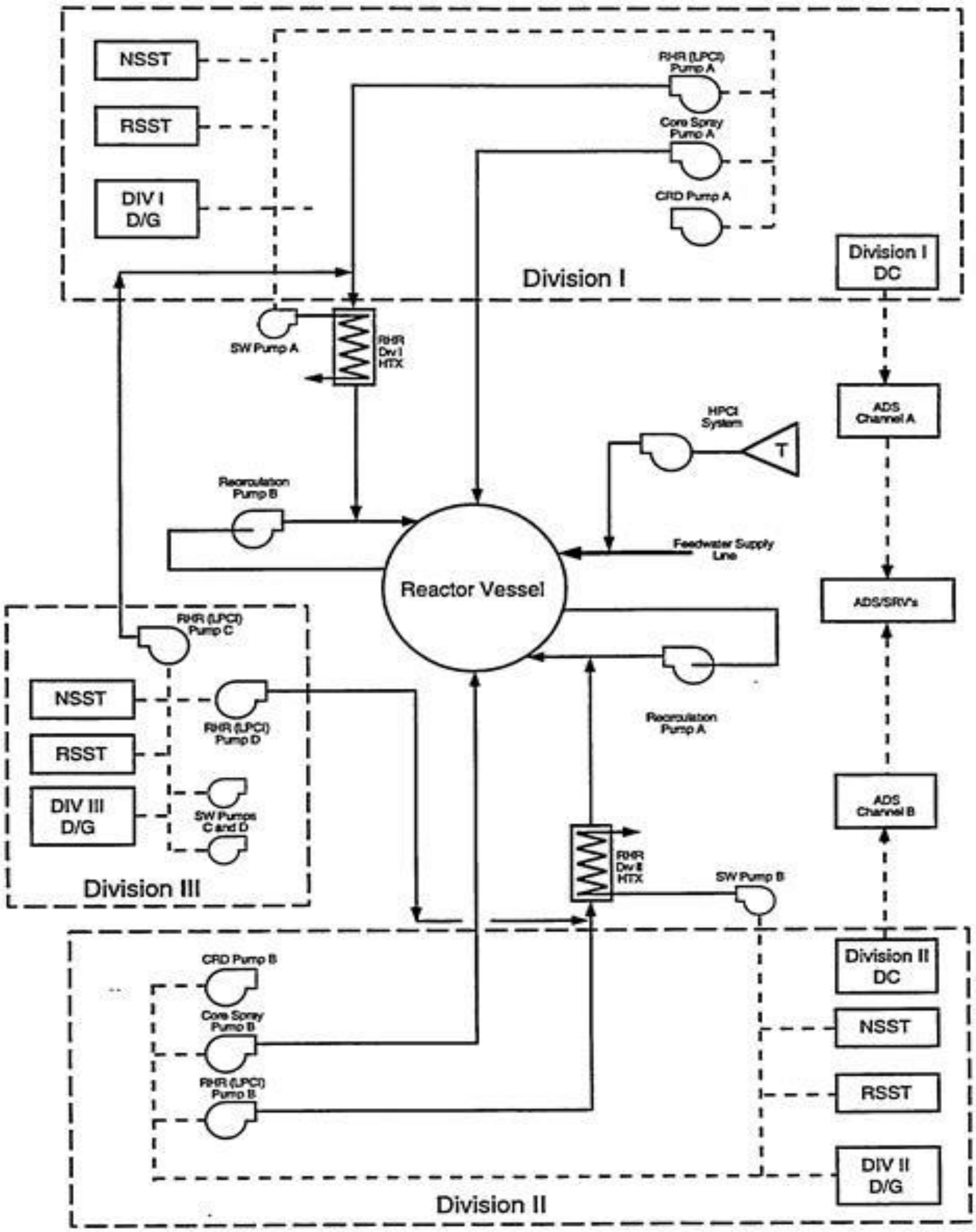

Figure 5. Emergency core cooling systems for a BWR/4 plant [13]

Figure 6 illustrates [13] a composite of various systems including the automatic depressurization system within the emergency core cooling systems.

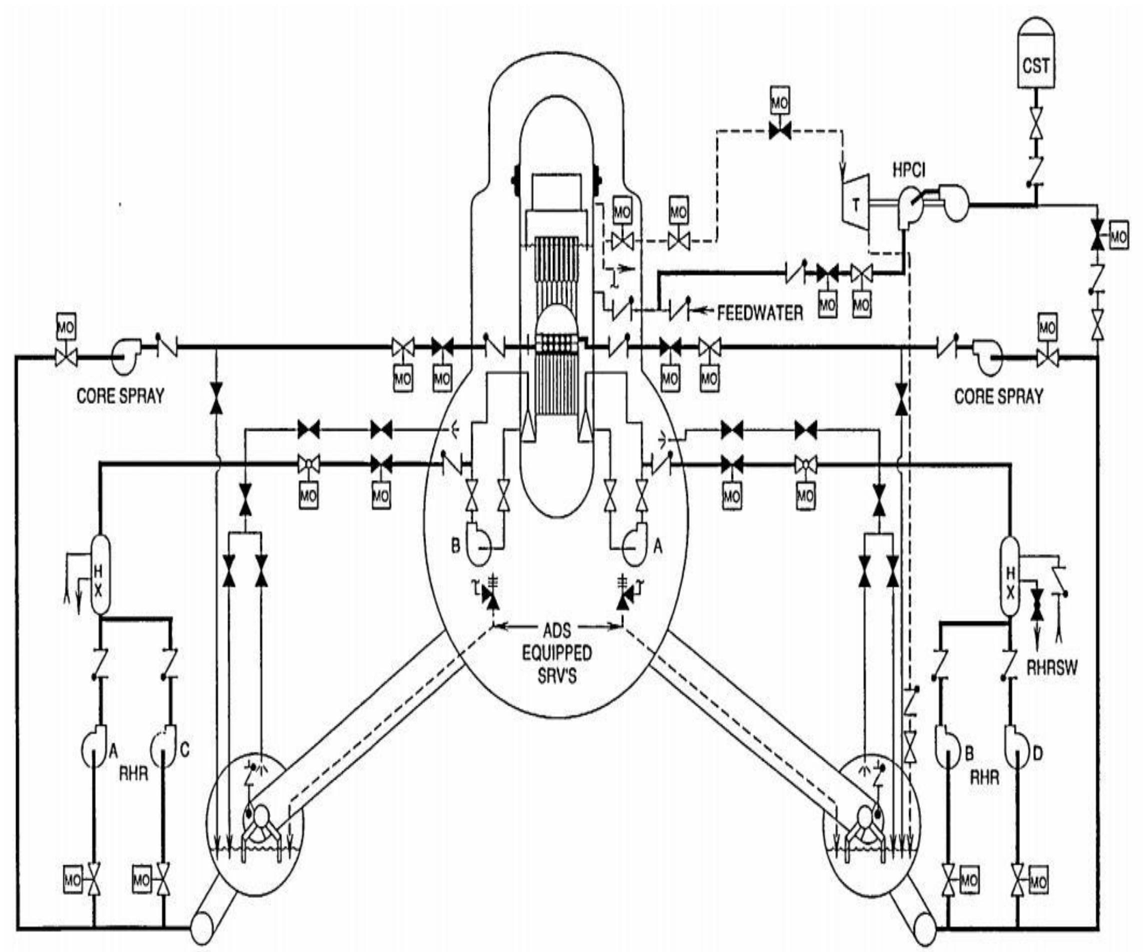

Figure 6. A composite of the emergency core cooling systems [13]

2.1.4.2. High-pressure coolant injection system

The high-pressure coolant injection system, illustrated in Figure 7, is the initial system to respond as part of the emergency core cooling systems, injecting 5000 gallons of water per minute into the reactor, over a range of 150 to 1150 psig. It prevents activation of the automatic depressurization system, low-pressure core spray and low-pressure coolant injection at high pressure. It is activated at low-low reactor water level or high drywell pressure. The high-pressure coolant injection turbine is powered by steam from the reactor, and spins from an initiating signal at any pressure above 100 psig.[14] It maintains the water level in a major emergency, such as a large break in the feedwater line, and can run without pumping water to the reactor vessel. The high-pressure coolant injection removes steam from the reactor and slowly depressurizes it without the need for operating the safety or relief valves. This minimizes the number of times the relief valves have to open and reduces the potential for one sticking open, causing a small loss of coolant accident, which occurred in the Three Mile Island (TMI) plant when a valve stuck open.

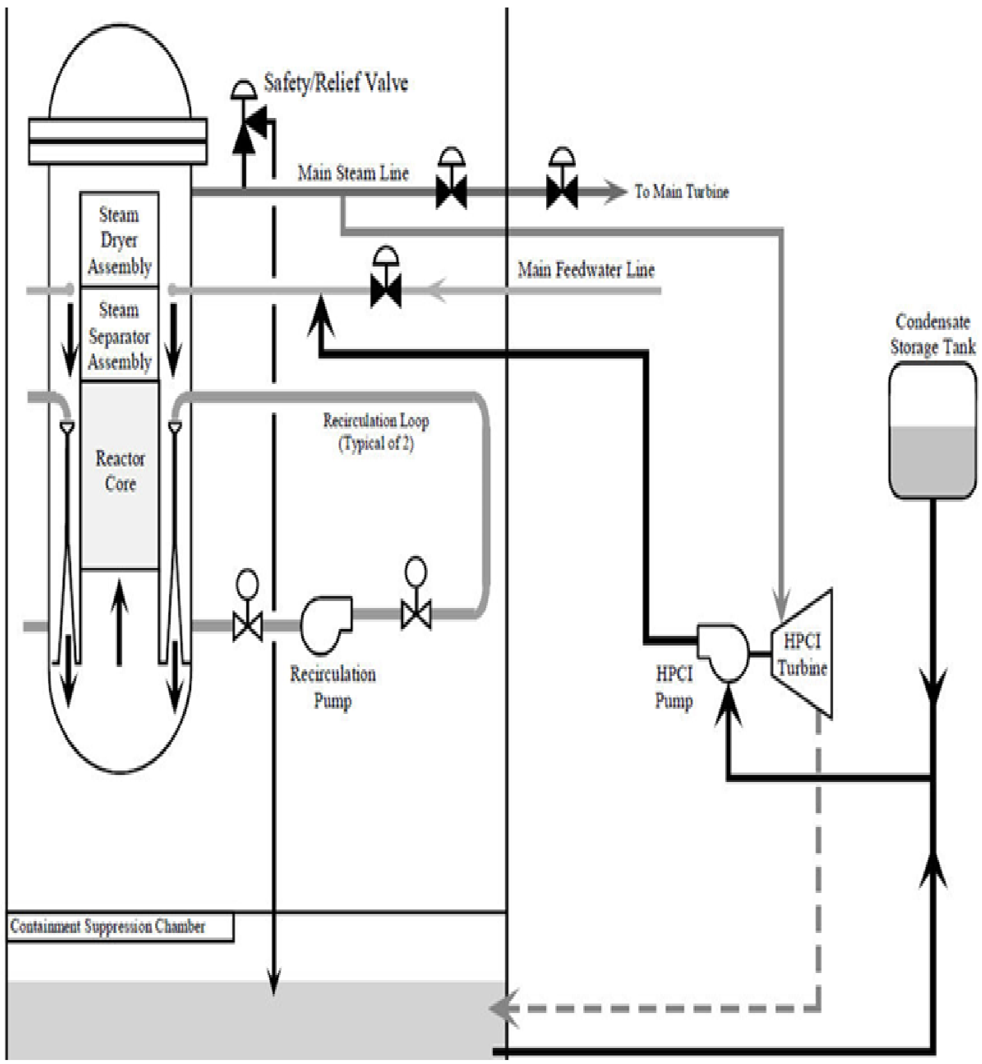

Figure 7. High-pressure coolant injection system [14]

2.1.4.3. Isolation condenser

The Fukushima Daiichi nuclear power station has an isolation condenser, as shown in Figure 8, [14] which is a passive system. This system includes a heat exchanger, located above the containment chamber, in a pool of water open to the atmosphere. When it functions, the decay heat is drawn into the heat exchanger in the form of steam where it then condenses back to water, which falls by gravity into the reactor. This process keeps the cooling water contained within the reactor, making it unnecessary to use electric-powered pumps. The water in the open pool evaporates as a clean steam to the atmosphere. This is part of the emergency core cooling systems and in normal operation, it is not used.

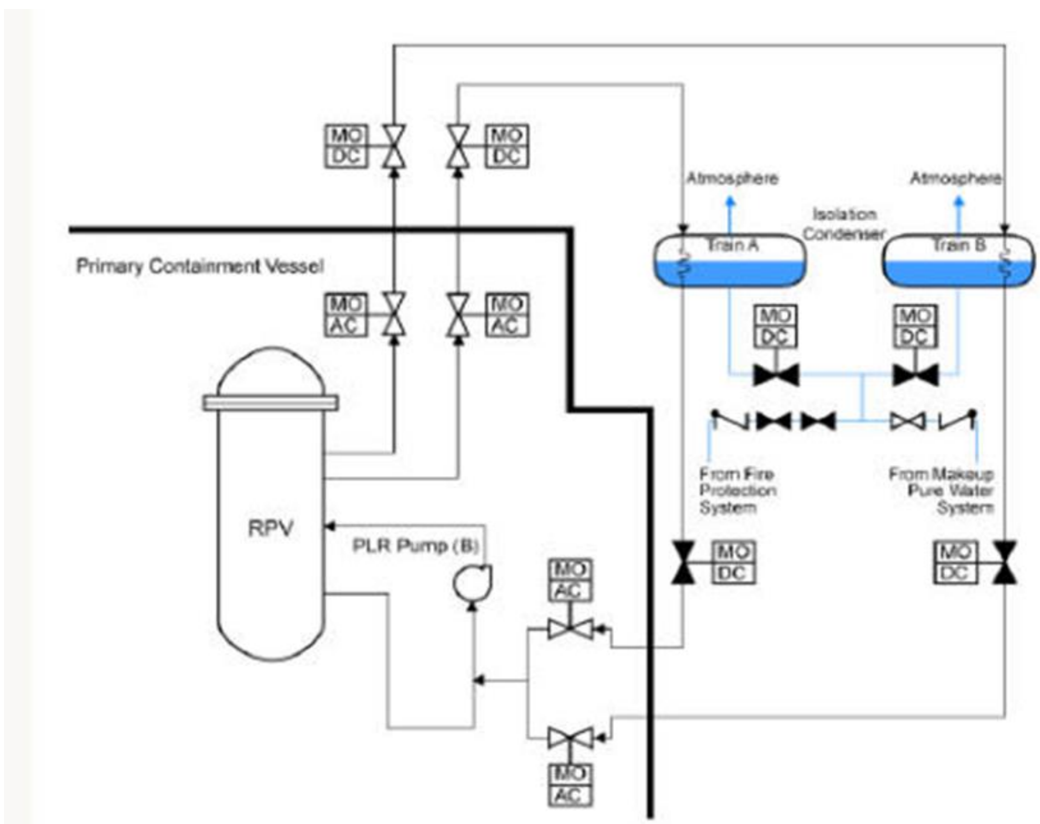

Figure 8. Isolation condenser system [14]

2.1.4.4. Reactor core isolation cooling system

Theoretically, the reactor core isolation cooling system is not part of the emergency core cooling systems; however, it is identified as such, because it performs an important safety function under an emergency condition. It keeps the nuclear fuel covered during a loss of normal heat removing function. The most important feature of this system is its functionality when AC power is unavailable, such as occurred in the Fukushima Daiichi accident. It does, however, need DC power. The reactor core isolation cooling system performed its function when DC power was available at the Fukushima plant. Once the DC power, supplied by the station batteries, became unavailable, this system lost its ability to cool the core. The system has a turbine-driven pump supplied by the reactor steam and delivers approximately 600 gpm into the core. It provides faster response than the high-pressure coolant injection system. It is sufficient to replace the cooling water boiled off by decay heat, and can even keep up with small break accidents. The reactor core isolation cooling system valves are used to control the system flow to maintain the correct water level in the reactor. In the event of station blackout, the reactor core isolation cooling system may be started manually and exhaust the steam into the suppression pool. Figure 9 [14] provides the system drawing of the reactor core isolation cooling system.

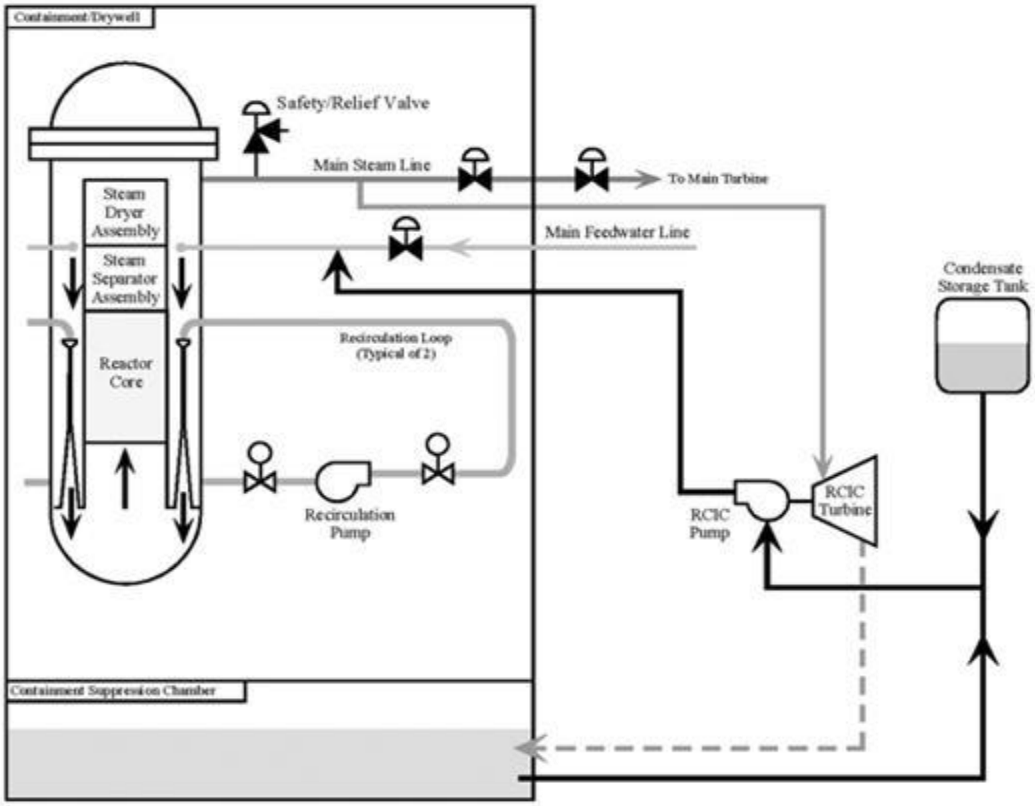

Figure 9. Reactor core isolation cooling system [14]

2.1.4.5. Automatic depressurization system

The automatic depressurization system, even though not considered as part of the emergency core cooling systems, is an important system that supports the emergency core cooling systems by way of reducing pressure in the reactor pressure vessel. It is essential for the low-pressure core cooling systems to add coolant to the reactor pressure vessel, and they can do so when the reactor pressure vessel's pressure is sufficiently low. It is the function of the high-pressure systems to reduce the pressure in the reactor pressure vessel to a sufficiently low level for the low-pressure systems to function and inject coolant into the reactor core. The automatic depressurization system can be manually or automatically initiated. When the water level reaches the designated low-level set point, it automatically starts. Additionally, the system requires a signal that one low-pressure pump is running. It then starts a two-minute timer to allow the water level to rise. When the timer expires, or when the system is manually activated, the system starts and rapidly lowers the pressure in the reactor pressure vessel by blowing steam piped to the pressure suppression pool. The steam is exhausted below the water level in the Taurus[8] as it did at the Fukushima Daiichi accident. The steam condenses to water in the voluminous pressure suppression pool. Figure 10 [15] illustrates the automatic depressurization and low-pressure core cooling systems.

The design of the automatic depressurization system is to reduce the pressure in the reactor pressure vessel below 465 psig. At this pressure, the low-pressure systems of the emergency core cooling systems can inject coolant into the reactor vessel. These injections will recover the water level in the reactor pressure vessel to cool the nuclear fuel. The removal of the decay heat in this fashion is enough to maintain the reactor fuel in a cool condition, even though the fuel may become uncovered for a short time. The water in the reactor will rapidly flash to steam, as reactor pressure drops, carrying away the latent heat of vaporization and providing cooling for the entire reactor, most importantly the nuclear fuel assemblies. Low-pressure cooling injection system floods the core

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[8] Taurus is another name for the pressure suppression chamber or suppression pool. It is a toroidal-shape chamber half filled with water that receives exhaust steam from the reactors' safety relief valves or the automatic depressurization system. It also provides a source of water for various emergency core cooling systems to inject water into the reactor pressure vessel when needed. This type of pool exists only in boiling water reactors with Mark I design.

prior to the end of the emergency condition, ensuring that the core and fuel assemblies remain cool during the entire emergency condition.

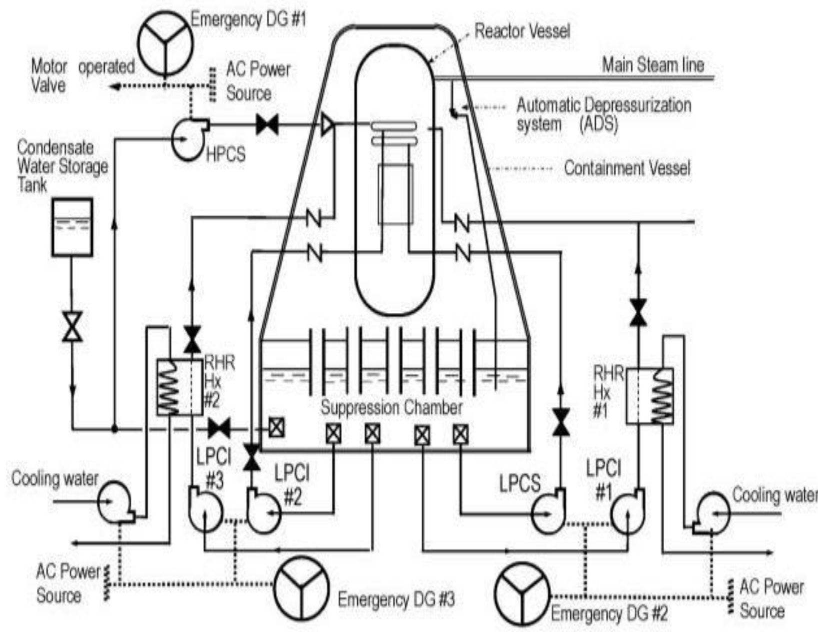

Figure 10. Automatic depressurization system and other ECCS [14]

2.1.4.6. Low-pressure core spray system (LPCS)

The low pressure core spray system suppresses the steam generated in a major loss of coolant accident. As the name implies, cool water is sprayed circumferentially inside the reactor pressure vessel in two circular pipes with holes in the bottom of the pipes. One pipe receives coolant from one low-pressure core spray pump, and the other from the second low-pressure core spray pump. The researcher observed the discharge of low-pressure core spray in an open reactor pressure vessel during the initial test period. The level rapidly rose in the reactor pressure vessel, necessitating the fast closure of the valves during initial testing. The two low pressure cooling injection pumps deliver 12,500 gallons of water [14] every minute. The core spray system collapses steam voids above the core, aiding further reduction of reactor pressure. Figure 11 [16] illustrates the low-pressure core spray system.

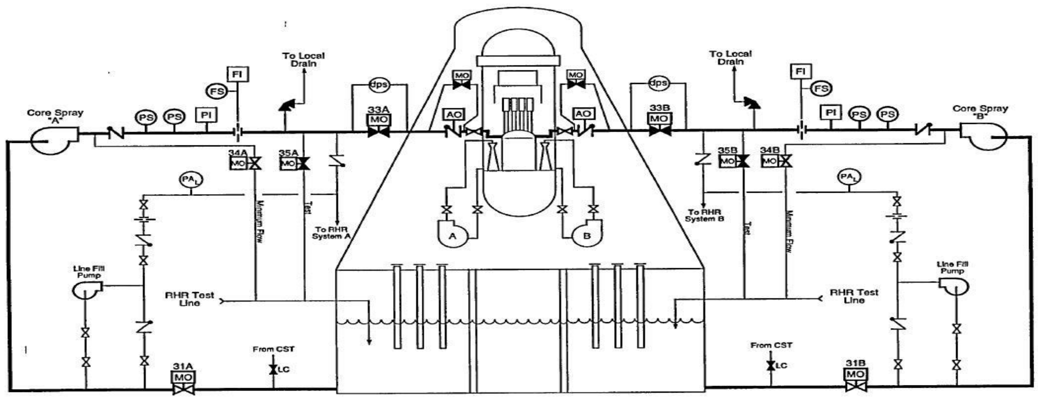

Figure 11. Low-pressure core spray system [16]

2.1.4.7. Low-pressure coolant injection system

The low-pressure coolant injection is part of the residual heat removal system, which is part of the emergency core cooling systems. The low-pressure coolant injection function, as the name implies, is an injection system discharging coolant to the reactor pressure vessel at a pressure below 465 psig, using the existing pipes connected to the reactor pressure vessel. Similar to the low-pressure core spray, low-pressure cooling injection consists of multiple pumps and injects 40,000 gal/min [14] of water into the reactor core. The low-pressure cooling injection and low-pressure core spray systems suppress the pressure by rapidly flooding the reactor pressure vessel. Low-pressure cooling injection uses the residual heat removal heat exchangers to remove the decay heat from the reactor and the primary containment. In the Fukushima Daiichi station, low-pressure cooling injection inserted coolant through the recirculation loop into the downcomer of the reactor (see Figure 4). Later, the system was modified to inject directly inside the core shroud, which minimizes time to flood the core, and to a greater extent, reducing the peak temperatures of the core during loss of coolant accidents. Figure 12 [14] illustrates low-pressure cooling injection system.

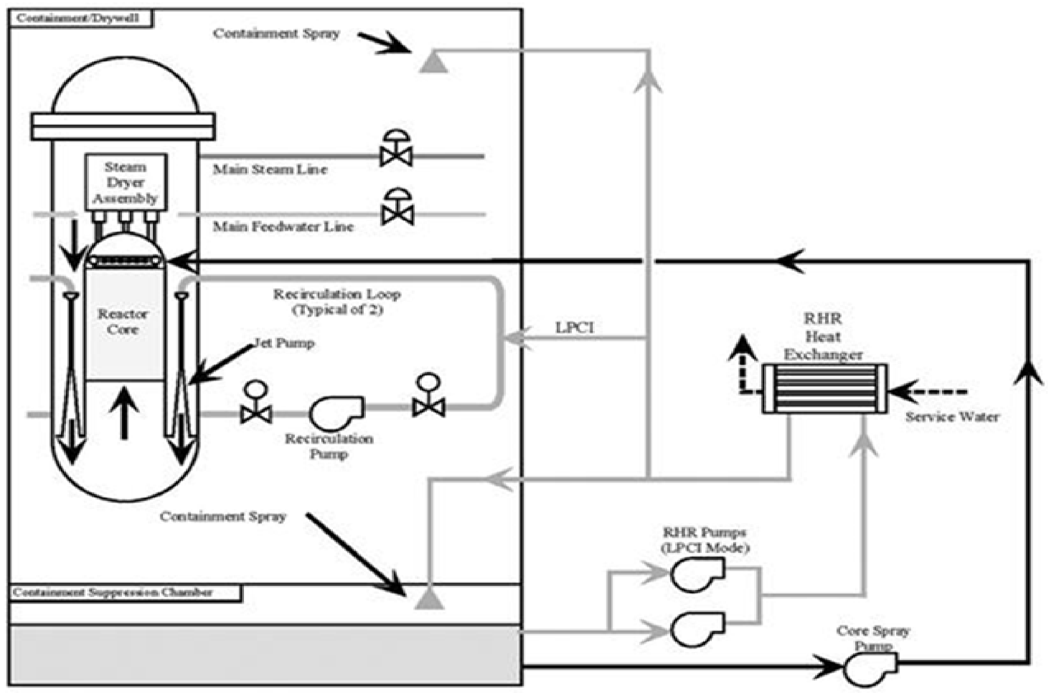

Figure 12. Low-pressure cooling injection system diagram [16]

2.1.4.8.   Standby liquid control system

The standby liquid control system is a backup for the control rod drive system. When the reactor protection system is unable to insert the control rods into the reactor, for any reason, the reactor core remains in the criticality [9] state, and core reactivity remains high. In that condition, the standby liquid control system will inject liquid sodium pentaborate into the reactor. This will bring the reactor to a shutdown condition prior to exceeding reactor core safety limits. The solution that is delivered to the reactor is 86 gpm of 13% [14] by weight sodium pentaborate into a 251-inch boiling water reactor vessel. This percentage and

[9] 'Criticality' is a nuclear term relating to a state when nuclear fission is self sustaining and there is no reason to add more fuel to increase fission rate. When a reactor reaches that state, it is referred to as critical.

flow rate varies depending on the size of the reactor pressure vessel. The solution is kept in a tank with heating elements to maintain the solution in a liquid state. If there is no heating, the solution can turn into crystals, thereby losing its ability to flow as a liquid. Another unique aspect of this system is that it has two parallel valves controlled by an independent explosive actuator. An electrical current is sent to the two valve actuators causing them to actuate a solenoid to open, creating a path for the solution to inject the solution into the reactor pressure vessel through its own reactor vessel penetration. Pressurized water reactors also use this system as a backup for their reactivity control when the control rods do not insert into the reactor pressure vessel. In boiling water reactors, hydraulic force inserts the control rods into the reactor from the bottom of the reactor pressure vessel. In pressurized water reactors, the control rods are inserted from the top of the reactor pressure vessel by gravity. The reason the control rods in boiling water reactors cannot be inserted from the top of the reactor pressure vessel is that at the top of the reactor core there are steam separators and steam dryers (see Figure 4). They occupy the cavity above the core inside the reactor pressure vessel. The boiling water reactors are a saturated steam system where water boils at 1000 psig, and pressurized water reactors are super-saturated system with a pressure of 2000 psig. Figure 13 illustrates the standby liquid control system.

The pressurized steam from a pressurized water reactor goes to a steam generator, which converts the super saturated steam from 2000 psig to saturated steam at 1000 psig, and the steam from the steam generators feed the steam turbines driving the generators. It is for this reason that the turbines from the boiling water reactor plants are enclosed, because they contain radioactivity, even though short-lived, (^{16}N isotope with 7.14-second half-life) [17], whereas, the steam in the turbines of a pressurized water reactor plant is clean and contains no radioactivity. One look at the San Onofre nuclear power plant from the Freeway 5 in southern California indicates that it is a pressurized water reactor plant. This is because the turbines are open to the atmosphere with no containment structures, hence, the steam from the turbines release no radiation to the environment. The steam does not contain the radioactive isotope ^{16}N.

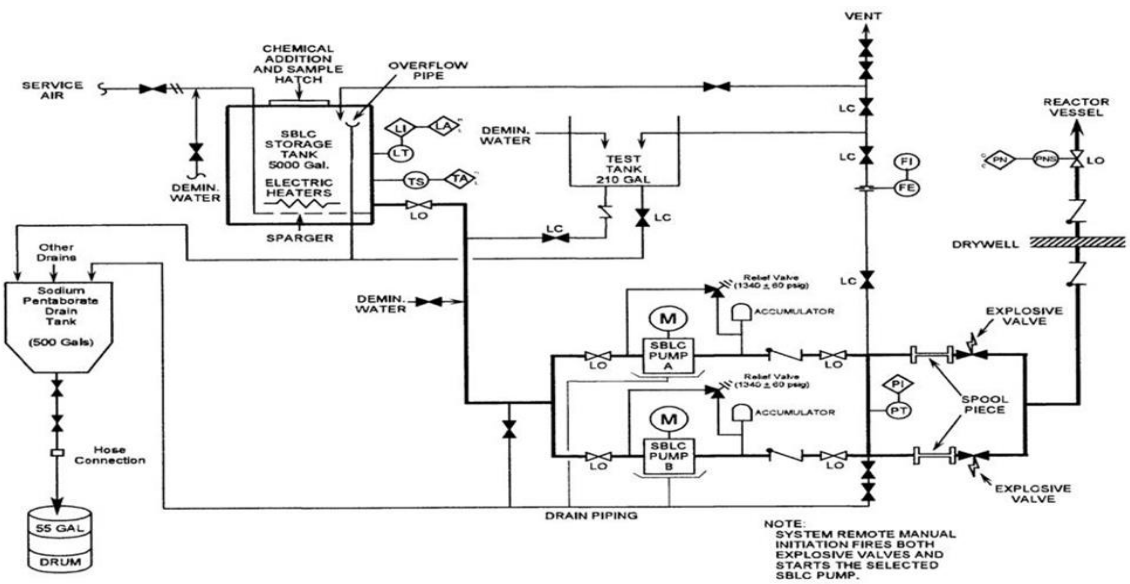

Figure 13. Standby liquid control system (SBLC) [16]

2.2. The U.S. nuclear power plants

The nuclear power plants that were considered for evaluation in the U.S. are located in the eastern, southern and western coastal states. The exceptions for consideration were those nuclear power plants that were either shut down or decommissioned. There are no similar risks associated with any plant where there are no irradiated fuel assemblies in the reactor pressure vessel. There can be irradiated fuel assemblies in the fuel pool, however, and if the fuel pools are flooded, the water cools the fuel assemblies with the support of the fuel pool cooling system. There are 15 US operating plants in 11 states that are susceptible to flooding by notable waves. Each plant is listed and addressed on an individual basis. For the remaining states where nuclear power plants exist, there is no potential for notable high waves caused by a major flood. Even though there is potential for earthquakes, such events are separately analyzed and evaluated, but not in this research. Below are the states with coastal nuclear power plants that are susceptible to a major flood:

STATES	NUMBER OF COASTAL PLANTS
California	1
Connecticut	1
Florida	2
Louisiana	1
Massachusetts	1
New Hampshire	1
New Jersey	3
New York	1
North Carolina	1
Texas	1
Virginia	2

There are about 61 commercially operating nuclear power plants in the U.S., with 99 nuclear reactors in aggregate. Figure – 14 shows the locations of coastal nuclear power plant sites. This graph also provides additional useful information, such as the number of reactors.

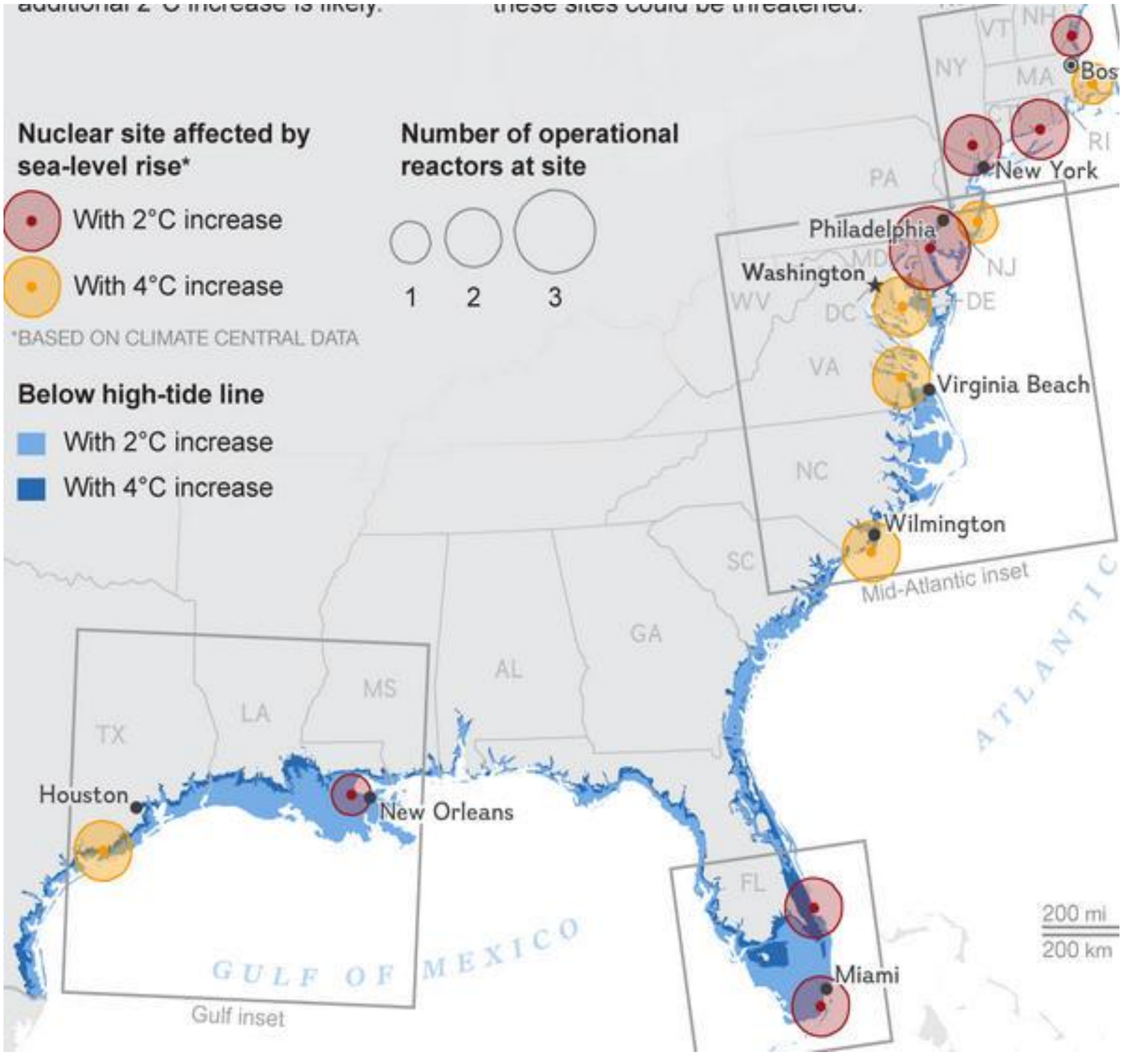

Figure 14. Location of the nuclear power plants in coastal parts of the U.S. [18]

The total number of reactors are located in 30 different states.[19] Within these plants, 35 have two or more reactors. The Palo Verde nuclear power plant in Arizona is the largest plant with its three reactors and a net electrical capacity of 3,937 MWe. The R. E. Ginna plant in New York is the smallest, having a single unit with a net electrical generating capacity of 508 MWe.[19] The last reactor to start operation, in October of 2016, was Watts Bar Unit 2 with 1,150 MWe. Two new reactors are currently under construction, the Vogtle Units 3 and 4 in Georgia.[19] Construction at two reactors located at the V. C. Summer plant in South Carolina was abandoned on July 31, 2017, after completing 40% of the project, and having spent $9.1B. The decision was reached to terminate the project due to forecast of high future expenditures to complete the project.

Clearly, all nuclear power plants in the U.S. are prone to earthquakes with varying degrees of probabilities and seismic strength. It is critical to note that, if it had not been for the tsunami that hit the Fukushima Daiichi plant, there would not have been any core damage and no release of radiation at that plant. The diesel generators initially performed their safety functions, and started to pump coolant into the reactor, thus keeping the core cooled. However, once the tsunami with its high waves arrived, the waves disabled all diesel generators responsible for cooling the cores. Since the main generator was tripped and the grid was too damaged to provide any AC power to the plant, there was no power to operate any of the emergency core cooling systems, except for the reactor core isolation cooling system with its limited capability. Consequently, after one day when the reactor core isolation cooling system lost its DC power and reached its capacity, there was no cooling system functioning to cool the operating reactors. The latent decay heat began the process of melting the fuel assemblies and the reactor cores. All nuclear power plants are designed, as was the Fukushima plant, to handle the design basis earthquake. Thus, for those plants that are not prone to a tsunami, the researcher did not perform an analysis of how to mitigate core damages due to earthquakes. There are two reasons for not performing risk analysis due to a seismic event. One reason is that the NRC has mandated that all plants in the US perform analysis using probabilistic risk assessment to ascertain that the value of core damage frequency falls within the acceptable risk based value of the design basis seismic events. If after the analysis, the core damage frequency is too large, the licensees must implement corrective measures to circumvent reaching too large a value of core damage frequency. This is already a requirement for the US licensees. NRC examines these analyses under the individual plant examination (IPE) [10]process.

Each licensee performs these required analyses and submits them to the NRC for examination and evaluation. For this reason, there is no basis to conduct research to determine the impact of a seismic event at a US nuclear power plant. The second reason that the researcher refrained from conducting a risk analysis for seismic events was based on the impact of the large seismic event at the Fukushima Daiichi nuclear power station. Many sources referred to the absence of gross damage after the earthquake on that day at that plant. Two TEPCO scientists published a book titled "Reflection on the Fukushima Daiichi Nuclear Accident" [20] In Section 2.2 of that book, they state: "*The reactor systems were found to be intact even with the impact of the Earthquake, from the observed plant operation status and the results of seismic assessment using observed ground motions; the main equipment having important functions for safety maintained ~~its~~ [their] safety functions during and immediately after the Earthquake."* [20] Thus, when a seismic event of 9.0 on the Richter scale did not jeopardize the functionality of the plant safety systems at Fukushima Daiichi,

[10] Each licensee must provide input to the NRC for each of their nuclear power plant in a process referred to as an individual plant examination (IPE), which is for severe accident vulnerabilities, based on a risk analysis using probabilistic risk ssessment (PRA) that considers the unique aspects of a particular nuclear power plant, identifying the specific vulnerabilities to severe accident of that plant, including a seismic event.

there is very little risk involved from seismic events alone when the plants are designed for such events.[20] Although the Fukushima Daiichi nuclear power station was a boiling water reactor, the majority of the 15 U.S. nuclear power plants in the coastal states are pressurized water reactors. Of the 15 plants, only four, each with one reactor, are boiling water reactors. One must note that the South Texas project has two advanced boiling water reactors planned for construction, yet at the time of this writing, construction has not begun. Of the 15 plants, one is located on the west coast (Diablo Canyon), one is on the Gulf of Mexico (South Texas Project), two are in the State of Virginia, (North Anna, and Surry) and the remaining plants are on the east coast (Brunswick, Oyster Creek, Millstone, Pilgrim, St. Lucie, Turkey Point, Indian Point, Salem, Hope Creek, Seabrook, and Waterford).

2.3. Restrictions and limitations of the research

This section lists the limitations and restrictions relating to this research. There are two categories of limitations and restrictions that apply to the Fukushima and U.S. nuclear power plants.

2.3.1. Plant systems

There are restrictions to the data used in this research relating to the systems and components of the nuclear power plants that were evaluated. Clearly, it is unnecessary to evaluate all systems' responses and their failure mechanisms under the accident conditions in order to meet the research objectives of this dissertataion. In a typical boiling water reactor type plant, there are thousands [4] of valves, thousands of pumps, and motors. Thus,It is unnecessary to evaluate the entire set of components of this type of a plant. This research only focused on those components of the safety-related systems that were essential for providing core cooling and containment integrity. For this purpose, there were over 100 components that were considered in this research.

2.3.2. Duration of the accident

The onset of the accident began with the seismic event. The focus of the research was to evaluate the sequence of events for the first four weeks. It took about three weeks before power was restored to the plant through an unusual process, and about six months before outside power using the electrical grid was reconnected. The consequences of the accident are still on-going, eight years after the accident. The latest challenge is a question about the effectiveness of the freezing wall around the reactor buildings of the four damaged reactors. As late as January 1, 2017, Japan’s Nuclear Regulatory Authority questioned the effectiveness of the frozen wall and it is pushing more towards pumping the radioactive and contaminated water from the bottom of the affected reactor buildings.[5] Tokyo Electric Power Company (TEPCO) stated that they would double their pumping capacity of the contaminated water by the end of 2017.[5]

2.4. Hydrogen issue, including accumulation and detonation

The accumulation and explosion of hydrogen played a major role during this accident as it caused the breach in four containments creating a pathway for the fission products to escape to the outside environment. There are great efforts during the design phase of nuclear power plants to contain the fission produced in the reactors inside the containment. One critical difference between the

Chernobyl design and the U.S. light water reactor technology design is the reliance on the containment itself for the U.S. nuclear plants, and maintaining its integrity during accident conditions. In the history of the U.S. commercial nuclear power plants, there has not been a single accident causing a breach of the containment, both primary containment (drywell in BWRs) and secondary containment, the reactor building. It is for this reason that the discussion on hydrogen needs its own subsection, which applies to both the U.S. nuclear power plants and the Fukushima Daiichi nuclear power station.

During normal plant operation, even though hydrogen is generated, it is not in a large quantity and goes through the plant discharge system, thus not accumulating. However, when the nuclear fuel reaches high temperatures, the zirconium (Zr), which exists in the fuel cladding and other reactor material, oxidizes and reacts with the steam as shown in the following reaction:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$ [14]

One molecule of Zr reacts with two molecules of water (steam) resulting in one molecule of zirconium dioxide and two molecules of hydrogen gas. Considering the high quantity of Zr in the fuel material and other components inside the core, the generation of hydrogen in a gaseous state is substantial. There is 76,000 Kg of zirconium in a typical boiling water reactor [21], and there are 6.02×10^{23} (Avogadro's number) of molecules of H_2O in 18 grams of water. The quantity of the released hydrogen is a function of how much cladding material converts through the above reaction. The hotter the fuel assembly, the more oxidation of the Zr; and the more oxidation of Zr, the more hydrogen molecules are released. As long as the fuel cladding integrity is maintained, the hydrogen molecules do not migrate to the outside of the fuel rods and combine with the reactor steam. Once there is a path for the hydrogen to escape through the cladding, the accumulation will remain inside the reactor pressure vessel. As long as the reactor pressure vessel is sealed, there is no path to the outside of the reactor pressure vessel. In the Three Mile Island accident, the engineers reached the conclusion that hydrogen molecules were present as a consequence of the fuel failures and degraded core conditions. However, a major adverse event would have been a hydrogen detonation inside the reactor pressure vessel, thus damaging the geometry of the reactor core. The Three Mile Island operators resorted to slow release of the hydrogen gas from the reactor pressure vessel and prevented conditions that could have resulted in a hydrogen explosion inside the reactor pressure vessel. In four of the Fukushima Daiichi reactors, where there were irradiated fuel assemblies, the fuel failed. Hydrogen gases were released, causing hydrogen detonation outside of the reactor pressure vessel. The hydrogen explosions resulted in failure of the containment integrity, providing a path for the fission products and radioactive radionuclides to the outside environment.

From the lessons learned from the Three Mile Island accident, the NRC mandated that [21] all U.S. nuclear power plants use a method to handle the accumulation of hydrogen and to prevent it from reaching the threshold of hydrogen detonation. This could be achieved by installing either a hydrogen recombiner or hydrogen burners The hydrogen recombiner uses catalytic oxidation that combines the hydrogen with oxygen to lower flammability. Recombiners do not require power, whereas the hydrogen burners, either as spark igniters or flammable burners, require electrical power to function.[21] Hydrogen burners would not have helped in the Fukushima Daiichi accident due to the loss of power, whereas the hydrogen recombiner would have helped prevent hydrogen accumulation.

2.5. Classification of failures

Based on the timeline and investigation of the failure of the components, the failure modes of key components of the selected safety-related systems have been analyzed. The key components of each system were selected, including pumps, valves, piping, motors, air operators (for valves), liquid containing tanks, diesel generators, and batteries. The failure modes were identified, and fell in the following categories:

a. Loss of suction
b. Pipe rupture
c. Loss of available signal to actuate
d. Flooding
e. Human error

Although these are various types of failures, performing analysis requires data. The analysis is based on performing probabilistic risk assessment and to perform probabilistic risk assessments. Performing probabilistic risk assessments requires sources of failure data related to component failures. For this analysis, human error was not taken into consideration as a cause requiring specific data. Instead, this will be one category of analysis requiring its own research. There are many types of human errors with their own root causes.

The causal elements of human errors relate to many factors, including training, experience, procedures, organizational structure, communication, concentration, clear instructions and so many other factors. In order to examine this aspect, there needs to be related data in each of the above categories. Aside from the human error, to achieve results, the probabilistic risk assessment was used and that analysis requires performing fault tree analysis and event tree analysis. Both analyses require failure frequencies of mechanical components. The two prominent sources of relevant data are published by the U.S. NRC and the International Atomic Energy Agency (IAEA). The latter source is more encompassing because it covers a wider array of components, whereas the NRC's data compilation is in the NUREG/CR 6928.[22] The IAEA document, "Component Reliability Data for Use in Probabilistic Safety Assessment," [23] was used in this research. The NRC's document, NUREG/CR 6928, [22] examines 51 types of components and 151 types of failure categories. In the NUREG/CR 6928, there are four types of events: component unreliability, (designated as UA), component or train unreliability, (designated as UR); system special events during testing or maintenance; and initiating events, designated as ERs. The failure frequency distributions for these types of failures are reflected by beta distribution for the first class, and gamma distribution for the others. There are mathematical equations given for these distributions, and the beta and gamma values are provided in many tables in the NUREG 6928. However, there are no available data on the failure of water sources for suction by various pipes. Further, there are no available data pertaining to dislocation of equipment and holding tanks, as a consequence of the flood waters with various velocities. There are more than adequate sources of data pertaining to procedural inadequacies for human errors. One key source regarding human error is a 1983 publication by the U.S. NRC. It is extensively used by researchers and utilities. It is the “Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plants Applications.” [24] Although this voluminous book (728 pages) covers many types of human unreliability and human failures, it does not correlate the failure of a component within a system to the type of a human error. In addition, there is a Ph.D. dissertation that provides a great compilation of data on human errors associated with a nuclear power plants.[25] Similarly, this research does not assign a component failure in a specific system to a specific type of human error.

2.6. Data gathering.

Data were obtained from several different sources. Various Japanese sources provided invaluable data pertaining to the plant and activities that took place on that fateful day. The data came from the plant, TEPCO, and their Japanese regulating agencies as well as from other nuclear and technical organizations. In addition to data obtained from the Japanese sources, data were also obtained from international organizations, notably, IAEA, and other reputable scientific sources. U.S. sources also provided significant and key data pertinent to the event. The U.S. NRC is one such source, as are EPRI, INPO, and other nuclear and scientific sources. There are many investigative sources that evaluated and analyzed this accident. The referenced Book -1 invoked in this document is one such example when many different entities provided data, associated with investigations of the event. The main shortcoming of the book is that it does not assign fault in contradiction to the findings by so many scholars and authors. Data used in this research was not altered in any way. The researcher, as a member of a university faculty who has taught ethics, especially ethics in conducting research, is intimately aware of the consequence of unethical data collection to substantiate a preconceived notion. Any alteration in data collection (except for erroneous data) or the selective data collection is inherently unethical and unacceptable. No entity, especially academic institutions, would accept or tolerate such behavior by any of its researchers, students or staff members. Data collection is a critical part of any research, whether in a laboratory or in a literature research.

2.6.1. Data from Japanese sources

The main credible source of Japanese data comes from Tokyo Electric Power Company (TEPCO).

2.6.2. Technical Data from NRC, INPO, NEI, IAEA, and EPRI

Each of these nuclear entities provides numerous sources of data. The NRC and Electric Power Research Institute (EPRI) are at the forefront of data providers. Both NRC and IAEA provide data on failure frequency of components used in the nuclear power plants. The NRC's data document is contained in NUREG/CR 6928,[22] which provides the nuclear power Industry's average performance data for system components. It also provides data on initiating events at the U.S. nuclear power plants. The data on component failure is defined and provided for 50 different components and 150 different scenarios. The U.S. utilities use and apply the data contained in this document for performing their probabilistic risk assessment analyses. Further, the NRC publishes many different types of data related to systems and components. There are many references to system descriptions in documents for both boiling water reactors and pressurized water reactors. EPRI has many types and sources of data. They have close ties with the Department of Energy (DOE), the NRC, and nuclear power plants. EPRI has extensive involvement in developing codes and models to facilitate analysis and operation of the nuclear power plants. EPRI works closely with academic institutions and provides funding support for students to conduct research. Currently, they are becoming heavily involved with the Fukushima Daiichi station.[26] In addition, another source of data is IAEA, which has strong ties with international utilities, including those in Japan. Japan has submitted many of its technical reports, particularly, on the Fukushima Daiichi accident to the IAEA. The IAEA was created in 1957 in response to the initiating motion by President Eisenhower in his “Atoms for Peace” program. He addressed the U.N.’s General Assembly on December 8,1953. The IAEA’s main location is in Vienna, Austria, and it has two regional offices. One is in Toronto, Canada (since 1979) and the other is in Tokyo, Japan (since 1984), as well as two liaison offices in New York City, (since 1957) and Geneva, Switzerland (since 1965). One of the key data sources produced in 1988 by the IAEA was a document titled “Component Reliability Data for Use in

Probabilistic Safety Assessment."[23] The document, lists the various components of safety-related systems, and provides the failure frequency of such component. There are many categories of components, such as pumps, valves, switch gears, batteries and battery chargers, diesel generators and other types of components used in a commercial nuclear power station. The failure frequencies are given for "fail to start", "fail to run", "fail to function", and "fail to operate". Failure frequencies are based on actual data from nuclear power plants and other sources including vendors and component manufacturers.

2.7. Data analysis

The probabilistic risk assessment analysis strongly depends on event tree analysis (ETA), and that is a function of the failure frequencies of the components, which are tabulated in either the NRC document (NUREG/CR 6928) [22] or the IAEA document [23] (component reliability data). Both documents specify failure frequency of various components used in the safety-related systems. The data in the NUREG 6928 [22] are more recent and the associated failures are not the same as the listing in the IAEA document. The reason is that the numbers in the NUREG/CR 6928 [22] include more recent data, and consequently, there is no reason to apply extreme conservative correction factors. The probabilistic risk assessment results using the NUREG/CR 6928 [22] are, therefore, different from the results based on the values obtained from the IAEA source. For the benefit of readers, the values of component failure frequencies are provided in the Table 2 below. It must be noted that the values are categorized differently between the two documents. The NUREG source breaks down the failure based on the type of failure, such as failure to open (FTO), failure to close (FTC), failure to run (FTR), whereas the IAEA source refers to fail to function (FTF). This is a partial list for comparison purpose only.

Table 2. Comparison of failure frequencies from IAEA and Nureg /CR 8926 documents

System	Component	Failure Type	Source	Failure frequency[11]
Diesel Generator	Water Heater	FTF	IAEA	3.0 E-3
Diesel Generator	Fuel Oil	FTF	IAEA	6.5 E-3
Diesel Generator	Switch	FTF	IAEA	4.8 E-4
Diesel Generator	The diesel itself	FTF	IAEA	1.25 E-3
Batteries	Batteries themselves	FTF	IAEA	6.0 E-6
Batteries	Connections	FTF	IAEA	6.4 E-6
Batteries	Flooding	FTF	IAEA	1.0 E-2
Battery Charger	The chargers	FTF	IAEA	6.07 E-6
Battery Charger	Connections	FTF	IAEA	1.06 E-5
Battery Charger	Flooding	FTF	IAEA	4.09 E-5
Low-pressure core spray	Motor-operated valve	FTF	IAEA	1.5 E-6
Low-pressure core spray	Signal	FTF	IAEA	3.9 E-6
Low-pressure core spray	Motor-operated pump	FTF	IAEA	7.1 E-6
Low-pressure core spray	AC power	FTF	IAEA	1.4 E-6
High-pressure coolant inje.	Turbine	FTF	IAEA	1.0 E-5
High-pressure coolant inje.	Pump	FTF	IAEA	1.0 E-5
High-pressure coolant inje.	Motor-operated valve	FTF	IAEA	7.0 E-5
High-pressure coolant inje.	Signal	FTF	IAEA	8.7 E-7
High-pressure coolant inje.	Flooding	FTF	IAEA	7.0 E-11
Diesel Generator	The diesel	FTO	NUREG	4.53 E-3
Battery	Batteries	FTO	NUREG	1.86 E-6
Battery Charger	The chargers	FTO	NUREG	5.08 E-6
Motor-driven pump	The pump	FTO	NUREG	5.0 E-4
Motor-operated valve	The valves	FTO/FTC	NUREG	1.07E-3
SBLC pump	The pump	FTR	NUREG	8.32E-6
Relief valves	The valves	FTC	NUREG	2.5 E-3
Safety Relief valves	The valves	FTC	NUREG	7.71E-3
Turbine-driven pump-RCIC	RCIC/HPCI pumps	FTR	NUREG	5.77 E-6
HPCI injection valve	RCIC/HPCI valves	FTO	NUREG	5.0 E-1

The values that appear in the fifth column, Failure Frequency, are obtained from their sources delineated in the fourth column. The numerical values of the failure frequency are used in the analysis

[11] The unit for the values offailure frequency from the IAEA document is mostly given for failures per hour. Some values have been given in failures per day, which requires to be modified to failures per hour.

CHAPTER THREE - LITERATURE REVIEW

In this section of the research, inquisitive evaluation has been made of prior works. At the end of this section, points are made about the benefits gained from this literature review, lessons learned from notable research efforts of others, and gaps in the work that need addressing.

3.1. Purpose of the literature review

The purpose of the literature review was to evaluate the information and data generated by others in matters germane to the FukushimaDaiichi accident. This information was essential to discern how their findings compared with and directed this research. The review showed that prior research work was different from this work with respect to scope, expertise, method of analysis and focus, thus, there was ample justification to proceed with this research. One key difference with previous research is the background, expertise, and experience of this researcher. His experience and knowledge of a boiling water reactor similar to the Fukushima Daiichi nuclear power station permitted the evaluation of the credibility of the data and findings especially with respect to recommendations being implemented and eventually becoming policy.

Another purpose of the literature search was to identify gaps between what others had done and what this research would accomplish. Part of the response to this inquiry rests within Section 1.4 of this dissertation, Unique Aspects of this Research. A more detailed discussion, referred to as literature gap analysis, is presented in Subsection 3.3 of this dissertation.

3.2. Focus of the literature review

The focus of the literature review was on the failure frequency of various key elements of selected safety-related systems, and how this research effort was different than those performed by others. There were finite numbers of components within each system as well as finite numbers of systems to be analyzed. In a commercial nuclear power station, there are large quantities of components and systems that function in order to make a power plant functional in various modes of operation. Not all systems are safety-related systems. In a typical boiling water reactor, there are eleven safety-related systems that were selected for analysis in this research. Depending on the type of systems, there are, on average, five to eight different components that must be functional for that system to perform its intended (design) safety functions. These components and systems are the focus of the literature review.

3.3. Literature gap analysis

Most of the existing literature refers to and discusses the impact of the earthquake and the tsunami that occurred in Japan, on March 11,2011. Examples are (these references are cited in this manuscript):

- Nathaniel Massey: “Fukushima disaster”
- Justin McCurry: “Fukushima disaster could have been avoided”
- Michio Ishikawa: “A study of Fukushima Daiichi nuclear accident process”
- Becky Oskin: “Japan earthquake and tsunami of 2011”
- G. Carusa: “The Fukushima Daiichi accident.”

Other articles and books covered the human elements of this accident. Examples are:

- Peter Melzer: “Failure by design”
- di Luca Carra: “Human errors in Fukushima”

- Maija Nedisan: “Crisis communication, liberal democracy and economic sustainability” – this article provides excellent coverage of the human aspect of the accident.
- Norihiko Shirouzu: “Design flaw fueled nuclear disaster”
- Hiroko Tabuchij: “Inquiry declares Fukushima a man-made error”
- Kiyoshi Kurakawa: “The Fukuishima nuclear accident independent investigation commission” - The Japanese government official report

The researcher was looking for reports, analyses, or technical discussions that examined in detail the safety-related systems that protect the reactor core. The search took the form of discussions with experts, and reading research documents that examined the role of the components in protecting the reactor core. A component-by-component analysis is required in order to identify potential failures.

There were no articles found or referenced by others that addressed the type of analysis the researcher was looking for. Even utilities do not perform probabilistic risk assessments based on the failure frequency of components using event tree analysis.

It is natural that other authors did not examine the failure impact of individual components. To pursue that option, the authors needed to be licensed and have extensive experience in order to know the role of that component, how the component functioned, what other components and systems served as backups. Each safety-related system has a logic that is integral to the system. Some have timers, some have interlocks and some cause trips or isolation. The common element of commercial nuclear power plants is its complexity. In order to obtain an SRO license, a candidate needs to take many courses and spend about two years to acquire an understanding of the complexity of a nuclear power plant. Thus, it would be highly unusual for a researcher without an SRO license and operating experience at a commercial boiling water reactor to perform an analysis on the Fukushima Daiichi nuclear power station using the probabilistic risk assessment process.

3.4. Perspectives

There are three categories of literature that are considered for this research. The first category includes research reports published by other PhD candidates. The usefulness of this research may be limited because these students do not have extensive operating experience in the boiling water reactor technology. Nevertheless, since it is important to examine a wide range of literature, this category of literature is included.

A second category of literature includes many articles, journal reports, newspapers or formal reports by a multitude of sources, including scientists. Most authors did not have extensive operating experience in a commercial nuclear power plant, however, some of the articles covered areas that supplemented the researcher’s understanding of the issues in this research and supported the objectives of this work.

The third category of literature comprises books and book-length reports dedicated to this subject. These were helpful in that most of these authors interpreted their findings based on their own experiences and opinions. One of the more encompassing documents is a five-volume compendium on the subject by the IAEA.

3.5. Coverage

Aside from the formal literature, 250 articles that were reviewed. The articles included newspaper articles, industry documents, and third party reports. If during the review of the literature, there appeared an interesting data or point, it was used in the dissertation, either as a direct quote, (the work was cited with a direct citation number for the phrase,) or a direct quote within quotation marks. If there was a need to modify other work within quotation marks, the changes were shown in a bracket, consistent with the standard use of reference guidelines. For this research, citations and quotes came from six Ph.D. dissertations, two books, and over 110 articles.

3.6. Lessons learned from the literature search

Significant insights and perspective were gained from reading the literature concerning nuclear safety issues and the specifics of the nuclear accident in Japan. Categories of knowledge/learning gathered in this review process are described below.

3.6.1. Reaffirmation of the technical knowledge

The review of the literature confirmed many of the key points the researcher understood to be the causal factors for the accident at the plant. Although the researcher learned little from the technical content of the articles discussing the boiling water reactors, the review reconfirmed the author’s understanding of the events at the plant and the associated safety system failures. These events, included not managing the decay heat, thereby allowing fuel to melt in the fuel rod bursting the fuel cladding containing the ceramic fuel pellets.

3.6.2. A working knowledge of the Japan's regulatory process.

The researcher gained knowledge of the working relationship between the utility and the regulator in Japan. Unlike the U.S., in many parts of the world, including Japan, the electric power companies are part of, or have extremely close ties, with the federal government regulators. In such regulatory systems, the potential exists that the same government officials who regulate the utilities, also manage the utilities. In this type of management structure, the potential is high for regulatory oversight to be lax and for the utility to misapply or misinterpret safety rules and of protocols set by NISA, the Japanese regulator.

3.6.3. More in-depth knowledge of boiling water reactor technology

Although the researcher had an SRO license on an identical plant as the Fukushima Daiichi plant, the license was obtained several decades ago. In search of data, the researcher benefitted from the refresher course in the BWR technology, which addresses important topics, such as the exact volume of the water capacity of the suppression pool and all of the triggering signals for a reactor scram or isolation conditions.

3.7. Organization of the literature search

The wide area of research was divided into different issues and topics, and based on those issues a literature search was organized and carried out. The list of five issues, was organized such as causes of the accident, the technical issues directed the review and included, 1) causes of the accident, 2) the technical issues of the accident, 3) the adverse impact of flooding on nuclear power facilities, 4) recorded history of tsunamis and floods in Japan, and 5) a list of coastal sites for nuclear power plants in the U.S. Once the list of issues was produced, the articles were examined and a list of references from each article was compiled. By reading the title and the abstract from these references, the researcher became confident that the appropriate literatures had been examined. As each section and subsection of this dissertation was being drafted, the researcher referred to the specified list and used the section of the article to inform the researcher's point of view as expressed in the written portion of the dissertation. The researcher only cites here the publication that informs the content of the dissertation. If a publication was not directly used in the dissertation, the publication is not cited and does not appear in the bibliography. Some of that literature appears in Section 10 of this manuscript as supplemental sources of useful information and can be used in future research or by interested readers.

3.8. Readers audience

Within the five issues/areas described above, the author reviewed literature with the potential audience in mind. For this dissertation, the audience is anticipated to be mostly nuclear engineers and technical staff who are familiar with the design, engineering, operation and regulation of nuclear power plants. This dissertation provides a technical description of systems and components, so readers interested the in nuclear plant safety may obtain a better understanding of technical issues raised in this dissertation. Another audience for this dissertation comprises readers who are interested in the management of a complex industrial enterprise. Thus, articles were reviewed and information included in this manuscript that address the complexities of managing a nuclear power plant and the potential for human error.

As a result of the wide range of interests in this topic and the breadth of the potential audience, this dissertation describes the technical issues so they can be easily understood by a reader without much expertise in nuclear engineering. Further, the researcher was reminded that this is a product of an academic institution and should be largely understandable by interested readers in a multitude of related fields.

3.9. Research outcome

As a result of the accident at Fukushima, the anti-nuclear movement had an outstanding opportunity to voice their opinions in support of their belief that fear of nuclear power generation technology is justified. Extensive documents and changes were proposed by the biggest regulator in the world, namely the U.S. NRC. Other industry monitors such as EPRI, NEI, INPO, IAEA, and DOE were not far behind with their opinions. Since this research uses an analytical approach to analyze data, probabilistic risk assessment was selected as a key term when examining literature sources addressing analytics. As described in the sections below, many sources focused on the failure of the humans in control of the plant, TEPCO, and its regulator, NISA. Selection of the literature reviewed was not predicated on how the author(s) supported or rejected the nuclear option as a viable source of electric energy. Instead, the selection of the literature was predicated on whether the research in some way supported this effort.

3.10. Literature review research method

The review of existing literature did not reveal details on how each investigation or research was conducted. However, this was not a handicap as the researcher developed a formal, organized plan to conduct the research, based on his knowledge and experience with various facets of the nuclear power industry.

3.11. Application of the research literature search

In order to identify and process information required for a successful outcome of this research, the researcher compared points raised primarily in dissertations, journal articles, and books. Several useful videos were also examined.

3.11.1. Dissertations

In this part of the literature review of dissertations by other doctoral students related to the Fukushima Daiichi accident were examined. Since this research focused on the Fukushima Daiichi accident, it was useful to compare it to the work performed by other doctoral candidates and examined the differences. Four Ph.D. dissertations are reviewed.

3.11.1.1. Solom, Matthew Alan (2016)

"Experimental study on suppression chamber thermal-hydraulic behavior for long-term reactor core isolation cooling system operation," Texas A&M University, 2016.

This dissertation discussed ways to improve the efficiency of the reactor core isolation cooling system for discharging the steam into the suppression pool.

Through the stratification of the discharged fluid into the pressure suppression pool, the reactor core isolation cooling system can operate longer. Although this is an improvement over the existing design, configuration, it does not allow the reactor core isolation cooling system to remove decay heat until the reactor loses its entire decay heat in the absence of DC power. At Fukushima Daiichi, DC power was not restored for several months, and this modification did not produce results for several months.

3.11.1.2. Metzger, Kathryn E. (2016)

"Analysis of pellet cladding interaction and creep of U_3Si_2 *fuel for use in light water reactors," University of South Carolina, 2016*.

In response to the Fukushima Daiichi accident, there were efforts to improve the design of the nuclear fuel pellets to tolerate higher temperatures than the existing uranium dioxide pellets. The proposed fuel design in this research is made of U_3Si_2 rather than UO_2. This 2016 publication did not address the existing design of the ceramic fuel manufactured by the fuel supplier. As stated in this article, this design change is forecasted in advanced fuel design. It will not address the existing fuel design and the fuel pellets that exist in current nuclear power plants.

3.11.1.3. O'Loughlin, Liam (2015)

"Cosmopolitan events: From Bhopal to the tsunami in South Asian Anglophone," University of Pittsburgh, 2015.

This dissertation examined contemporary global events, focusing on the Bhopal accident and the accident at the Fukushima Daiichi station. This author compares the sociopolitical impact of these events, and examined the human factor in these two accidents. The responsible entities were Union Carbide, the chemical company responsible at Bhopal, and GE, the manufacturer and the designer of the boiling water reactor at the Fukushima Daiichi nuclear power station. The dissertation does not fault GE, the reactor manufacturer, and instead blames the Japanese government (the regulator) and TEPCO for the extensive damage at Fukushima Daiichi. The dissertation indicates that Union Carbide was partially responsible for the accident in India. The author also alludes to other events, including the 1998 nuclear tests in India and Pakistan, a coal mine collapse in 2001, and the 2004 tsunami in Southeast Asia. This research did not address the technical aspects of the Fukushima Daiichi accident.

3.11.1.4. Rosen, Steven (2011)

"The socioeconomic effects of earthquakes, volcanoes, and tsunamis." The Cooper Union for the Advancement of Science and Art, 2011.

Rosen of the Cooper Union for the Advancement of Science and Art published this dissertation on the socioeconomic effects of various natural disasters, including earthquakes, volcanoes, and tsunamis. These natural disasters include the 2010 earthquake in Haiti, the 2010 eruption of Eyjafjallajökull volcano, and the 2004 Indian Ocean tsunami. This paper addresses what caused these natural disasters and includes many useful supplemental technical data pertaining to these accidents. The information includes various intensity scales, a graphical calculation of Richter magnitude, a volcanic explosivity index table, and tables listing the effects of various natural disasters. This research does not technically evaluate the accident in Japan.

3.12. Journals

In order to achieve objectivity in the literature review, there needed to be an examination of documents written by other authors on the impact of external events on the functionality of the plant systems and components. To achieve this objective, the researcher examined many technical journals and commented on the points made by their authors. For emphasis, six journals were selected and reviewed below.

3.12.1. Journal 1 - Flooding of Nuclear Power Plants[27]

This article in the *Nuclear Monitor*, an anti-nuclear newspaper, addresses the consequences flooding in of nuclear power plants. [27] This article focuses on a dam failure causing flooding affecting a nuclear power plant downstream of the dam. It does not examine the impact of a major flood caused by a high wave. It emphasizes the incident at the Fort Calhoun nuclear plant. Fort Calhoun was inundated by flood water due to heavy rainfall on June 20, 2011 (Figure 48). The author of this article strongly accuses the NRC and the utility for not disclosing the truth to the public about the consequence of floods to nuclear power plants. The plant, even though licensed through 2033, was shut down in October of 2016.

3.12.2. Journal 2 - System Study: RCIC 1998-2012 [28]

This article generated by Idaho National Laboratory presents an unreliability study of the reactor core isolation cooling system for 31 plants using boiling water reactors.[28] It examines system demand, run time by the system, and failure data from 1999 through 2012. This is done for selected components generated by Idaho National Laboratory (INL) using the equipment performance index (EPIX). EPIX was developed by INPO and the NRC uses the data for its analysis of system reliability and unreliability. This document refers the users the U.S. NRC NUREG/CR 6928 [22] data that specifies equipment reliability for various systems.

3.12.3. Journal 3 - Preventing an American Fukushima[29]

This article generated by an anti-nuclear group, the Union of Concerned Scientists, questions the role and effectiveness of the NRC.[29] As stated previously, if TEPCO had adhered to recommendations and requirements of the NRC, the majority of the adverse consequences of the accident at the Fukushima Daiichi station would not have occurred. Each accident at a nuclear power facility, (e.g., Three Mile Island, Chernobyl, and Fukushima), resulted in major changes and improvements in the nuclear power plants around the world. The article states: “The NRC required most US reactors to be able to cope with a [station] blackout for only four to eight-hours; Fukushima Dai-ichi suffered without [AC] ac [30] power for more than a week.”

The four to eight hours is the length of time required for a boiling water reactor to reduce the decay heat after a reactor shutdown. This is the premise for the station blackout condition.[31] In a station blackout condition, it is assumed that DC power is available and the 4-8 hours of power are related to the reactor core isolation cooling system receiving a signal from the DC power supply system, or batteries, for its functionality. The reactor core isolation cooling system was not designed to remove the entire decay heat, from full power to cold shutdown which takes about a week. The design basis of station blackout assumes one of the remaining emergency core cooling systems becomes functional to remove the decay heat, and to achieve this objective, the equipment requires AC power. No nuclear power plants in the U.S. were without AC power for a week.

3.12.4. Journal 4 - Mitigation Strategies[32]

This article by the U.S. NRC is a direct response to the Fukushima Daiichi accident.[32] The NRC issued Mitigation Strategies Order, EA-12-049,[32] on March 12, 2012, a year after the Fukushima Daiichi accident. This order required all US nuclear power plants to develop and implement strategies to enable them to possess and rely on alternative and independent electrical power sources for an indefinite period of time. This power must be adequate to remove the decay heat by powering those systems and components to achieve that purpose. Of course, the primary goal is to keep the reactor pressure vessel and the fuel assemblies cool, whether in the cores or in the spent fuel pools. The order requires that the stations be surrounded by thick concrete walls to protect the buildings, the reactors and the spent fuel pools. The order has five (5) specific requirements for license holders of nuclear power plants in the U.S.. These mitigation strategies are delineated in Attachment 2 of the ‘Order’ titled “Requirements for mitigation strategies for beyond-design-basis external events at operating reactor sites and construction permit holders.”[32] As the title specifies, this is for all license holders or those planning to receive a construction permit, and it is designed for external events affecting the nuclear power plants. For those readers who become cognizant of the requirements imposed by the U.S. NRC to protect the nuclear power plants within the U.S., this document provides a great source of information.

3.12.5. Journal 5 - Sustaining Resilience of U.S. Nuclear Power Plants to External Events[33]

This is a slide presentation developed by Mr. Mike Franovich, a director within the Nuclear Regulatory Commission highlighting the key factors of the NRC’s Lessons Learned Near-Term Task Force (NTTF) from the Fukushima Daiichi accident [33]. The presentation package includes the key factors of the NTTF doctrine issued in July of 2011, as well as the related activities up to March of 2017. The recent activities include the inspection of plants for compliance with the earlier orders and performance of safety evaluations and verification of compliance with the NRC requirements and its rule making. This package of information is contained in the slides.

3.12.6. Journal 6 - Human Error at Fukushima[34]

This article, published in an Italian journal, clearly blames human errors for the extent of this accident.[34] It begins by stating that once the Investigation Commission of the National Diet (Japanese parliament) published its official report on the accident, few doubts remained about what caused this accident. It refers to the conclusions of a ten-person Commission that the disaster was a consequence of collusion between the regulatory agencies and TEPCO, the owner of the Fukushima Daiichi station. The article lists six categories of human errors: (1) assessment error, (2) organizational error, (3) confusion of roles, (4) handling the evacuation, (5) post-accident management, and (6) the mindset. The article alluded to other issues such as the emotional impact of the accident, the environmental damages, and the health effects. The article also provided a link to the official document produced by the National Diet of Japan. Its title is "The Fukushima Nuclear Accident Independent Investigation Commission" [35].

3.13. Books

After careful consideration, book reviews are not covered here, but are covered in Sections 4.3.1 for human errors and in Section 5.1 for flooding issue.

CHAPTER FOUR - HUMAN ELEMENTS

There are indications that had it not been for human errors, the Fukushima Daiichi accident would not have suffered to the degree it did.[36] For this reason, a lengthy discussion is covered in this section of the manuscript.

4.1. Human factors

This chapter addresses the second objective of this research, a discussion of the human element and the role it played in the accident. Despite the earthquake and the tsunami, to a large extent the accident was the result of human error. The human factors in this accident are widespread and are important, as evidenced by claims made by international organizations and Japanese government officials.[37] The human errors were not attributed to a single entity, such as TEPCO, or NISA. Errors were made by GE, Ebasco, Toshiba, and Hitachi as well. The fundamental premise is that humans are fallible, and, as such, they cannot create infallible products or systems. Although the product and services suppliers must meet the necessary codes and standards for their products, this approach does not always produce optimal results. For example, it appears logical that parties involved in the planning, design, and construction of the sea wall at the Fukushima Daiichi station did not adequately examine past data pertaining to the wave height in that part of their country, or did not have access to wave height data. The Fukushima Daiichi nuclear power plant had a seawall that was designed to withstand a tsunami with waves 5.7 meters (18.7 feet) high. The wave that hit the plant was 15 meters (49 feet) high. Dr. Peter Yanev is one of the world's best known consultants in designing nuclear power plants to withstand earthquakes and tsunamis.[38] He stated that the seawall at the Japanese plants probably could not handle the tsunami waves that struck them. Further, he stipulated that diesel generators were located in a low spot with the assumption that the seawalls were high enough to protect against tsunami waves. This assumption led to be a fatal miscalculation. It apparently is not enough to predict and forecast an impending earthquake. When predicting and forecasting an an impending earthquake, it is also essential to predict the size of the resulting tsunami "The underestimation of the seismic hazard provides evidence of systemic problems in disaster prediction and management."[36] The Japanese scientists should have cautioned key facilities to comply with their findings. As discussed later in this manuscript, the Japanese Society of Civil Engineers published a report on "Tsunami Assessment Method for Nuclear Power Plants in Japan",[39] prior to the Fukushima accident. This report was not taken into consideration by the designers, plant owners, or the regulatory agency.

Table 3 [40] lists tsunamis that have occurred along Japan's coast since formal data became available. As the data reveal, there have been high flood waves in the last 125 years. The wave that hit the Fukushima Daiichi plant had a wave height of 13-15 meters, depending on the source[12].

[12] There are variations in stipulating the size of the wave that hit the Fukushima Daiichi plant. The values range from 13 meters to 40 meters. Some use the value of the wave at Sendai.

Table 3. List of tsunamis close to Japan's coast

Year/Dates	Events (Tsunami)
July 3, 869	Large scale tsunami
1596	The tsunami destroyed the island of Uryor Jima
1/26/1707	Large scale tsunami
1737	One of the largest recorded tsunami reaching 64-meter wave-after an earthquake
8/29/1741	Huge tsunami caused by a volcano that resulted a tsunami wave height of 16-meters [41]
6/15/1896	Waves as high as 38-meter high after an earthquake of 8.5 magnitude
9/1/1923	After the Great Tokyo Earthquake 11-meter high wave
3/3/1933	29-meter [42] wave height
5/1960	After the earthquake in Chile, the tsunami reached Japan
7/12/1993	Okushiri earthquake resulted in a tsunami with 6-10-Meter high wave

At a minimum, greater scrutiny and diligence should have been used in selecting a different location, or building a much higher wall to block waves that had been previously recorded at 38 meters. Although there was no code or licensing requirement specifying a barrier wall of a certain height, the plant designers should have considered building a to match the recorded wave height data. The ultimate responsibility for such a decision rests with the plant owner, in this case TEPCO. However, TEPCO management must have based their decision on the judgment of the entities that were designing their plant. Ebasco designed Waterford-3, Vermont Yankee, St. Lucie, Turkey Point and the Fukushima plants. Bechtel designed and built over 25 nuclear power plants, Sargent and Luncy built over 11 plants and Stone and Webster, designed and built 10 nuclear power plants.[43] This data is presented here to signify that Ebasco had the least number of nuclear power plants they designed in comparison with other architect engineering firms. The Ebasco design team likely had a significant input into designing such a barrier wall. Since erecting such a barrier wall was not in accordance with any specific code or design requirement, there appears no plausible basis to assign blame on the part of the design team for not specifying a barrier wall of a particular height. It appears that the risks associated with a short wall and high waves as a direct consequence of a tsunami were not diligently evaluated. In retrospect, we now know that the cost of erecting a 38-meter (or 35-meter as was suggested) barrier wall would have been far less than the estimated cost of this accident, $630 Billion.[1] During the past century in that region of Japan, there have been eight tsunamis caused by earthquakes of 7.7 to 8.4 on the Richter scale, [44] producing tsunamis recorded waves of 38 meters in the Tohoku region.[40] In light of these events, it appears questionable why the plant designers and plant owners would settle on erecting a 10-meter wall, even without a code required of a specific height. Engineering principles advise us not to pursue that approach, when there were plenty of indications that a short wall will be useless for high tsunami waves. This is, in fact, what transpired on March 11, 2011, at the Fukushima Daiichi nuclear power station.

General Electric, Toshiba, and Hitachi companies, knew the significance of the reactor core isolation cooling system and its dependence on DC power to function. Based on their differing geographical locations, there is a significant difference with respect to flooding of the battery rooms between the Quad Cities Nuclear Power Station (Sargent and Lundy was the architect engineering firm) and Fukushima Daiichi nuclear power station, both boiling water reactors-3 Mark

I plants. It appears that Toshiba and Hitachi replicated GE's plant design in other locations without serious consideration for the location of the location of the Fukushima plant and differing demands on the battery room location and providing a protective seawall.

4.2. The natural elements: earthquake and tsunami

It is not possible to accurately predict or to prevent a seismic event or the follow-up tsunami. This randomness in frequency, location, and size adds to the complexity in planning, designing components and systems intended to make nuclear power plants safer. That is, these unknowns increase the chances for human error. When thmentum of the tectonic plates oveerides resisting frictional forces, the plates move suddenly resulting in a seismic event.[45] Companies creating and maintaining energy facilities are required to consider these potential external forces and the damages inflict damage on the plant they design, fabricate, operate, and regulate in a specific location. However, they encounter a dilemma when historical records indicate that earthquake forces or tsunami height at a particular site are greater than required by code or commonly followed as standard practice. For example. At Fukushima a standard 10-meter wall was installed to prevent a wave from entering the facility, yet records indicated waves in the area had been as tall as 38 meters. As a result, when the battery and charger rooms were placed at relatively low elevations, the low wall allowed flooding, which rendered the DC power to fail and incapable of providing electrical power to the reactor core isolation cooling system needed for critical core cooling.

When the U.S. Nuclear Regulatory Commission (NRC) mandated upgrades in the training programs after the Three Mile Island accident, such as installing hydrogen recombiners, or venting the containment for Mark I containments, plant owners around the world should have heeded the recommendations. These recommendations were based on the probable occurrence of natural events such as earthquakes and tsunamis, but it is the probabilistic nature of these events that increases the likelihood of errors in judgment. Additionally, U.S. steps to divide its Atomic Energy Commission (AEC) into the NRC and the ERDA (later Department of Energy) should have served as a global example as well. These steps were taken to reduce the possibility of interest by separating the promotion of nuclear energy from the regulation of nuclear power plants.

4.3. Book review

In this manuscript, two books on the subject of this research are examined in detail. Book - 1 is described below and presents human error issues, whereas, Book - 2 is discussed in Section 5.1 and examines the flooding issue.

4.3.1. Book - 1

"Lessons Learned From the Fukushima Nuclear Accident for Improving Safety of U.S. Nuclear [Power] Plants". [46]

This book was published in a collaborative effort by a committee of 24 scientists and academicians. The book consists of contributions by the U.S. NRC, EPRI, NEI, INPO, the U.S. Embassy in Tokyo, the Japanese Embassy in the U.S., the Japanese government, and other prominent nuclear sources in the U.S. and Japan. This book was published by the National Research Council, in 2014.

4.3.2. General Objection 1

The first general objection to this book has to do with the background and qualification of its authors. Nearly all the authors had a Ph.D. in nuclear-related fields, but few, or none, of them had operating experience in a commercial boiling water reactor or had an SRO license. Without such experience, they were rendering judgment on the adequacy and accuracy of a complex system that can be rendered only after actual experience in such a plant.

4.3.3. General Objection 2

The authors of Book-1 disregarded the events that went wrong and errors that were made from the design phase, to the operation of the reactors, for nearly 40 years leading up to the accident. Instead, they lauded TEPCO for its conduct in the accident, in spite of the fact that nearly all other reports and investigators, including members of the Japanese government, assigned some responsibility to TEPCO for its short-sightedness and disregard for major issues. The authors of the book blamed natural events (the earthquake and the tsunami) for the accident and the extent of the consequences of the accident. The researcher's specific objections and observations are presented in the sections below.

4.3.4. Specific objection 1

On page 1, paragraph 2, the book states, in part, "However, several factors relating to the management, design, and operation of the plant prevented plant personnel from achieving greater success".[22] It does not appear that a 'success story' exists when the outcome is a tragic and preventable accident of this magnitude. The accident resulted in a destruction of four out of six nuclear power plants causing damage exceeding $630 Billion.[47]

4.3.5.   Specific Objections – 2 through 13

Nowhere in the synopsis section (pages1 and 2) of Book-1 do the authors contend that the Japanese government and its nuclear industry should have heeded the recommendations that were formulated by the U.S. NRC. Nor do the authors refer to TEPCO's disregard for the U.S. NRC's numerous guidelines and mandates developed for U.S. nuclear power plants, which, if they had been followed, it would have diminished the extent of the disaster that befell the Fukushima Daiichi nuclear power station. It is true that the Japanese utilities are not regulated by the U.S. NRC. However, the U.S. NRC is the best qualified regulator in the world. Nearly all international regulators follow the NRC on matters germane to the regulation of their nuclear power stations. The NRC's voluminous Code of Federal Regulations is difficult to fabricate by other governments because other international regulatory agencies do not have adequate resources to create their own. Further, the six nuclear reactors installed at the Fukushima site were designed and fabricated by the American companies, i.e., the General Electric Company and Ebasco, and all American nuclear companies are regulated by the U.S. NRC. The authors of Book-1 instead refer to eight factors that prevented plant personnel from achieving greater success in preventing multiple core damages. It is the opinion of the researcher (the author) that the lack of references in Book-1 to 12 other important contributing items (items 2-13) render Book-1 less effective than it could have been. Those 12 items are presented below as specific objections to Book-1. They represent the opinion of this author based on his own reading, research and experience.

1. Too short of a barrier wall, initially designed to be 35 meters, but TEPCO installed a 10-meter tall wall. The wave striking the plant was 14 meters tall.
2. TEPCO's isolating itself from the reactor designer, GE, because it appears that TEPCO believed it was self-sufficient.
3. TEPCO's withdrawing from the boiling water reactor owners group deprived themselves of the development and the progression of the emergency procedures guidelines. Use of these procedures would have been effective in reducing damage to the plant. For example, the operators would have taken the plant out of hydrogen accumulation mode, thus potentially preventing hydrogen detonation.
4. GE's retaining the location of the battery and the charger room at the same elevation as the other boiling water reactors, which were not prone to flooding due to high waves. The book's authors and consultants should have identified this as a major oversight.
5. Ebasco, the architect engineering firm, allowed TEPCO to modify the external barrier wall from 35 meters to 10 meters, even though there was evidence indicating that the short wall was inadequate.
6. There was no reference to Toshiba and Hitachi using GE's design of the location of the battery and battery charger rooms, leaving them at low

elevation and making them susceptible to flooding. These entities accepted GE's design without rendering a judgment about the increased risk due to Fukushima's location. It appears that GE, Toshiba, Hitachi, and Ebasco had not examined the height of prior tsunami waves in that area.

7. Whenever something goes wrong in a U.S. nuclear power plant and a component fails, The NRC requires all licensees to perform a root cause analysis to determine the true cause of the failure in order to prevent recurrence. Any plant modification requires a 50-59 analysis according to federal regulation. When NRC inspects a licensee, the inspectors must evaluate the performance of the root-cause analysis used to make any changes to a plant's system and component that are safety-related. Yet, the NRC blessed this book without performing the root cause analysis of the failures, including contributions by the plant owners.
8. There was no reference in this book as to why TEPCO did not follow numerous NRC guidelines such as installing hydrogen recombiners, installing hardened vents, and following the emergency procedure guidelines of boiling water reactor owners group. The book indicated the plant had not installed hardened vents or hydrogen recombiners, yet the authors and contributors did not address why TEPCO opted not to add the recombiner. The boiling water reactor Mark I and II in the U.S. had to install a hydrogen mitigation system, even though for these plants,

they inerted[13] the containment. At Fukushima hydrogen detonation breached the containment integrity and caused the release of large quantities of nuclear radionuclides to the outside environment.

9. There was no reference as to why the plant had not tested the isolation condenser since its commercial operation began 40 years before. "A subsequent investigation by NHK [14] would later reveal that the isolation condenser units had not been tested or operated in over 40 years."[48] Testing of the isolation condenser would have made plant operators familiar with the steps for using the system during the accident.
10. The researcher questions why plant personnel did not have authorization from management to vent the containment. In fact, the plant manager decided on his own volition to vent the containment when he was instructed not to do so.
11. The lack of an accurate and adequate financial cost benefit analysis regarding the barrier walls by GE, Ebasco, Toshiba, and/or Hitachi might have encouraged TEPCO to consider a taller wall.
12. Lack of familiarity of the authors with the core design of a boiling water reactor is evident as the book makes its recommendation 5.1.A (the fourth bullet on page 7): "instrumentation for monitoring critical thermodynamic parameters in reactors, containment, and spent fuel pools".[22] There are no reasons given for installing flow orifices, pressure

[13] Inerting is filling the primary containment with nitrogen gas to reduce the oxygen content of the environment to mitigate hydrogen detonation.
[14] NHK (Nippon Hoso Kyokai) is the Japan's largest broadcasting organization broadcasting in 18 different languages.

gauges, and power level sensors in the spent fuel pool or in the containment (except for pressure sensors in the containment which already exist). Clearly, improvements can be made to water level sensors to cover a wider range of water levels in the spent fuel pool, and water temperature, in case of inadequate fuel pool cooling using the fuel pool cooling system. However, there appear no reasons to install other instruments in the spent fuel pool such as flow sensors, pressure gauge, or power level. As long as there is an adequate supply of water in the spent fuel pool and cooling is being provided for the spent fuel assemblies, the irradiated fuel is safe.

4.3.6. The primary human causes of the accident

There were numerous human errors that contributed to this accident. The designers of the reactor system were: GE for Units 1, 2, and 6; Toshiba for Units 3 and 5; and Hitachi, for Unit 4. The architect engineering firms were Ebasco for Units 1, 2, and 6; Toshiba for Units 3, and 5, and Hitachi for Unit 4.[33] The site selection and approval, including considerations for seismic events and tsunamis, normally fell within the purview of the architect engineering firms.[49] Low sea wall height was the primary cause of the extensive damage to the plant. Had there been a sufficiently high wall, around 16 meters, it would have prevented flooding the plant. The maximuim wave height that hit the site, was 13-15 meters.[50] Among the various articles discussing this issue is one authored by Erik Hollenagel and Yushi Fujita.[42] In that article, the authors clearly state that the wall they initially considered was to handle 5.7-meter wave.

The authors state in Section 5.1, in the fourth paragraph:

"*The probability of a large earthquake hitting the affected areas was known to be very high, well before the earthquake actually hit the region. But the initial investigation apparently did not assume that more than [a] few faults would be activated simultaneously. With the Tsunami, the assumptions were also based on a historical review, but tsunamis are few and far between. The tsunami wall was designed with a height of 5.7 meters, although the reason for that is not clear. In 2002 the Tsunami Evaluation Subcommittee of the Nuclear Civil Engineering Committee of the Japanese Society of Civil Engineers published a report on "Tsunami Assessment Method for Nuclear Power Plants in Japan". In 2008, TEPCO used this method to confirm the safety of the nuclear plants at Daiichi. But there is, of course, no way this could have influenced the design decision when the plant was built in the 1960s. After the Tsunami had happened, it became clear that a historical study had revealed that a much larger Tsunami occurred in the middle of the ninth century (estimated to be in AD 869), and that a researcher had made a strong recommendation for refurbishment of the plant in 2006. But, the recommendation was reportedly turned down for the reason that the tsunami was hypothetical, and because the claimed evidence was not accepted by specialists in the nuclear sector."*

The same authors in the same subsection stipulate:

"*One of the authors found meeting minutes in which an expert representing an electrical company said that it might be generally worthwhile to show that nuclear power plants would be able to withstand a Tsunami. The connotation for this person was that the result of [the] assessment was already given, before the assessment was actually conducted during the design process. In other words, the assessment was conducted to support a belief that the plant would be safe. It cannot be ruled out that such an atmosphere prevails in the nuclear industry – in Japan and elsewhere*."

The researcher does not agree with the article's conclusion concerning safety attitudes in industries located using the word "elsewhere." It is the researcher's opinion, based on his experience in the U.S. nuclear power industry of 27 years, that this type of ideology and outlook would result in such a person being dismissed from the U.S. nuclear power industry.

4.3.7. The loss of power issue

Loss of power was a primary cause of the plant's inability to cool the core, thus resulting in degraded core conditions in three of the six reactors. It appears that the battery rooms in the Fukushima Daiichi plant were not in a seal-tight condition at the time of the tsunami. This is another example of poor judgment regarding safety-related components, given the increased likelihood of flooding due to wave height, the low barrier wall, and the relatively low elevation of the battery rooms. In other plant areas where there have to be a seal tight doors, the break in the seal is annunciated locally and in the control room. However, this protection is not a design feature for earlier boiling water reactors. Even if the battery rooms were not flooded, the battery chargers would have been unable to charge the batteries for longer than one day due to loss of AC power. One day of reactor core isolation cooling system the DC power is insufficient to handle the decay heat that lasts approximately seven days. It would be too harsh to criticize TEPCO for not having available additional diesel generators in standby condition in case of such an accident, even though it is now mandated by the NRC for U.S. nuclear power plants.[51] The station blackout is an analyzed event and the licensees use probabilistic risk assessment to evaluate the core damage frequency under the station blackout condition. However, station blackout condition, assumes the availability and functionality of DC power and the reactor core isolation cooling system (requiring DC power).

4.3.8. Missed Opportunities by TEPCO

In a recent report issued by the Science and Technology Journal, [52,53] the author states:

> "*In the peer-reviewed Philosophical Transactions A of the Royal Society, researchers Costas Synolakis of the USC Viterbi School of Engineering and Utku Kânoğlu of the Middle East Technical University in Turkey distilled thousands of pages of government and industry reports and*

hundreds of news stories, focusing on the run-up to the Fukushima Daiichi disaster in 2011. They found that "arrogance and ignorance," design flaws, regulatory failures and improper hazard analyses doomed the coastal nuclear power plant even before the tsunami hit."

Dr. Synolakis, professor of civil and environmental engineering at USC Viterbgi stipulated that "there were design problems that led to the disaster that should have been dealt with long before the earthquake hit,".[52] "Earlier government and industry studies focused on the mechanical failures and 'buried the lead.' The pre-event tsunami hazards study, if done properly, would have identified the [lack of] diesel generators as the linchpin of a future disaster. Fukushima Daiichi was a sitting duck waiting to be flooded."[52] It appeared that TEPCO underestimated tsunami height, relied on its own faulty data and incomplete modeling and ignored warnings from Japanese scientists that higher flood waves were possible. Prior to the accident, TEPCO estimated that the maximum possible rise in water level at the Fukushima Daiichi nuclear power station was 6.1 meters, a number based on earthquake of 7.5 magnitude, even though the record existed that earthquakes of 8.6 were recorded at the plant site.[52]

In the article referenced above, there are indications that prior to and during the accident, TEPCO made decisions and missed opportunities at the plant that contributed to the extent of the accident. Those indications are summarized below:

4.3.8.1. Independence from GE services

Apparently, TEPCO's decided to manage their plant independently from their reactor manufacturer, General Electric. As a part of that decision, TEPCO did not engage GE Nuclear and their many offers of products and services, such as issuing Information Notices and Service Bulletins to their customers based on its customers' experience with their product. Not training TEPCO's engineers and plant personnel using GE's training programs and training facilities was not the best practice for the Fukushima Daiichi station. TEPCO may argue that they were linked with GE-Hitachi when they merged. However, the merger took place in 2007 and the Fukushima station went commercial on March 26, 1971, providing a technology and information gap of 36 years. For example, after the Three Mile Island accident, most U.S. licensees and international utilities received numerous GE engineering training courses, including training on the subject of hydrogen accumulation and managing degraded core conditions. TEPCO's refusal to install a passive hydrogen recombiner for use in degraded core conditions is a testimonial to the lack of adequate training. Without this training, it appeared there was a lack of understanding of the consequences of not controlling the hydrogen as a consequence of melted uranium dioxide fuel and the reaction of the Zircaloy fuel cladding with steam.

4.3.8.2. Withdrawal from boiling water reactor owners group

TEPCO made another error in judgment by pulling out of the boiling water reactor owners group (BWROG). Because there is a difference between boiling water reactors and pressurized water reactors, two organizations emerged, a boiling water reactor owners group and a pressurized water reactor owners group (PWROG). GE was the principal agent for the boiling water reactor owners group and Westinghouse for the pressurized water reactor owners group. Not being a member of the boiling water reactor owners group, TEPCO deprived itself of access to a valuable tool, that being the formulation and progression of the emergency procedure guidelines. This program exists in every nuclear power plant, regardless of type, and it is the first thing the control room operating and supervising personnel use in cases of emergency situations. They follow the guidelines and the procedures step-by-step based on the indications and the evolution of plant conditions.

Although TEPCO and plant staff had access to their own emergency procedures, those procedures were not the same as emergency procedure guidelines, and were not as effective. Foe example, TEPCO did not vent the containment early enough, thus permitting hydrogen accumulation resulting in hydrogen detonation blowing four containments apart. It is unclear if GE-Hitachi made the emergency procedure guidelines available to TEPCO, and if they did, what version was on March 11, 2011. The hydrogen explosion at Fukushima was the consequence of losing containment integrity. Currently in a U.S. nuclear power plant, sets of emergency procedure guidelines are available in the main control room. In case of a transient, the plant operators under the supervision of a shift supervisor with an SRO license pull out the emergency procedure guidelines and pursue the pre-approved procedures each step of the way. As part of the certification and operator training requirement, accidents are simulated in a site-specific simulator. Currently, each U.S. nuclear power plant has its own site-specific simulator. In Japan, several nuclear power plants use the same simulator. The simulator training requires that the trainees use the normal and emergency procedures for simulated plant accident scenarios.

4.3.8.3. Not adhering to NRC mandates on safety of boiling water reactors.

As mentioned above TEPCO was not required to adhere to the NRC requirements and mandates. However, it is the opinion of this researcher that noncompliance with the NRC post-Three Mile Island recommendation was another error in judgment. Each U.S. nuclear power plant receives an operating license in order to produce power for the licensee's customers. All U.S. licensee are required to comply with the plant-specific 'Technical Specifications,'[15], the terms of the operating license, and with all the applicable codes of 10 CFR Parts, 1-199.[16] There are numerous ways for the NRC to enforce conformance to the provisions of the existing codes of federal regulations. This point is emphasized because without such enforcement, an ineffective Japanese regulator was partially responsible for the evolution of this accident. This point is clearly stipulated by a TEPCO spokesperson that the company knew safety improvements were required before the accident, but had failed to implement them.[54] In a report released in July of 2012, a "parliament-appointed panel criticized years of 'collusion' between TEPCO, industry regulators and politicians, and described the disaster, as 'man-made'.[54] When there is a collusion between a regulator and an entity that is being regulated, the consequence cannot result in an effective enforcement ".[54]

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[15] Technical Specifications is an Appendix to the operating license that is issued by the U.S. NRC. All nuclear power plants retain a copy of the Technical Specifications in the control room and maintain the reactors according with the specifications contained in this document. It is called the plant 'Bible'.

[16] The use of the word 'applicable' is a protective clause as there are some parts of the 10 CFR document do not yet exist within subparts 1-199.

4.3.8.4. Not installing hardened vents for containments

On 09/1/1989, NRC issued Generic Letter 89-16,[55] recommending that boiling water reactors with Mark I containment implement corrective actions by installing hardened vents to mitigate the consequences of a severe accident condition. A pathway from the primary containment to the suppression chamber would allow the particulates and fission products to be filtered in the water of the suppression pool during a severe accident.[55] The changes that were recommended included (1) improved hardened wetwell vent capability, (2) improved reactor pressure vessel depressurization system reliability, (3) an alternative water supply to the reactor vessel and drywell sprays, and (4) updated emergency procedures and operator training.[55] The NRC was fully aware of the existence of the boiling water reactor owners group and the development of the emergency procedure guidelines. They were cognizant of the significance of early venting during accident conditions as they were invoked in the emergency procedure guidelines. By not implementing the NRC order to implement hardened vents in their boiling water reactors, TEPCO provided another example of poor judgment by not complying with the NRC mandates.

4.3.8.5. Not installing hydrogen recombiners

The installation of a hydrogen recombiner or hydrogen burners was a critical element for the reduction of the hydrogen concentration, which could have prevented hydrogen detonation. Although the hydrogen burner would not have helped the Fukushima Daiichi station, as they require electrical power to burn off the hydrogen, the passive catalytic hydrogen recombiner would have prevented the hydrogen detonation blowing four containments apart, resulting in the release of fission products to the environment.

4.3.8.6. Inadequate planning to allow containment venting

One of the consequences of degraded core conditions is the increase in hydrogen generation, which may result in detonation, thereby, compromising the integrity of the containment. This occurred at the Fukushima Daiichi station and was the cause of the breach of three containments, Units 1, 2, and 4. Plant management wasted time before venting for various reasons, including miscommunication, equipment failure and environmental conditions.[56] In August 2015, an external committee of experts tasked with examining TEPCO's performance, referred to its culture and strategies by stating: "There is an organizational culture of the company for officials to avoid clarifying where responsibility lies and implementing planned countermeasures".[2] This report is highly critical of TEPCO for its lack of transparency about the true state of the accident and the radiation release to the environment.[2]

4.3.9. Conflict of Interest between NISA and TEPCO

A major issue in Japan was the conflict of interest between the Japanese regulator, Nuclear and Industrial Safety Agency (NISA), and TEPCO, which owned and operated the Fukushima Daiichi and other nuclear power stations. This is a consequence of the revolving-door practice of the regulators seeking employment in the industry, including TEPCO, after completing employment as a regulator. This is referred to as "Amakudari" (descent from heaven).[57] This conflict of interest results in relaxed regulatory enforcement and ineffective regulation.[57] Because of this factor, the new policy created after the Fukushima Daiichi accident prohibits the former regulatory employees from joining the industry for five years.[57]

Within the nuclear community in Japan, there were several governmental agencies regulating the nuclear power industry to provide public safety. The main regulating agency was the Nuclear and Industrial Safety Agency (NISA), within the Ministry of Economy, Trade, and Industry (METI).[58] NISA used the Japan Nuclear Energy Safety Organization (JNES), formed in 2003, to perform on-site safety inspections. The Ministry of Education, Culture, Sports, Science, and Technology (MEXT) was responsible for protecting the environment, promoting nuclear energy, and conducting nuclear safety regulation for the research reactors. The Nuclear Safety Commission (NSC) was an independent agency under a cabinet office to develop and administer safety regulations and monitor the work of NISA and MEXT. Additionally, the Japan Nuclear Technology Institute (JANTI), shared information and best practices on safety issues while the Federation of Electric Power Companies (FEPC) served as the lobbying arm of the nuclear industry.[58] This arrangement reduced the role of an effective and independent Japanese regulatory agency. An important factor was NISA's position within METI which was not at all independent. Other world regulatory bodies have been established with full autonomy from the organizations they regulate. "Although NISA was an independent agency from METI, the reality was that NISA was routinely overpowered by industry interests when it came to promoting rigorous safety standards".[58]

Appropriate research into earthquakes and tsunamis would have led to changes in the Fukushima Daiichi station that could have prevented the severity of the accident. For instance, both TEPCO and NISA were aware of studies relating to near-term probability of a powerful earthquake.[58] Such earthquakes clearly can cause tsunamis, and the greater the magnitude of the earthquake, the greater the tsunami waves, all other conditions (distance and duration) being the same. These entities either knew, or should have known, that such a tsunami would exceed the Fukushima's design basis value. This scenario did not lead to any regulatory requirement that NISA imposed on TEPCO. "Rather, TEPCO, like other utility companies in Japan, was left on its own to incorporate new information into accident management plans and to take countermeasures at its leisure. After the new guidelines were issued in 2006 by NISA to assess earthquake-related risks, TEPCO was allowed to unilaterally announce in 2009 that it would not report on its progress in conducting seismic 'backchecks' or take appropriate countermeasures that year (as requested by NISA) – and, in fact, would not do so until 2016".[58]

"*Also troubling was [the fact] that utilities were under no specific obligations to prepare adequate countermeasures in an accident; rather, everything was voluntary. In its report to the IAEA, Japanese regulators on May 10, 2013 frankly acknowledged that ' accident management measures are basically regarded as voluntary efforts by operators, not legal requirements, and so the development of these measures lacked strictness. The guideline for accident management has not been reviewed since its development in 1992, and has not been strengthened or improved'. Given that Japan has experienced multiple criticality accidents since that time and is vulnerable to earthquakes, tsunamis, and other natural disasters, there was no excuse for not reviewing these guidelines for nearly twenty years, or for making them regulatory requirements. The outcome at Fukushima was a plant that was [greatly] under-prepared for a severe accident; its disaster response plans called for only one stretcher, a satellite phone, and fifty protective suits"*.[58]

A GE employee reported safety problems at the Fukushima Daiichi plant. His identity was reported to TEPCO, resulting in the loss of his employment with a potential of becoming blacklisted from the industry.[58] "In order to ensure effective regulatory oversight of the safety of nuclear installations, it is essential that the regulatory body is independent and possesses legal authority, technical competence, and a strong safety culture".[58]

4.3.10. Failure of the Japanese regulator

The U.S. NRC has one major purpose mandated by the U.S. Congress. The agency regulates the nuclear industry to ensure that the public is safe and protected from the adverse consequences of the facilities. NRC does not promote nuclear power or have concerns for the economic advantages of this type of energy. In earlier years, the AEC both regulated and promoted nuclear power. Clearly, there was a conflict of interest, and due to pressure from citizens, the U.S. Congress dissolved the AEC through the Energy Reorganization Act of 1974. This act created two government agencies: the Energy Research and Development Administration (ERDA) and the Nuclear Regulatory Commission (NRC). On August 4, 1977, President Jimmy Carter signed into law the creation of the Department of Energy (DOE), which was responsible for the promotion of nuclear power, and the NRC for the regulation of nuclear power. The structure of these two organizations has remained the same since 1977. The Japanese government should have followed suit as the major cause of the Fukushima Daiichi accident was the lack of accountability of TEPCO to the regulator. All entities that analyzed the events of the Fukushima Daiichi accident agreed on this point, including the Japanese government and the investigating commission for this accident.[37] In fact, one of the first major changes after the Fukushima accident was to reorganize the former regulatory body into an entity dedicated solely to regulating the nuclear power stations in Japan. Events such as Bhopal and Chernobyl are other examples of cases where the regulators did not play an effective and independent role. This condition is normally referred to Regulatory Capture, [59] a form of government failure occurring when a regulatory agency, that is created to act in the interest of public and society, instead it advances the commercial or political benefits of interest group. The citation invoked above provides numerous examples of such a condition, including the accident at the Fukushima Daiichi nuclear power station.

In one case in the U.S., a senior member of a licensee was forced to retire as a consequence of a push made by the NRC behind closed doors. The person did not maintain a sufficiently conservative attitude towards operating a nuclear power plant under his control. This scenario is instructive; particularly because such a power was non-existent in the Japanese nuclear power regulatory hierarchy at the time of the Fukushima Daiichi accident. The NRC is a powerful entity that regulates nuclear power plants in the U.S. Similarly, NISA should have the same type of power as U.S. NRC to regulate TEPCO. As a consequence of this accident, a major reform in Japan’s regulatory process was demanded, and on September 19, 2012, NISA was dissolved and a new independent regulatory agency known as the Nuclear Regulation Authority (NRA) under the Ministry of the Environment was formed.[60,61]

4.3.11. Japanese culture of self-reliance

The Fukushima Nuclear Accident Independent Investigation Commission, which was created by the Japanese government after the Fukushima accident, challenged a portion of the main story presented by the government and TEPCO.[61] In the report, as stated by Kiyoshi Kurokawa, the commission's chairman: "It was a profoundly man-made disaster — that could and should have been foreseen and prevented,"[61] and "and its effects could have been mitigated by a more effective human response." [61]

The report blamed "TEPCO's tepid response on collusion between TEPCO and the Japanese regulators," stating "they had all 'betrayed the nation's right to safety from nuclear accidents'." Further, the report stated that "TEPCO manipulated its cozy relationship with regulators to take the teeth out of regulations".[61] One of the results is a lack of urgency in reporting information and taking action. For example, the Chūetsu-Oki Earthquake of July 2007, shook the Kashiwazaki-Kariwa nuclear power plant with maximum seismic acceleration that exceeded the design assumptions. This accident added urgency to the efforts to take preventive measure.[58,62] To cite another case, the Japanese Nuclear Energy Safety Organization (JNES), one of the technical support organizations in Japan, released results of a tsunami probabilistic safety assessment (PSA) in December 2010. This was five months prior to the March 11, 2011, event. This study showed, with high probability, that tsunami levels above a certain height would result in a degraded core condition. However, these results were not relayed to the regulatory body, NISA.[58]

CHAPTER FIVE – FLOODING OF NUCLEAR POWER PLANTS

This chapter is dedicated to the third objective of this research delineated in Section 1.2 of this manuscript, as the issue of flooding and its impact on nuclear power plants. The discussion will begin with a book review that focuses on the impact and the consequences of flooding on nuclear power plants.

5.1. Book - 2

The book is titled Flood Hazard for Nuclear Power Plants on Coastal and River Sites[63]. The book prepared by the International Atomic Energy Agency (IAEA), provides recommended safety requirements for site evaluations at nuclear installations regarding protection from floods for both coastal and river sites. It also provides guidelines for monitoring flooding events as well as numerous recommendations on how to design a plant with consideration given to the impact of flooding. The recommendations, if followed during design and construction of new plants, can mitigate the impact of flooding on a nuclear power plant. Considering that other parts of the world have the potential of designing new nuclear power plants, this book is an important tool for the plant designers and owners to use. This researcher, however, has three fundamental reservations about this book, and they are presented below.

1. The book focuses on those nuclear power plants that are in the design phase, and it is not for operational plants. The effectiveness of the book would be enhanced had it addressed nuclear power plants that are operational have provided approaches to retrofit those plants that are at high risk, in order to make them less vulnerable to catastrophic damage from floods.

2. A key aspect in in design is the reduction in the risk of failure, and a key aspect to understand risk is assessing vulnerability. This book does not mention the values of reviewing past flood data over a statistically significant period of time. Indeed, evaluating the history of floods or tsunamis at the site should have been a primary step in site risk evaluation, yet the designers of the Fukushima Daiichi nuclear power station did not examine the history of tsunamis.

3. In Item 3.13, the book reads:

"The potential for offshore seismic or volcanic activity and the vulnerability of the site to tsunamis emanating from both local and distant areas should be investigated even though no such waves from these areas may have been recorded over historical time."

The interpretation of this statement can be two fold. One interpretation is that the statement is fundamentally flawed, as it is clearly established that high tsunami-based waves have been recorded at many times. A countering interpretation of the above statement is that even if there were as no floods, the vulnerability of the site to tsunamis should be investigated.

The findings, observations, and recommendations in this book are based on the reviews of reactor operations of global nuclear power plants in recent years. The authors should have examined the impact of flooding prior to recent years as years. Flooding is a parameter of external causes of problems for industrial facilities, including nuclear power stations, especially those installations at coastal locations.

5.2. The nature of floods at nuclear power plants.

The book stipulates that 80% of all tsunamis occur in the Pacific Ocean. This is a rather large percentage of the tsunamis that occur on Earth and result from the “ring of fire” that encircles the Pacific Ocean, where the majority of seismic activity occurs. The main floods that affect the nuclear power plants are those due to high waves and the failure of man-made structures such as levees, sea walls, and dams. Floods due to precipitation or tides are not generally considered in design due to the extremely low probability of large consequences. Further, they often do not result in a total loss of AC and DC power simultaneously, as occurred at the Fukushima Daiichi plant. Another category of external flooding causing damage typically not considered in design, and that is the impact of ice blocks during spring melt.

5.3. Flooding of key safety-related equipment and spaces

Flooding poses a great risk to critical components of a safety-related system, making them fail, even if other components of the system remain functional. Effectively, a safety-system is only as robust as its most vulnerable critical component. Of particular importance are the electrical components, such as batteries, battery chargers, circuit breakers, electrical busses, circuit breakers, switches, and electrical circuits. It is noteworthy that a major flood can disable components by creating electrical shorts in electrical junctions and electrical circuits. Another cause of failure instigated by flood is submerging critical components under water. Further, if the flood water carries momentum, it can dislodge a component or break it loose from other components, which can become a projectile that damages another component by striking it. There have not been extensive analyses that break down the types of damage caused by floods in a commercial nuclear power plant. This contrasts to the large volume of documents, articles, and books on the adverse consequence of floods on residential and industrial businesses, and on how to clean up their aftermath. One explanation may be that there are a large number of fatalities as a direct consequence of residential and industrial flooding. The largest number of fatalities occurred in China in 1931, when between

Aside from electrical damage, waves, (especially large waves) exert tremendous static and dynamic forces against the walls of a plant and its equipment. These forces can move material and equipment, which may become projectiles that can strike and damage other equipment. Examination of the images from the Fukushima Daiichi accident reveals this source of damage. Storage tanks containing water, fuel oil (for diesel motors), and other liquids are prone to loss of integrity. Broken intake lines or discharge lines can expedite the loss of the liquid contained in their respective vessels. Luckily, the tank containing sodium pentaborate for the standby liquid control system is generally located at a high elevation in a boiling water reactor plant that cannot be reached due to flooding. The consequences of flooding, as with other sources of damage, are covered in the emergency operating procedures (EOP) developed and produced by the boiling water reactor owners group.[63] Flooding is specifically identified in the severe accident guide. Although the emergency procedure guidelines do not specifically address the impact of flooding on a specific system, it does address the impact of flooding on the emergency core cooling systems and the containment.

5.4. Failure of dams

The book also examines the failure of dams and the potential impact of the released water impinging on a nuclear power plant. Flood waters can reach most systems. The research may be useful information for designing a new plant close to a major dam, but this research did not examine the flooding issue caused by a dam failure. This research also did not examine flooding at river sites. In the U.S., there are no nuclear power plants that are subject to ice block movement. There are boiling water reactor plants (e.g., Nine Mile Point and James FitzPatrick) that are prone to the lake snow effect that can dump several feet of snow on these sites. However, no significant adverse experience has occurred in these plants.

5.5. Coastal floods, including tsunamis

The book covers the hazards associated with flooding, including tsunamis. It makes numerous recommendations for the designers to become well-informed in their design effort to consider the adverse impacts of tsunamis, especially coastal nuclear power plants. There are recommendations for the designers to become well-informed, including in the design phase, with the potential of adverse impact by tsunamis. Again, this book does not address how to prevent the impact of tsunamis on the existing nuclear power plants.

5.6. The impact of flooding on nuclear power plants

Water can become a common cause failure for safety-related systems of a nuclear power plant. The systems can include some or all of the emergency core cooling systems, emergency power supply systems such as the emergency diesel generators, batteries, or the plant switchyard. Losing all AC and DC power for longer than a day can lead to degraded core condition. This is what transpired at the Fukushima Daiichi nuclear power station in Japan. In addition, water pressure on the walls, foundations, and enclosures can challenge their structural integrity resulting in failure of the components maintained inside the walls.

Floods may impact transportation and communication between key organizations necessary for managing the emergency planning around the plant site. The effects of floods can jeopardize the steps necessary to implement safety measures and activities for emergency planning and managing the floods at the site. Flooding can also contribute to the spread of contaminants and radioactive particles released from a leaked source.

5.7. Flood Barriers

This research makes several references to the inadequacy of the sea wall at the Fukushima Daiichi nuclear power station and its to protect the plant by mitigating the consequences of a high wave. During the dissertation review process, a question was raised that an erection of a high wall may have a tendency to be toppled by a high wave, implying the higher the wall, the more probable the toppling effect. On the basis of a consultation with a Ph.D. at a civil engineering firm, who specialize in structural support and facilities, it is apparent that creating a high, effective wall against flooding is doable with an appropriate design.

5.8. Types of high flood walls

Although there are different types of high flood walls, selection of the appropriate walls depends on the environment, the foundation, the static and dynamic loading of the wall. High walls must sustain not only the impingement and forces of the flood waves, but also protect the content of the facility being within the wall. The wall designers take all of these factors into consideration when designing a sea wall protecting facilities. A ten-meter sea wall installed at the Fukushima Daiichi nuclear power station was inadequate to resist the tsunami waves it encountered on March 11, 2011.

6.1. Statistical treatment

The data used in this research is for the key components of 11 safety-related systems for a typical boiling water reactor are compiled in Table 4. The failure mode of each component, and its failure mode (regardless of the cause of the failure), was analyzed using the probabilistic risk assessment process.

Table 4. Partial list of BWR components used in the analysis

#	System	Power	Pump	Valves	Turbine	Signal	Timer	Water /Fuel
1	Diesel Generator	✓				✓		✓
2	Battery/Charger	✓						
3	High pressure core injection	✓	✓	✓	✓	✓		✓
4	Reactor core isolation cooling	✓	✓	✓	✓	✓		✓
5	Low-pressure cooling injection	✓	✓	✓		✓		✓
6	Low-pressure core spray	✓	✓	✓		✓		✓
7	Isolation condenser	✓		✓		✓		✓
8	Automatic depressurization sys			✓		✓	✓	
9	Safety relief valves			✓			✓	
10	Shutdown cooling system (RHR)	✓	✓	✓		✓		✓
11	Standby liquid control	✓	✓	✓		✓		✓

In the analysis, component failure is predefined, that is, failure-to-run, failure-to-open, failure-to-close and alike. Once a failure mode was selected the model determined the resulting core damage frequency. The results of each analysis indicated whether the failure, in of itself, would result in a core damage frequency of at least 1×10^{-5}, assuming functionality of the remaining components and systems. Three criteria are adopted for this probabilistic analysis as given below:

A. *Concurrent* failures of multiple components are not allowed. For instance, the low-pressure cooling injection system has two independent loops, each with two pumps. The analysis of the low-pressure cooling injection system did not consider the failure of both pumps simultaneously within a loop.

B. The analysis would terminate *when a failure* rendered a component *inoperable*. As an example, if a pump is assumed to fail because of lack of suction, causing cavitation, the analysis would not include the loss of power to that pump and other components.

C. Under the design base ruling that the NRC has established, the failure analysis assumes a single failure criterion. It does not require that more than one system fails at the same time with concurrent failure of its backup system. For example, when the high-pressure core cooling system fails, the design basis requirement does not consider that the automatic depressurization system fails as well.

6.2. Statistical model

The statistical model used for this research is the probabilistic risk assessment. It takes an analytical approach to determine risks, based on uncertainty analysis and failure frequency.[64] The probabilistic risk assessment is based on a model that was developed by Idaho National Laboratory, funded by the U.S. NRC for use by U.S. nuclear power plants. Essential input information to run this model included plant design, thermal hydraulic analyses of plant response, system drawings, performance criteria, operating experience data, emergency procedures, abnormal procedure guidelines, system operating procedures and maintenance procedures. Transients will become the initiating event, including turbine trip, generator trip, loss of off-site power and others. Probabilistic risk assessment, was applied by using event tree analysis that models the sequence of events from the starting point to an end point when the reactor is in safe condition. In the nuclear application, there are several levels in the output. Level 1 probabilistic risk assessment refers to core damage frequency. Level 2 probabilistic risk assessment refers to radiological release, and Level 3 pertains to the radiological consequences to public.[65] In this research, the scope of coverage performs the Level 1 PRA. Levels 2 and 3 are plant specific and require site-specific data and normal/emergency procedures.

Two illustrative graphs are provided (Figures 15 and 16) to help the readers understand the process of probabilistic risk assessment. Figure 16 [66] shows a sample probabilistic risk assessment for a simple case when a parachute fails to open. It illustrates the interface between the fault tree analysis and event tree analysis. Figure 16 illustrates the overall probabilistic risk assessment flow diagram providing the interrelationship among various facets of the probabilistic risk assessment. There are a few abbreviations used in Figures 15 and 16 as follows:

IE: Initiating event, it is the beginning of the probabilistic risk assessment.

LOCA: Loss of coolant accident – a notable accident where many of design basis of emergency core cooling systems are based.

LOOP: Loss of off-site power – a classic example of loss of all AC power from the grid, the plant relying on the diesel generators.

SBA: Small break accident – the basis of coolant accident that does not reduce the pressure rapidly, unlike a large break accident.

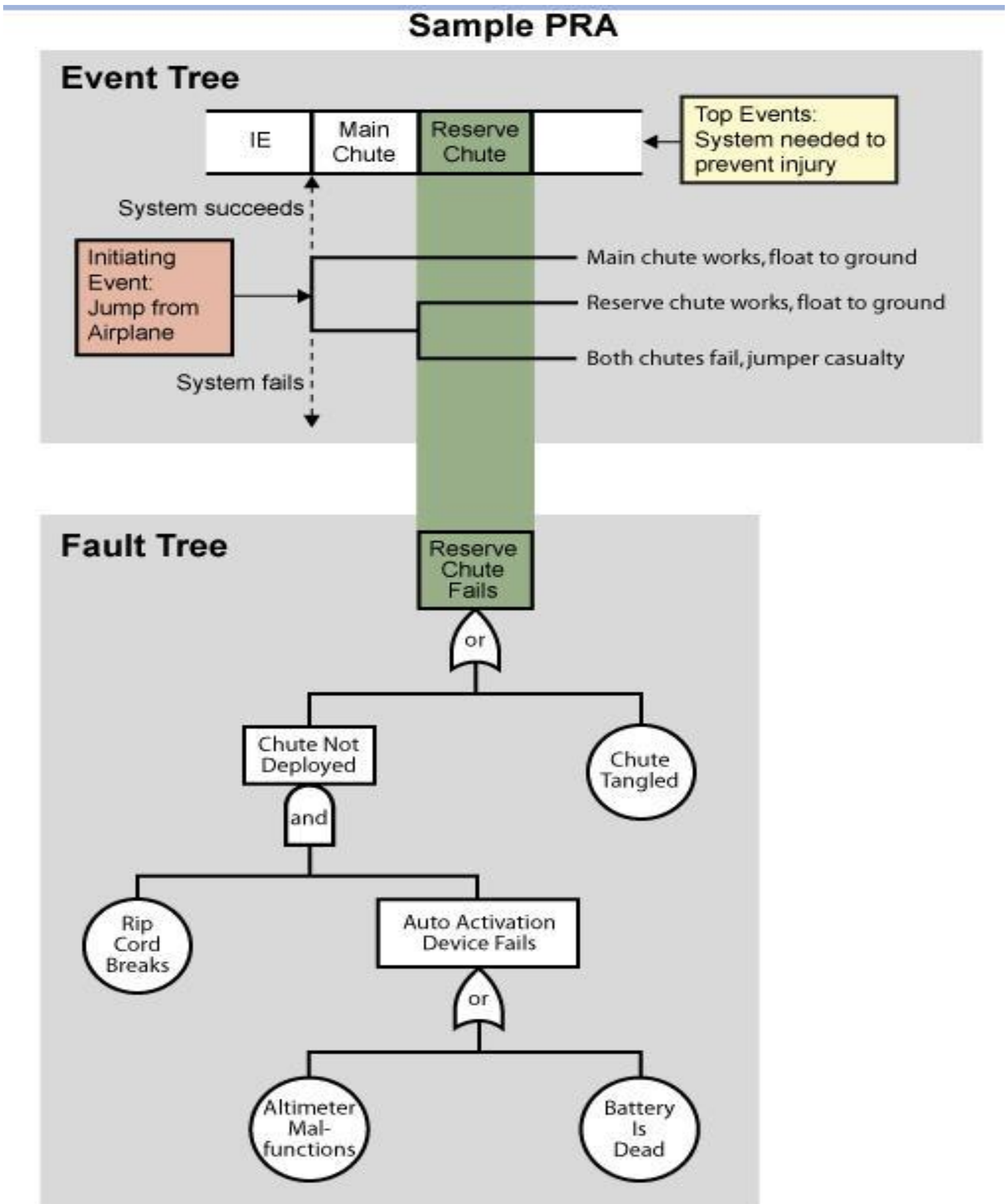

Figure 15. Sample probabilistic risk assessment in a simplified case [66]

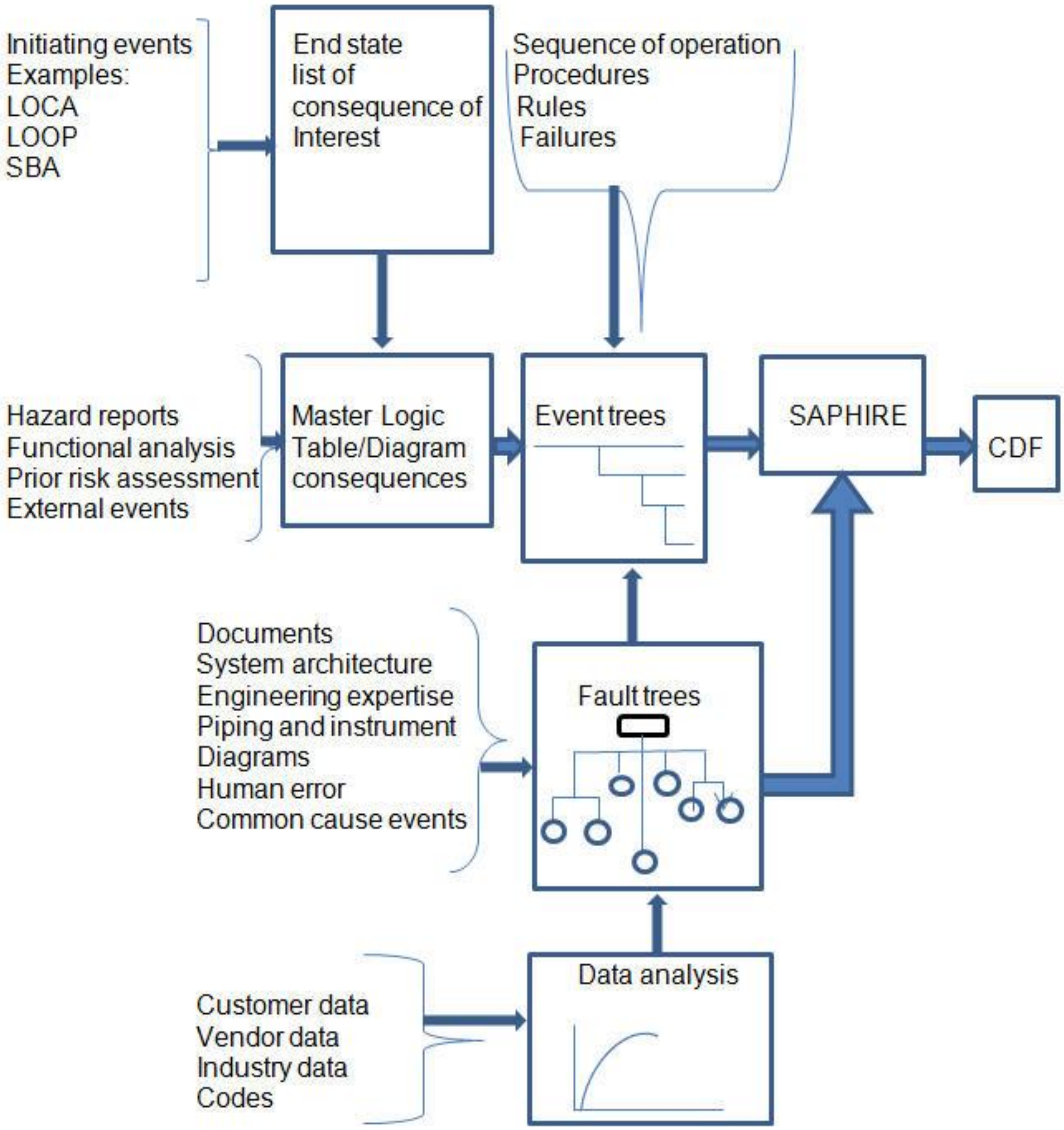

Figure 16. Probabilistic risk assessment flow chart

6.3. Process boundary for the analysis

The following sketch shows the process boundary used for the analysis. Different components provide input to various systems and the systems provide output. The output ranges from electrical power, AC and DC, to electrical signal and water flow. The recipient of the water flow is the reactor pressure vessel, and the containment for cooling purposes. The signals provide information and input to various logic and interlocks to protect the reactor and the containment.

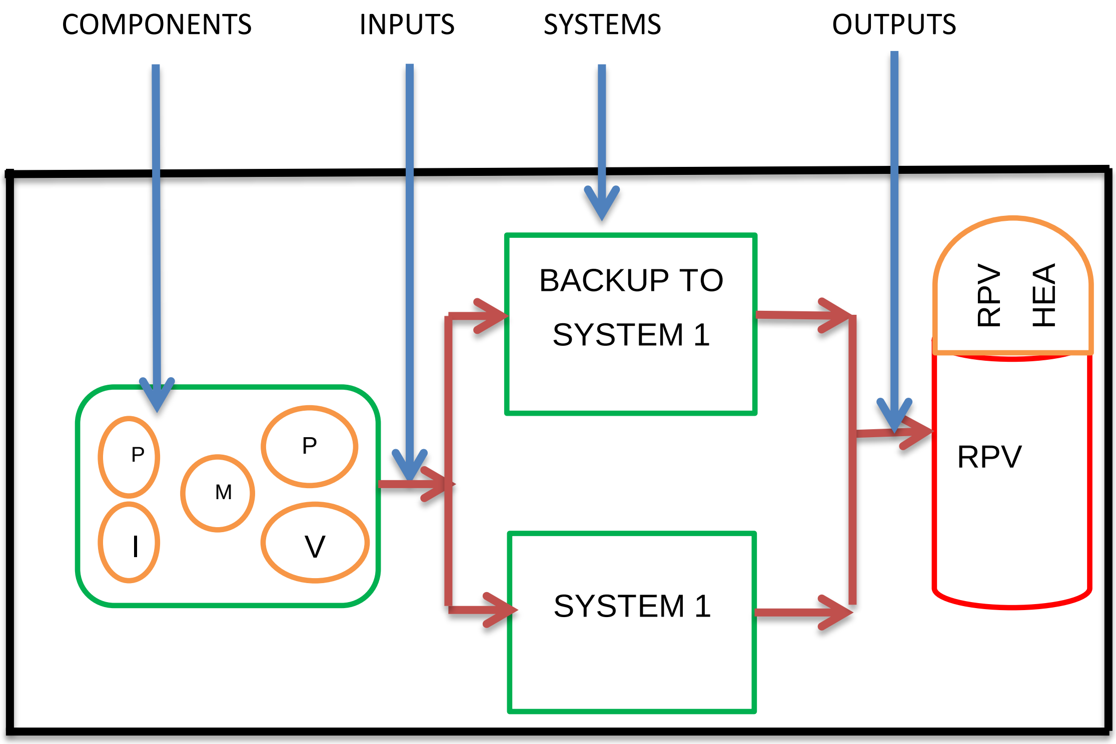

Figure 17. Process boundary for the analysis

COMPONENTS INPUT	EXAMPLE OF SYSTEMS	OUTPUT
P: PUMP POWER, WATER	HPCI, HPCS, RCIC, LPCI, RHR	WATER TO THE RPV
V: VALVES	DIESEL GENERATORS	ALLOW WATER FLOW
PI: PIPING	BATTERIES, SBGT, SBLC, ABD	ALLOW WATER FLOW
M: MOTORS	SAFETY VALVES, RELIEF VALVES	AC POWER
I: INSTRUMENTATION	ALL SYSTEMS	SIGNAL

6.4. Statistical Formulae

To calculate a total expected risk value of R (Risk) is obtained by using:

$$R = \Sigma \; F_i{*}C_i \quad [67] \qquad (E\ \text{-}1)$$

In the above equation, F_i is an individual probability of an undesirable event and C_i is its associated consequence. For instance, let us examine the following case: In a typical nuclear power plant, there are two redundant trains for a safety-related system. We assume each train consists of a pump and a valve. The pumps are driven by an AC-operated motor. There are two failures for the motor, one being failure to start (FTS) and the other, failure to operate (run) (FTR). The valve failure mode is a failure to open (VO).[67]

6.5. Method of the analysis

In this research, risk analysis is performed by the SAPHIRE software.[68] The software performs two separate, but related computation; one being the fault tree analysis (FTA), and the other being event tree analysis (ETA). The results of the FTA feeds the ETA. With this software, ETA is used to determine the core damage frequency, and to do that, it needs failure frequency of the components of the system for which the analysis is being performed. For the benefit to the reader, the system drawing for each of the safety-related systems are exhibited on Figures 5 through 13 of this manuscript. The reader should examine these drawings to identify individual components within each system. This is how the researcher identified the components of each system necessary for an individual system to remain functional. This was the foundation of the research analysis. (It must be noted that because the researcher had a previous a license to supervise operation of such a facility, the detail of the systems and components were quite well-known to him). If a system consists of a pump, motor, valves (intake and discharge), power source, piping, signal for activation, or other key components, it requires the failure frequency of each of these individual components. Such values of failure frequencies were obtained in a published document from the IAEA, or similarly from the NRC's NUREG/CR 6928. The next phase in the analysis was how these components are related to each other, i.e., by "AND gates" and "OR gates." The analysis also assigns the "BASE events" and "TOP event" (the undesired event). Using the relationship between based on these inputs and the failure frequencies SAPHIRE software calculates the core damage frequency. The content of Subsection 6.12 of this dissertation provides numerous examples of the FTA, ETA, and the values of core damage frequency calculated by the SAPHIRE software. There are other software programs that perform FTA, and from those analyses using the failure frequency data, produce the core damage frequency. One such program is computer aided fault tree analysis (CAFTA), developed by EPRI and used by most utilities to perform their core damage frequency. [69]

When the probabilistic risk analysis is performed by SAPHIRE, it does not use the causal factors that result in failure to function or failure to run of components. Instead, it uses the existing data based on the component operating and established data for failure frequency. If flood water reaches a component, the failure frequency of that component is assumed to be 1.0. The majority of the components in a nuclear power plant, except for the sump pump, are not designed to function underwater. If the component is in an "OR gate", the "TOP event" is reached by a failure of any component. If the component is in an "AND gate", it may not result in the TOP event [17] and will require the failure of all inputs. Thus, when there is a flood reaching a component, it results in the "TOP

[17] Top Event is an undesirable event that is identified at the top of the diagram in the FTA analysis

event”. If the flood reaches all components and all systems within the emergency core cooling systems are submerged, including the diesel generators, the Fukushima Daiichi accident can be replicated. This is in light of the reactor core isolation cooling system not being able to cool the core for longer than a day. In other words, under such conditions, the value of core damage frequency becomes 1.0. Core damage frequency of $1x10^{-5}$, one failure event in 100,000 years is acceptable. Within the resulting values of core damage frequency of $1x10^{-4}$ to 1x10-^{5}, the licensees have two options. One option is by way of analysis, demonstrate and convince the NRC that under such a scenario, there will not be any core damage. The other alternative is to propose modifications to the plant systems and components that would further change the core damage frequency from $1x10^{-4}$ to $1x10^{-5}$. The U.S. NRC will not authorize continued operation of a plant where a failure would result in a core damage frequency below $1x10^{-4}$ (one event in 10,000 reactor operating years) without making a plant or a system change. Since the post Fukushima Daiichi accident, NRC has changed its position and now requires for CDF of $5x10^{-5}$ and $1x10^{-6}$, the proposed change to a plant configuration is acceptable with further analysis. For values of core damage frequency of equal or greater than $1x10^{-6}$, a plant meets the required safety goals and no further analysis is required.[70] The unit of core damage frequency is the frequency of an undesirable event in a year.

6.6. Failure analysis of key components

This segment of the research includes the failure analysis of various key components within safety-related systems, when subjected to external conditions (seismic events and flooding) similar to the Fukushima Daiichi event. The PRA analysis will evaluate the plant performance and determine if such failure will result in a high value of core damage frequency, and if so, what would be that value? For example, when an earthquake occurs causing a probability of a core damage frequency larger than $1x10^{-5,}$ the plant owner must analyze the vent or undertake corrective measures to lower that probability. If there is an event or condition that yields a higher probability (larger core damage frequency), then by instituting modifications to plant configuration, a plant owner may be given the authorization to continue to operate. This research focuses on the prevailing conditions of a plant when a key component fails. If that failure occurs, based on the probabilistic risk assessment, with a core damage frequency smaller than the accepted value, no corrective measures are necessary. However, if it results in a higher core damage frequency, then corrective measures are envisioned and recommended. For instance, if a holding tank of emergency water supply becomes non-functional due to flooding and results in a core damage frequency of greater than $1x10^{-5}$, e.g., $1x10^{-4}$, a seal-tight enclosure would be recommended. Another alternative could be to place the tank on a seismically-qualified structural support (pedestal) at a higher elevation making it immune to flooding. This analysis was systematically performed for all key components of selected safety-related systems.

For boiling water reactors there are variations in the containment design: Mark I, II, and III. Each of these designs has its own individual features and characteristics,[71] and Mark I and II were considered in this research.

6.7. Probabilistic analysis

In general, the determination of risk of an undesirable outcome is based on a probability assessment. If an action is taken, there is a need to determine what would be the probability of a negative consequence. To a large extent, risk evaluation is based on the consequences and how impactful the ultimate outcome would be. For example, if the consequence of driving a vehicle at a speed of 90 miles per hour will result in the loss of driving privileges, the driver will not accept that risk. If, however, the driver knows there will not be a ticket for driving one mile over the speed limit, that risk may seem worth taking. In reactor safety, the “gold standard” for determination of risk is referred to in the Reactor Safety Study, published by the U.S. NRC, and referred to as: WASH 1400.[72]

6.8. Evaluation of probabilistic analysis

This part covers the risk analysis for a typical system within the emergency core cooling systems when certain components of the system fail to function. Once the source of the failure, or an undesirable event, is identified, the next step is to determine the risk, which has two components, and is the product of these components. One is the probability of the occurrence of the undesirable event, and the other is its consequence. Mathematically, it is written as:

$$R = P \times C \quad [67] \qquad \text{(E-2)}$$

Where R is the risk, P is the probability ranging from zero to 1, and C is the consequence. A zero risk implies a zero probability of an undesirable event, or negligible consequence, or both. The probability can also be interpreted as frequency of occurrence. Monitoring undesirable events results in determining failure frequency.

There are at least five different and redundant emergency core coolant systems (high-pressure coolant injection, reactor core isolation cooling, low-pressure coolant injection, isolation condenser, and automatic depressurization). Further, there are at least eight components in various system (pumps, valves, motors, turbines, power supply - AC or DC -, signals for activation, instrumentation, and piping). The risk of failure is calculated according to the following equation:

$$R_{(system)} = \frac{1}{N*S*M} = \frac{1}{100*5*8} = \frac{1}{4000} = 2.5*10^{-4} \qquad (E\text{-}3)$$

Where N is100 (number of the reactors), S is 5 (number of a system within the emergency core cooling systems), and M is 8 (number of components in each emergency core cooling system.) The risk of failure of an emergency core cooling system, R, can therefore, be 1/4000th of the published risk, $2.5x10^{-4}x10^{-6.}$

One of the key documents used and referenced in association with undesirable events in the nuclear power industry is known as WASH 1400. It was first published by the U.S. NRC in 1975. This document, modified in 2016, defines risk as three elements, identifying the source of an undesirable event (what can go wrong), its probability, and its consequence.[72]

In order to have 10 fatalities as a consequence of the failure of a component within a system, the risk is $1.5x10^{-9}$, which is the product of $2.5*10^{-4}$ and $1*10^{-6}$.

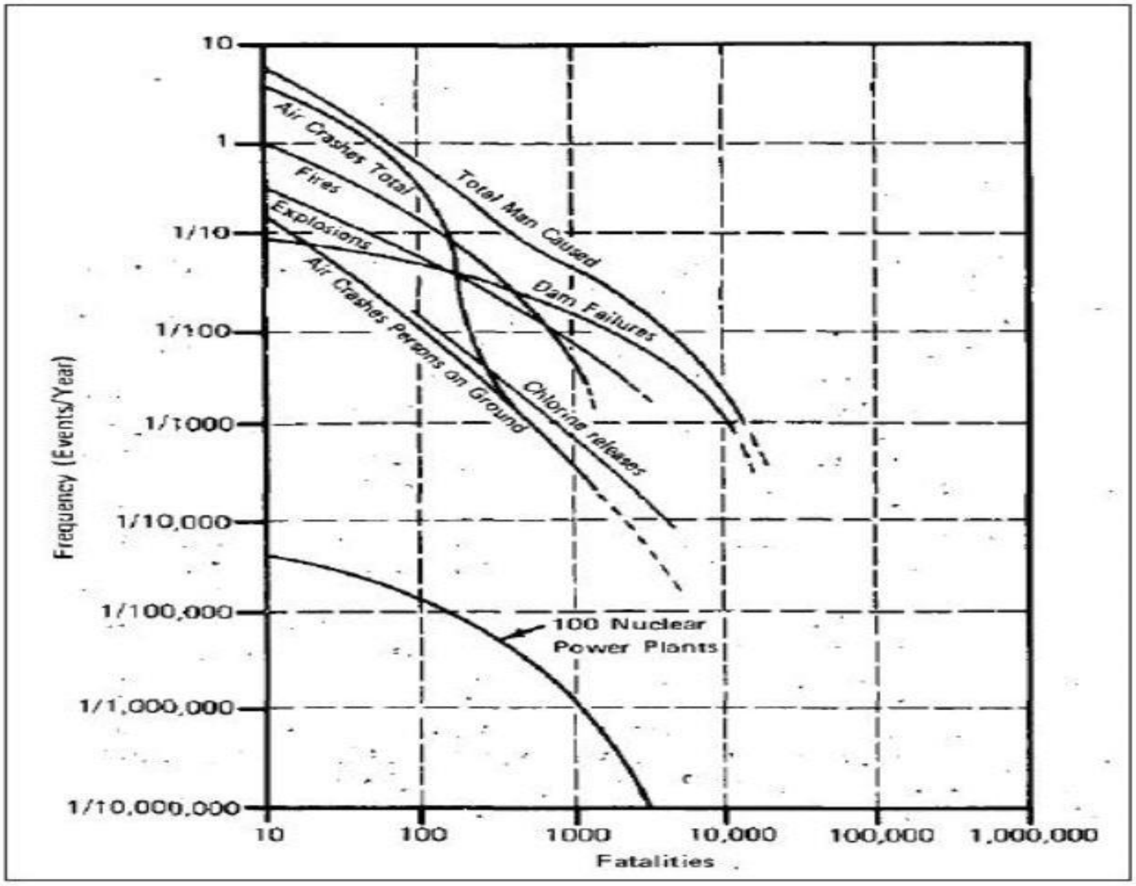

Figure 18. Fatalities from nuclear plants versus man-made causes from WASH 1400 Report

As shown in, Figure 18, the fatalities from the nuclear power plants, [73] 100 of them, are compared with fatalities from other man-made causes. In aggregate, the products of these three parameters will become 4000.

For 1000 fatalities, the risk will become 2.5×10^{-10}. There is also a correlation between failure frequency and damage. The higher the damage, the lower the frequency of occurrence.

In order to understand a system failure, one must examine the number of possible causes of failure of the individual components. These components are the reactor core isolation cooling turbine, its turbine-driven pump, the suction and discharge valves, the presence of the initiation signals and the associated piping.

There needs to be a provision that provides the failure analysis and presents the failure frequency of the individual components. To achieve this purpose, values are obtained from the IAEA document that lists failure frequencies of reactor components in a typical nuclear power plant. The document most often used, published by the U.S. NRC, is the NUREG/CR 6928.[22] To calculate the impact on the core damage frequency, both fault tree analysis and event tree analysis can be used. These analyses are discussed in a later section.

6.9. Performing probabilistic risk analysis

This step was accomplished using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE-8) software program. This software was funded by the U.S. NRC and was developed and managed by the Idaho National Laboratory. In addition to this software, which requires a license to acquire, there are other software programs developed by other firms and designed for U.S. utilities. The SAPHIRE software performs fault tree analysis and event tree analysis. These analyses are widely used by many researchers around the world. The FTA uses “AND gates”, “OR gates” and “TRANSITION gates.” As an input to these gates, there are “Basic Events.” A Basic Event can be a failure of a valve to open (FTO) and another could be a failure of a valve to close (FTC). Some analysts do not distinguish between FTO or FTC and use the syntax of a failure to function (FTF). If a failure to close prevents a valve from performing its function, it is a FTF. Any Base Event can cause a system failure if it goes through an OR gate. Most of the Base Events in this research went through OR gates. If a valve fails to open, a pump fails to run, power is lost, or there is loss of a signal for activation, that system will fail to function. Any of these Base Events will make a system such as a high-pressure coolant injection system fail to function. In an AND gate, there has to be more than one failure for a system failure to occur. For the low-pressure cooling injection system to fail to function, the Basic Events go through an AND gate. Since there are two low-pressure cooling injection loops, each loop with two independent pumps go through an AND gate, implying failure of both loops will cause the low-pressure cooling injection system fail to function. This holds true for the low-pressure core spray or isolation condenser. However, there is one high-pressure coolant injection system with only one turbine-driven pump. That FTA will reveal the failure goes through an OR gate. The SAPHIRE-8 software calculates the core damage frequency for each system based on the values of failure frequencies of the components, whether they are based on AND gates or OR Gates. In a complex system, it may comprise of several AND gates with OR gates. The results of these FTA and ETA for each safety-related system are included in Section 6.12 of this manuscript.

The root cause of the FTC or FTO were not investigated in this manuscript. There could be numerous factors causing a valve not to open. During a maintenance or modification process, the Code of Federal Regulations prohibits implementing a change without performing a root cause analysis. Performing a root cause analysis for a modification requires compliance with the provisions of 10 CFR 50.59. In the probabilistic risk assessment process, it is not required, or expected, to determine a root cause for why a valve did not open, a pump did not run, a circuit breaker did not close, a switch was not reset, a procedural step was not followed, or other causal factors. For example, when the researcher was the utility's coordinator for evaluating anything that had failed in the QCNPS, he met with NRC inspectors routinely to discuss the abnormal events and transients that had taken place, as well as the modification process. Within the probabilistic risk assessment process, often the analysts do not investigate the causal factors to find for a component failure. Further, there are no data that support such a breakdown, including the U.S. NRC's related NUREG that has been referred to multiple times in this dissertation. The values of the core damage frequency are given for each of the event tree analyses for the systems analyzed in this dissertation, Sections 6.12.1 through 6.12.10.

6.10. Failure analysis

Failure analysis is the end result of the research, and it was performed for each safety-related system. The first step of the failure analysis was to gather the failure frequency for various components within the targeted safety-related systems. For example, the high-pressure coolant injection system consists of a high-pressure coolant injection turbine, which is linked by a shaft to a pump. The other key components were inlet and outlet valves, check valves, instrumentation, suction piping, discharge piping, minimum flow line with its corresponding valves, and recirculation piping for periodic testing. (For a better understanding of this and other systems, the readers should refer to the schematic diagram of different systems shown on Figures 5 through Figure 13). For example, the steam isolation valve would, upon initiation, open up causing the reactor steam to hit the high-pressure coolant injection turbine propeller to cause it to spin. Upon spinning, the pump would suction water from a source, either the condensate storage tank (CST) or suppression pool (SP), to provide cooling water to the reactor pressure vessel, thus cooling the reactor vessel and the core. In order for the flow to take place, the pump inlet and outlet valves must open. The valves and the instrumentation are operated by the DC power. Failure of any of these components would prevent the return of the cooling water to the reactor. One key attribute of the high-pressure coolant injection system is that by taking steam from the reactor pressure vessel, it will cause a pressure reduction in the reactor pressure vessel. By lowering the reactor pressure, the low-pressure cooling systems such as low-pressure core spray or low-pressure cooling Injection, can be actuated to inject cooling water into the reactor.

Therefore, the failure analysis was contingent upon the failure frequency of various components of the system being investigated. Access to the failure frequency data was a challenge since there are limited sources of information. Since the focus of the research was on Fukushima, and it has a close tie with the IAEA, the data used were gathered by IAEA. The data were published in a document called “Component Reliability Data for use in Probabilistic Safety Assessment”.[23] Later, when the research used the NRC’s failure frequency data, which is a separate document. It became evident that using the IAEA document was the right choice for selecting the failure frequency.

To compare the differences between the two different data sets, a case was selected where a utility used the data based on the NRC document to perform their analysis, and those results were compared with the method used in this research using the IAEA data. This comparison was essential to demonstrate that the two different methods would result in relatively the same result. There were sufficient differences between the two processes that are stipulated for the benefit of the reader and thos who wish to emulate this process. Table 5 highlights the differences between the two different processes. One process reflects this research and the other is a process pursued by many utilities. The utility stipulated in this comparative analysis is a typical utility and not a specific utility. There is a need to mention that conducting a PRA for a site-specific utility is a highly proprietary information not divulged to a third party.

Table 5 Differences between the utility's process and the researcher's process.

PARAMETER	THE UTILITY	RESEARCHER
PROBABILISTIC RISK ASSESSMENT SOFTWARE	CAFTA	SAPHIRE - 8
DATA SOURCE	NUREG /CR 6928	IAEA DOCUMENT
BY COMPONENT	NO	YES
BY SYSTEM	YES	YES
FAILURE TYPES	BY SYSTEM	BY COMPONENTS
COMMON CAUSE FAILURE	BY SYSTEM	NO DATA
PERFORMING ETA	NO	YES

In the table below, Table 6, the differences in the results are shown.

Table 6. Results of analyses providing the core damage frequency by systems and comparing this research results with a utility's result

SYSTEMS	CDF BY RESEARCHER	CDF BY UTILITY
DIESEL GENERATOR (1)	8.09 E-3	
DIESEL GENERATORS (2)	6.5 E-5	
BATTERIES	2.15 E-5	
LPCI	1.39 E-5	3.0E-6
IC	6.37 E-6	N/A –(IC)
HPCI	2.79 E-5	
LPCS	1.39 E-5	
RCIC	9E-5	
SBLC (2 valves)	2.9 E-5	
SAFETY RELIEF VALVES	1.7 E-4	
ADS	1.9E-5	

Since the utility's run only used the LPCI, that was the value that was compared with the research's selected system. The difference between $1.39E^{-5}$ and $3.0E^{-6}$ is $1.09E^{-5}$. The ratio of 1.09 to 1.39 is 0.784, which implies, the difference between these two results are within 78.4% of each other. This signifies that different data sets and different software models provided results that were adequately close to each other. This close proximity authenticates the research process.

6.11. Calculation of the core damage frequency

This section discusses and provides a manual calculation of core damage frequency (CDF) without the use of the SAPHIRE software. Key terms are introduced in this discussion.

PRA is performed to include identification and analysis of initiating events and circumstances that places, in this application, a nuclear power plant in an abnormal condition. Initiating events (IE) are those events that are undesired states of the plant, such as a failure of a system to function, or a flood that can render a major challenge to systems and components in a nuclear power plant. In essence, an initiating event (IE) can prevent a safety system from performing its intended function.   Another example of an IE is an accident sequence. Depending on the sequence of events, the outcome could be either success or failure. If with the IE, a safety system performs its safety function, it is a success. If the IE causes a system not to perform its function, the IE has caused a failure. The successful response is that the plant transitions to a safe and stable end-state for the desired period of time. The PRA examines the frequency and consequences of NOT achieving a safe and stable end-state. The basic components of PRA are:

- Event trees to model the sequence of events from an IE to an end state.
- Fault trees to model the failure of mitigating functions, including equipment dependencies, to function as required.
- Frequency and probability estimates for the model elements (e.g., initiating events, component failures).

The outputs may include:

- Core damage frequency (CDF) ("Level 1" PRA)
- Radiation release frequencies ("Level 2" PRA)
- Radiological consequences to public ("Level 3" PRA).

In this research only Level 1 PRA is performed, as Levels 2 or 3 require access to a plant's procedures and containment designs.

In order to enhance the understanding of the readers to the analysis of various safety-related systems, there is a need to use a graphic image of the working relationship of various systems and components with the core. This image appears in Figure 19. In this figure, various safety-related systems are numbered sequentially from 1 to 4. For each system, the image shows three components. In real life, the numbers of components and the systems are different.

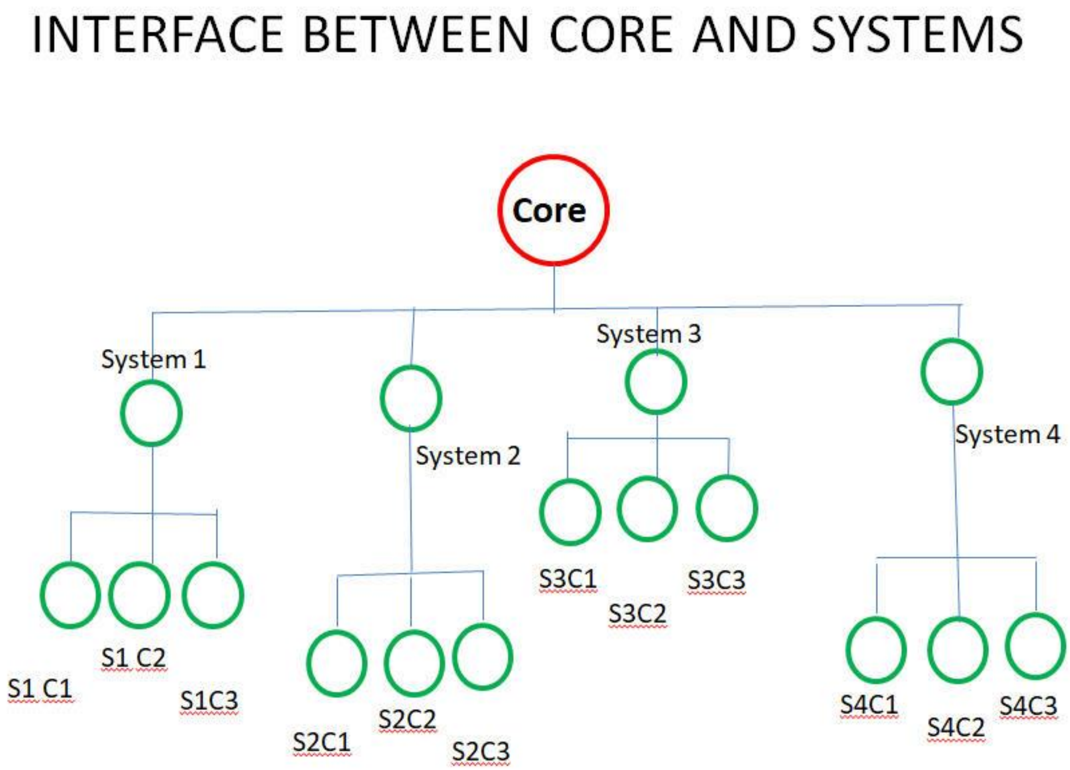

Figure 19. Interface between core, systems and components.

In Figure 19 the letter S is for system and C is for a component. S1 could be for shutdown cooling, reactor core isolation cooling, high-pressure coolant injection, or others. C could be for a pump, a valve, a switch, a turbine, or other component. The numbering scheme makes it easier to understand which component belongs to which system, Ultimately, all systems and components serve to protect the core.

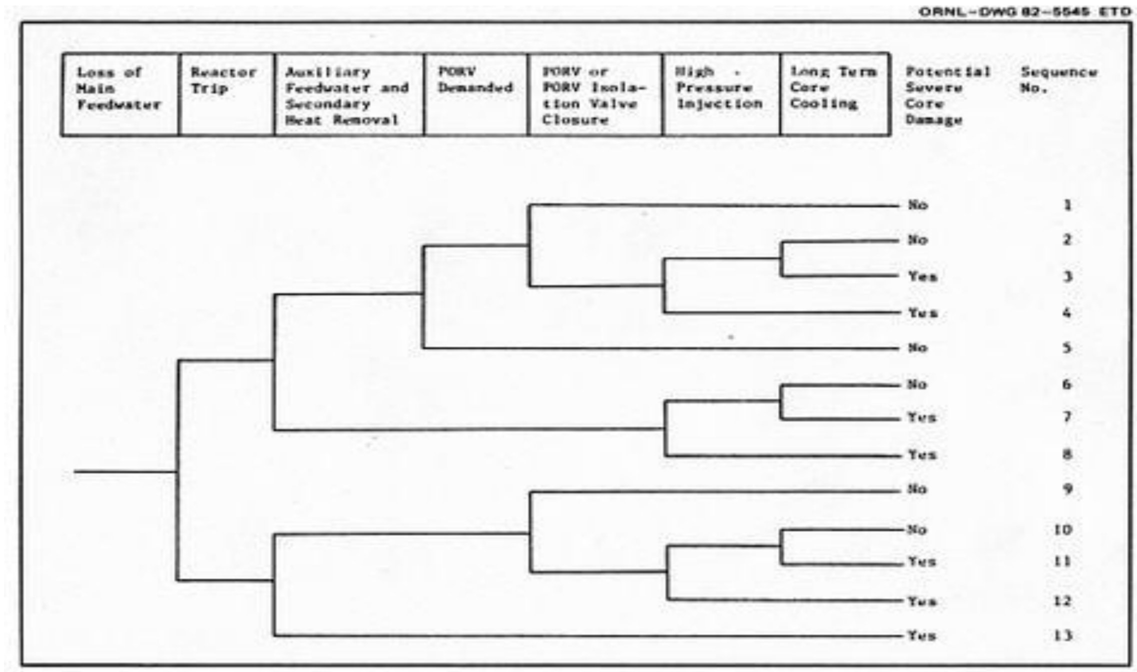

Figure 20. Provides an event tree analysis for a failure causing core damage[74]

The above figure, Figure 20, illustrates an event tree analysis of failures of multiple systems within the emergency core cooling systems resulting in core damage. The bottom entry is a transition stage leading to anticipated transient without scram. This is when the top event, reactor shutdown, does not take place using the control rod drives, which is the normal method to shut down the reactor.

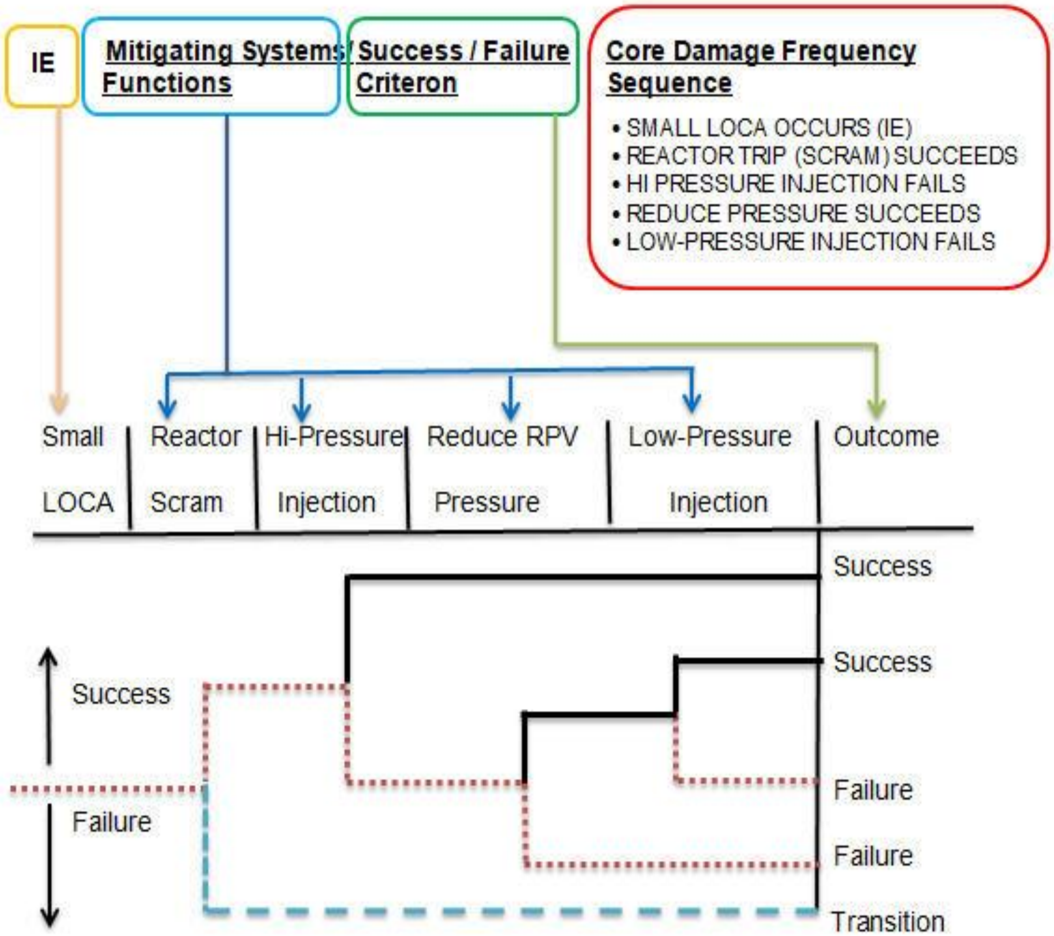

Figure 21. An event tree resulting in a failure causing core damage

In Figure 21 the dotted red lines depict the failure and black solid lines depict success. Transition (blue dashed line) is to proceed to the standby liquid control system as an alternative approach to shut down the reactor. The failure of high-pressure and low-pressure injections and the failure to reduce pressure in the reactor pressure vessel may cause core damage.

Success Criterion: Flow injection takes place (solid black line)
Failure Criterion: Flow injection does not take place (red dot line)
Transition state: (blue dash lines)

A SYSTEM WITH A FAILURE NEEDS DEVELOPING A FAULT TREE

Figure 22 shows a simplified power plant system consisting of a water supply tank, two independent pumps and two valves per each pump.

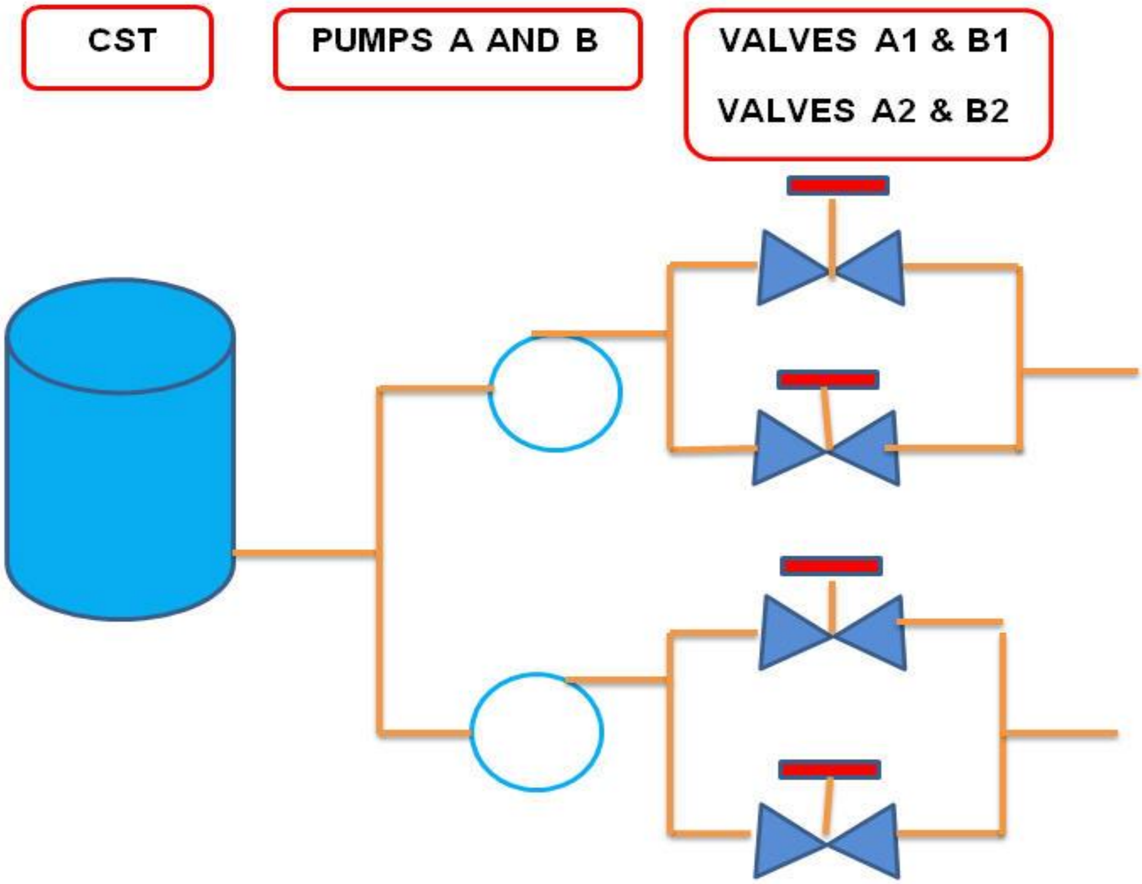

Figure 22. A simplified power plant system

SUCCESS CRITERION:

Flow from a tank, CST here, through the pumps A or B to the valves 1A, or 1B or 2 and 2 B injection path.

FAILURE CRITERA:

No flow from the tank, or

No flow from the pumps, or

No flow through the injection valves, through the available paths.

6.12. Individual risk for the safety-related systems

In order for a system within the emergency core cooling system to be functional, the roles of various subsystems need to be analyzed, including the diesel generators and the station batteries. The diesel generators and the station batteries are not part of the emergency core cooling system, but are essential for systems function. In the following 10 subsections, there are two figures for each system. These figures represent a fault tree analysis and an event tree analysis.

6.12.1. Diesel generators

The diesel generators supply AC power to run critical safety-related systems and components in a reactor during an accident condition when other sources of AC power are not available. Normally, AC power is supplied by the unit generator itself or the grid. The main generator connected to the unit turbine meets the power needs of the entire nuclear power station. This is referred to as carrying the house load. The remaining output of the main generator feeds the grid and is sold by the utility to its customers. During any event when a trip occurs in the electrical power supply from the grid, the reactor trips via the reactor protection system in anticipation of an adverse transient. At the same time the turbine and the generator also trip. Under this condition, the diesel generators that were in standby condition come to rated speed and remain in standby condition. Under this prevailing condition, the emergency core cooling system awaits the response from its logic to determine if there is a need to inject reactor coolant into the reactor pressure vessel. Receipt of low-low water level or high drywell pressure signifies an accident condition, whereas a loss of coolant inventory from the reactor pressure vessel has probably taken place. Under either of these two conditions, various systems within the emergency core cooling system, powered by the diesel generators, feed the reactor coolant into the reactor pressure vessel. Figure 23 illustrates two images of a diesel generator, mostly designed for earlier boiling water reactors. One image is a side view and the other is a front view. The front view shows a ten cylinder diesel generator.

Boiling water reactor -3 plants such as the Quad Cities and, Dresden reactors have three diesel generators for two reactors, where one unit is dedicated to each reactor and the third, a common unit, can be connected to either unit as a backup diesel generator. Whenever a unit diesel generator becomes nonfunctional, the common diesel generator on standby starts to pick up the load for the failed unit. When two diesel generators become unavailable, the utility has to make one unit available or proceed to shut down the reactor.

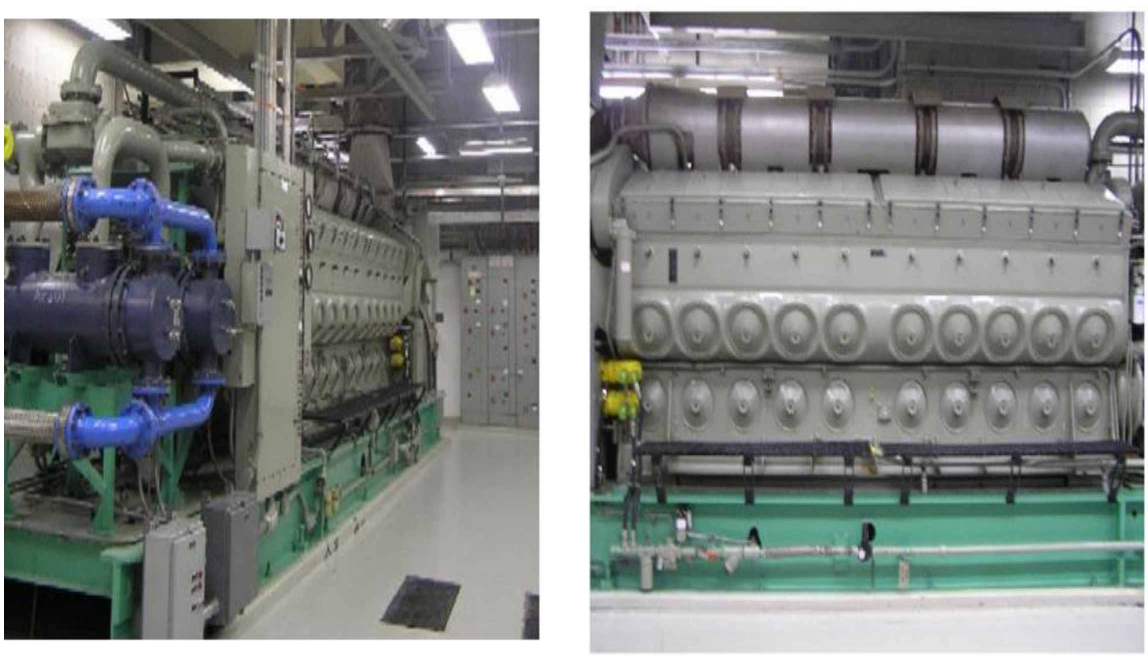

Figure 23. Two views of a typical diesel generator [75]

Next follows a series of figures showing the fault tree analysis and event tree analyses performed for various safety-related systems.

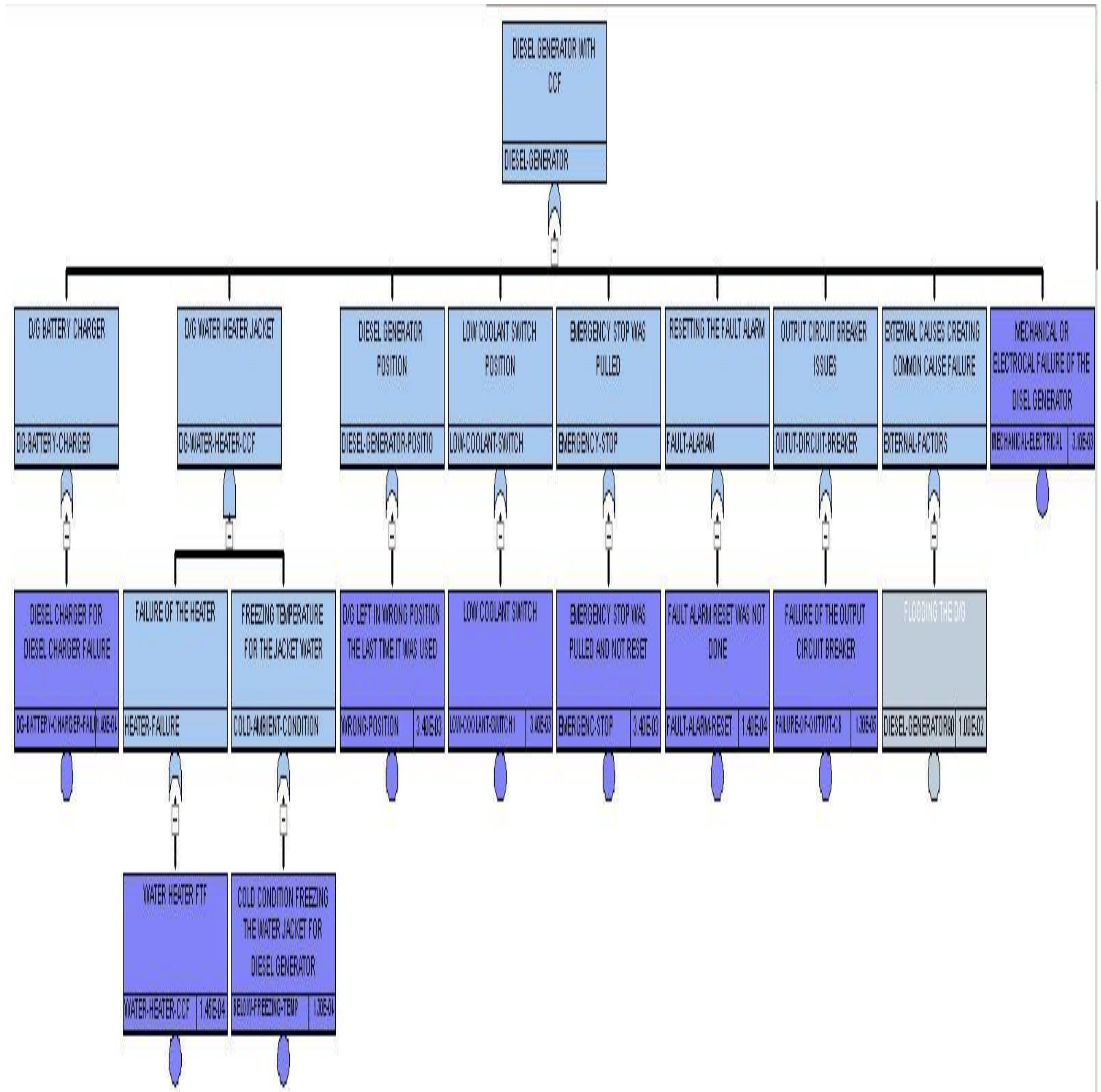

Figure 24. The fault tree analysis of a diesel generator

Figure 24 shows a fault tree analysis of a diesel generator.

For “and gate”: $f_t = f_1 * f_2 * f_3 * \ldots\ldots f_n$
For “or gate”: $f_t = f_1 + f_2 + f_3 + \ldots\ldots f_n$
$F_{t(D/G)} = 0.00125+0.003+0.003+0.0003+.00048+.00006=0.00809=8.09*10^{-3}$
This implies that the core damage frequency due to unavailability of one D/G is: **$8.09*10^{-3}$/hr.**

This implies that for one diesel generator to fail, the parts and starting pose the greatest portion of the contribution to the core damage frequency. However, for two DG to fail, the FDC will be **$6.5x10^{-5}$.**

Please note the values of the damage frequencies are extracted from an IAEA document and is attached as an html linkage directly below

http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/20/019/20019171.pdf?r=1

References from the above link

Please note that the bold letters are used to emphasize the result of the value for each code, which appear in the extreme left column. By using the codes, one can obtain the values specified here. Further, this cross reference is specified for this case. Other systems have their own respective codes and values. They do not appear in this manuscript. Also, please note the quotation marks before and after the content. These entries appear as in the original document, thus not subject to changes for abbreviations and other entries..

"DEIAE Diesel engine No.2 fuel oil, 4 stroke, in-line
Component boundary: detail n/a Operating mode: all Operating environment: normal
Generic failure mode: all modes, failure rate or probability
Source: IEEE 500 (1984) pg. 828 Component: Reference: NUREG 2232 rec Original failure mode: all modes
$6.5E^{-3}$/hr high: 6.5E·2/hr low: 6.5E-4/hr Ultimate source: expert judgment and experience

DEARW Diesel engine general
Component boundary: Complete plant, including starters pumps, fuel system.
Operating mode: emergency condition Operating environment: normal
Generic failure mode: fail to run Original failure mode: failure rate or probability median: **$3.0E^{-4}$/hr** 95%:
Source: WASH 1400 (table Ill 4-2) Ultimate source:
Failure to run given start **$3.0E^{-3}$/hr** 5%: 3.0E-5/hr ERROR FACTOR: 10
Assessed from industry experience and expert opinion

Comment: Diesel engine mentioned in this source is an engine used to run an emergency AC generator. Because of possible variation in redundancy of aux equipment, failure rates are separated for the engine and whole plant.

DGARB Diesel generator emergency AC
Component boundary: SEE IREP DG failure to start
Operating mode: standby Operating environment: normal
Generic failure mode: fail to run failure rate or probability mean
Original failure mode: **3.0E^{-3}/hr** max: failure to run, given start 2.0E-2/hr min: 6.0E-5/hr
Source: NUREG 2815 (table C.l.) Ultimate source: expert opn1on aggregation and IREP data
Comment: Failure to run is failure to run for more than 1/2 hour, given a start. Failure rates applicable to emergency condition.

DGARU Diesel generator emergency AC Component boundary: detail n/a
Operating mode: standby Operating environment: normal
Generic failure mode: fail to run. Original failure mode: failure to continue operation
Failure rate or probability, mean: **3.0E^{-3}/hr**
Source: Sizewell B (pressurized water reactor/RX312 pg.13) Ultimate source: assessed from nuclear and industrial experience and data
Comment: assessment based on W data item, WASH 1400 and 3 SRS data items, **(3.0E^{-3}/hr**) (1.3E-3/hr op.exp.8.7E+6hours) (1.4E-3/hr applicable to average industrial use).

DGASE Diesel generator emergency AC
Component boundary: SEE IREP DG failure to start Operating mode· standby Operating environment: normal
Generic failure mode: fail to start original failure mode: failure to start
Failure rate or probability mean: **6.0E^{-5}/hr** max: 4.0E-4/hr min: 3.0E-5/hr Source: NUREG 2815 (table C.1.) Ultimate source: expert opinion aggregation and IREP data Current: Failure to start is failure to start, accept load and run for 1/2 hour.

DGASW Diesel generator emergency AC
Component boundary: complete plant, including starters, pumps and fueling system operating mode: standby operating environment: normal Generic failure mode: fail to start Original failure mode: failure to start Failure rate or probability: median: **3.0E^{-2}/d** [**1.25E^{-3}**/hr.] 95%:1.0E-^{3}/d 5%: 1.0E-^{2}/d Error Factor: 3
Source: WASH 1400 (Table III 4-2) Ultimate source: assessed from nuclear and industrial experience and data “Comment: None”

6.12.2. Station batteries and battery chargers

Figure 25 exhibits the fault tree analysis for the batteries and chargers

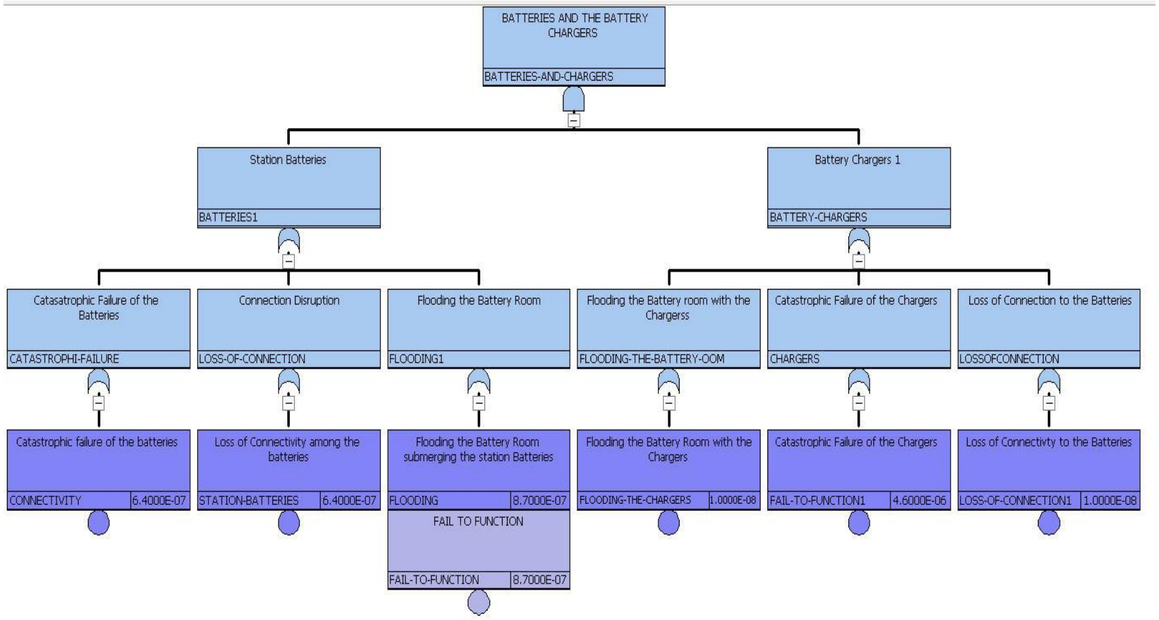

Figure 25. Fault Tree Analysis for Batteries and Battery Chargers

A graphic presentation of an event tree analysis of the station batteries and chargers appears in Figure 26.

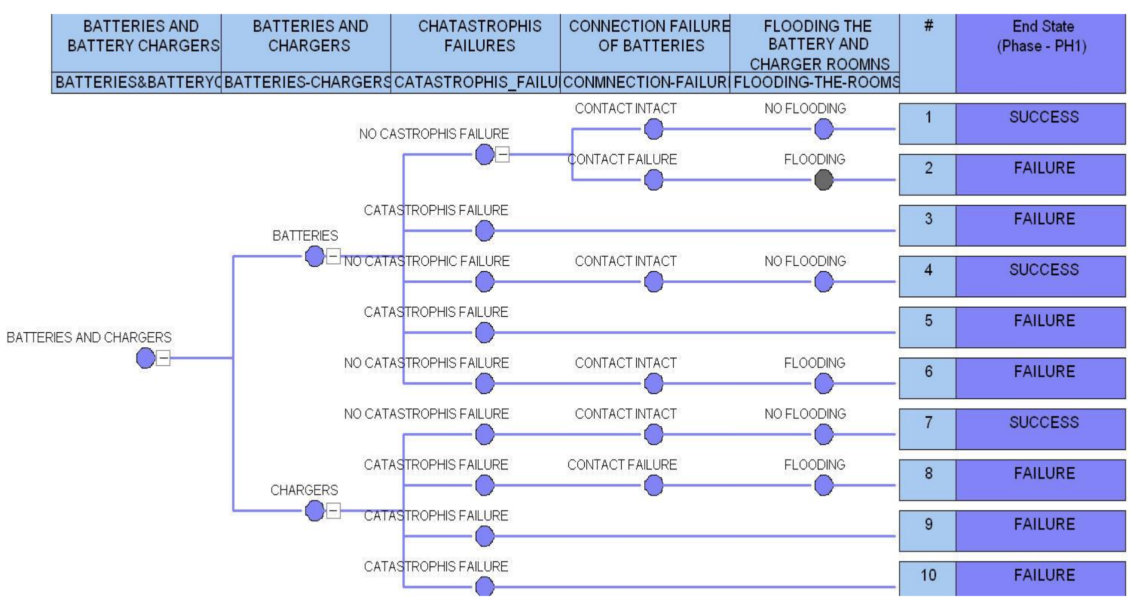

Figure 26. Event tree analysis for the station batteries and chargers

FOR “AND GATE”:

$F_T= F_{1*}F_{2*}F_{3*}......F_n$

FOR “OR GATE”:

$F_T = F_1+F_2+F_3+......F_n$

For the total batteries, the values of an event tree analysis are given below:

$F_{T(BATT)} = F_{BATTERIES} +F_{CONECTION} + F_{FLOODING}$ =0.0000064+ 0.0000064+0.0000087= **$2.15*10^{-5}$/hr.** For the battery room flooding, the value becomes 1.00 assuming no AC power with the decay heat present.

For the battery chargers, the values of an event tree analysis are given below**:**

$F_{T(CHARGER)} = F_{CHARGER} +F_{CONECTION} + F_{FLOODING}$ =0.00000607+ 0.0000106+0.0000409= $5.76x10^{-6}$/hr. These values are obtained from the IAEA document.

When taking into consideration presence of flooding, the value becomes1.0.

F_t (Battery + Charger) = $2.67x10^{-5}$x $5.76x10^{\mathbf{-6}}$ = $1.538x10^{\mathbf{-10}}$ without flooding. However, with flooding, the F_t (Batteries and the chargers become): $1x10^{-4}$ or even possibly 1.0, as it did at the Fukushima Daiichi plant.

This implies that the for the station battery and the chargers not to be functioning, that the probability is quite small.

6.12.3. Low-pressure cooling injection (LPCI)

For a discussion of this system and its drawing, see Section 2.1.4.7. of this manuscript. Figure 27 below shows the fault tree analysis of this system, and Figure 28 shows the event tree analysis of this system.

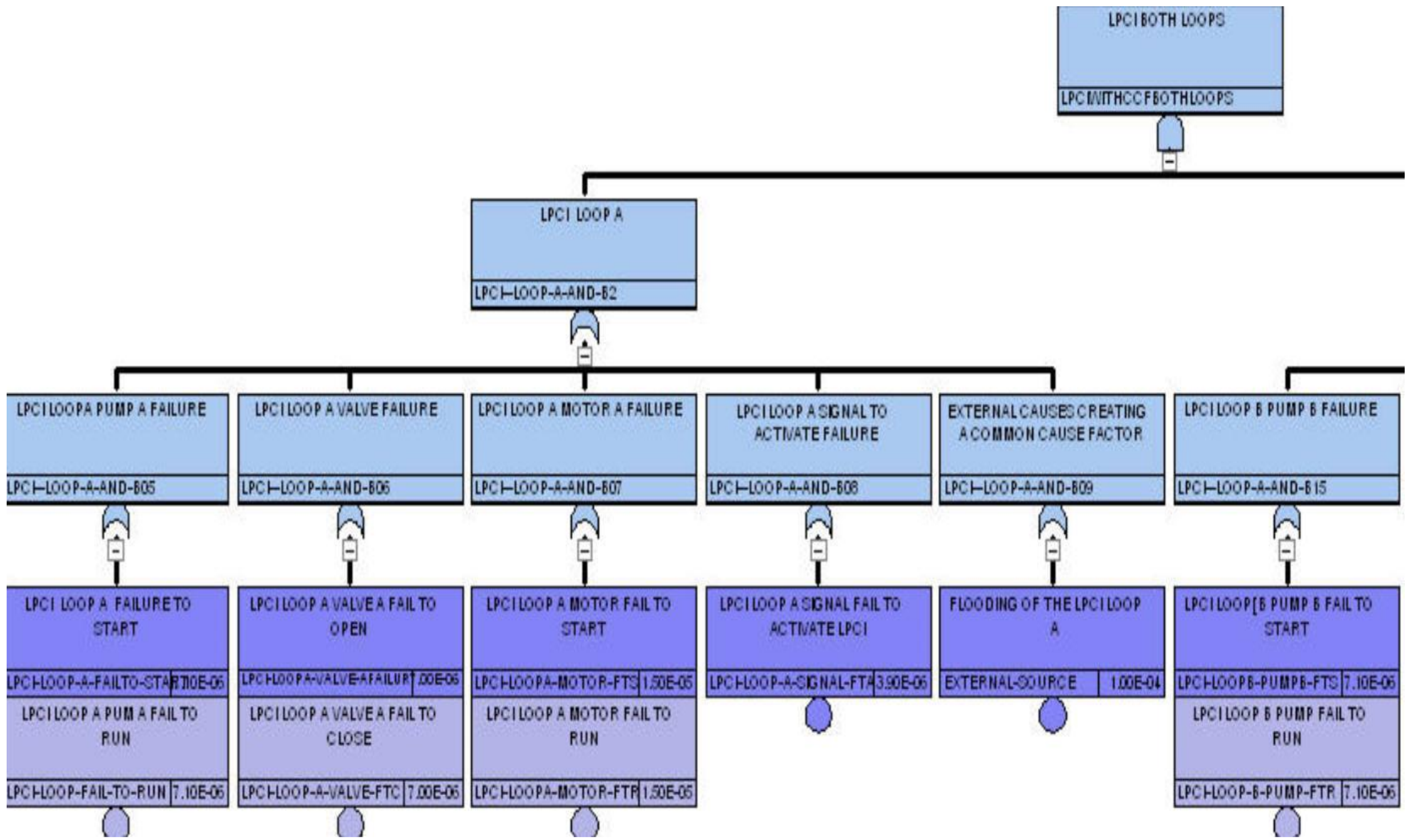

Figure 27. Fault tree analysis for the low-pressure cooling injection system

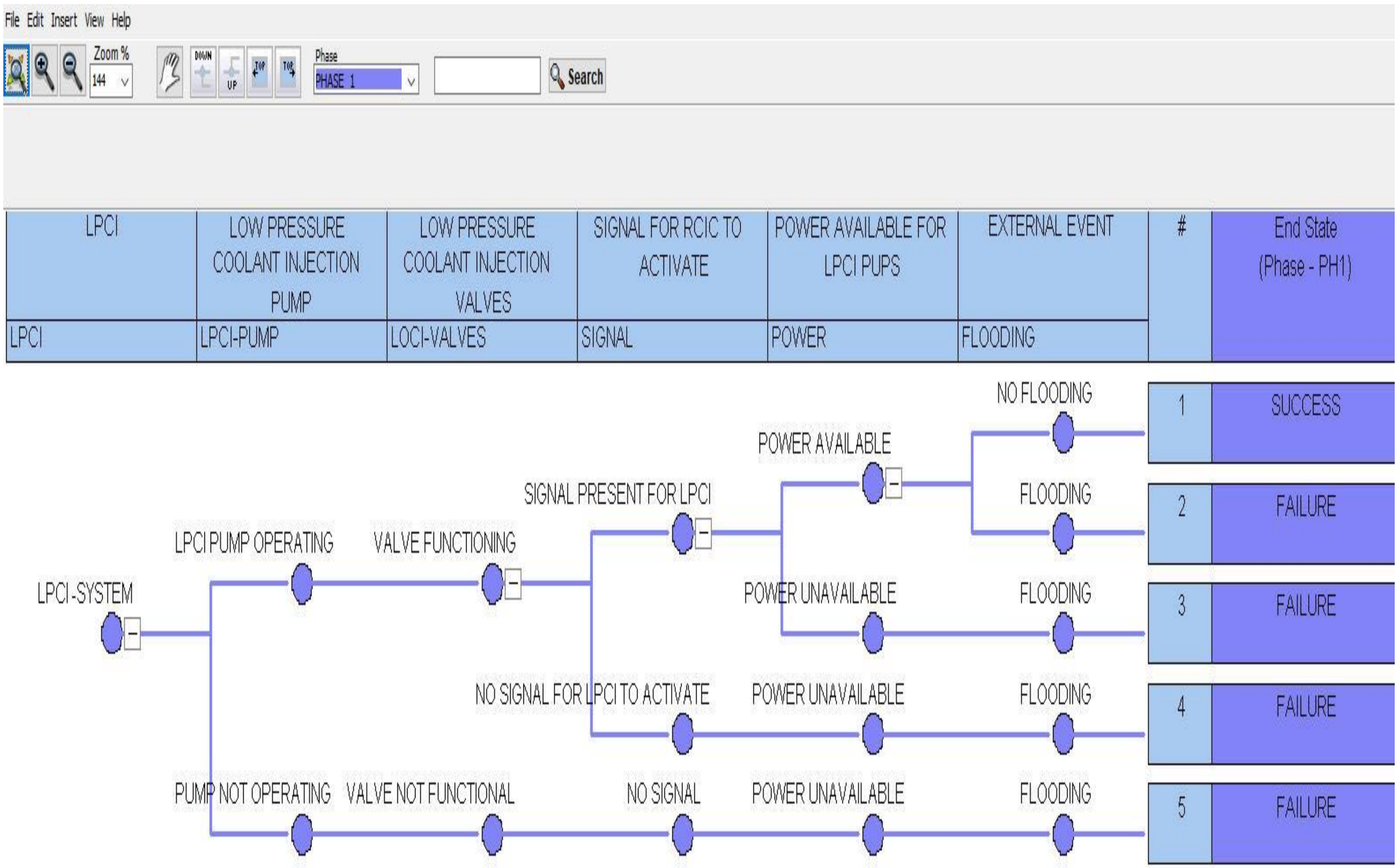

Figure 28. Event tree analysis for the low-pressure coolant injection system

FOR “AND GATE”: $F_T = F_{1*}F_{2*}F_{3*}\ldots\ldots F_n$

FOR “OR GATE”: $F_T = F_1+F_2+F_3+\ldots\ldots F_n$

$\mathbf{F_{T(\text{low-pressure coolant injection})} = F_{VALVE} + F_{SIGNAL} + F_{PUMP} + F_{POWER} = 0.0000015 + 0.0000039 + 0.0000071 + 0.0000014 = 1.39*10^{-5} = 1.39*10^{-5}/hr.}$

This implies that the low pressure cooling injection system failure does not pose an unreasonable risk to a LOCA condition. This core damage frequency is still acceptable per the NRC requirement. It must be noted that if all the other emergency core cooling systems fail and low pressure cooling injection system fails as well, the core damage frequency becomes equal to **1.0**, as happened at the Fukushima Daiichi nuclear power station

6.12.4. Low-pressure core spray (LPCS)

For a discussion of this system and its drawing, see Section 2.1.4.6. Figure 29 shows the fault tree analysis of this system and Figure 30 shows the event tree analysis of this system.

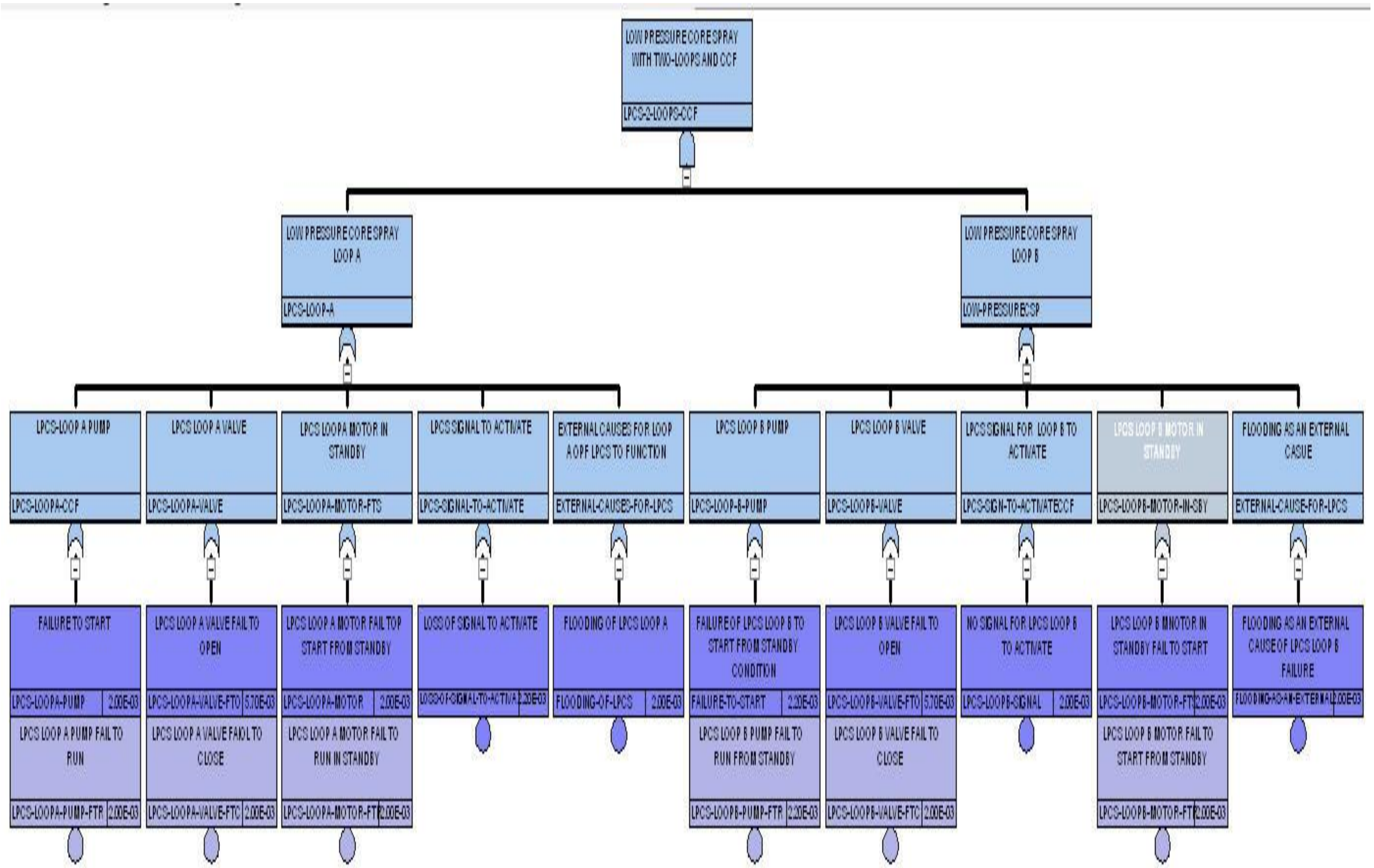

Figure 29. Fault tree analysis for low-pressure core spray system

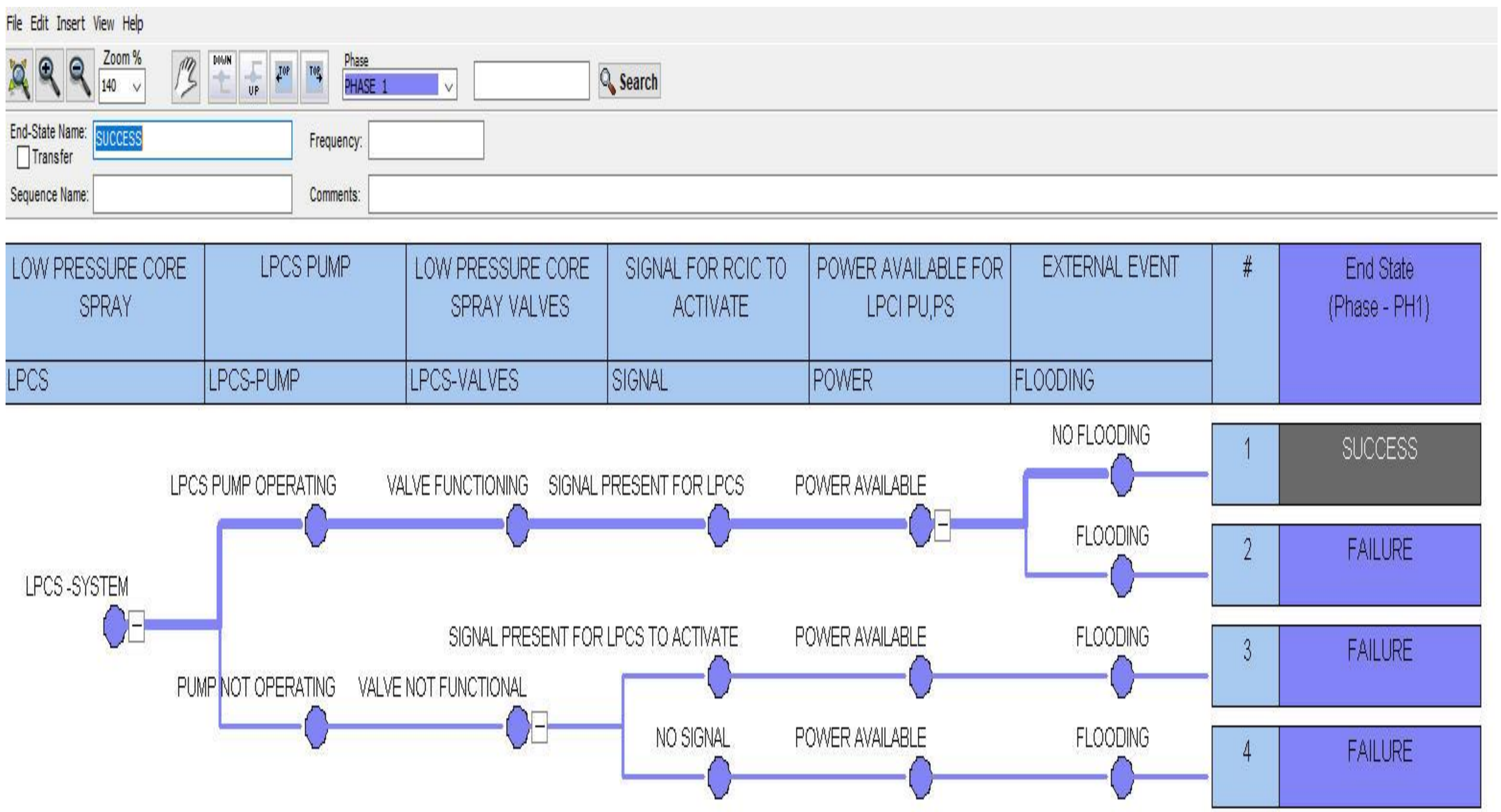

Figure 30. Event tree analysis for low-pressure core spray system

FOR “AND GATE”: $F_T= F_{1*}F_{2*}F_{3*}\ldots\ldots F_n$
FOR “OR GATE”: $F_T = F_1+F_2+F_3+\ldots\ldots F_n$
$\mathbf{F_{T(low\ pressure,\ core\ spray)} = F_{VALVE} + F_{SIGNAL} + F_{PUMP} + F_{POWER} = 0.0000015 + 0.0000039 + 0.0000071 + 0.0000014} := \mathbf{1.39*10^{-5}/hr.}$

This implies that the low pressure cooling spray system failure does not pose an unreasonable risk to a loss of coolant accident condition. This core damage frequency is still acceptable per the NRC requirement. It must be noted that if all the other emergency core cooling systems fail and low pressure cooling injection system fails as well, the core damage frequency becomes equal to **1.0**, as happened at the Fukushima Daiichi plant

6.12.5. High-pressure coolant injection system

For a discussion of this system and its drawing, see Section 2.1.4.2. Figure 31 shows an image of a high-pressure coolant injection turbine and its pump.

Figure 31. An image of a high-pressure coolant injection turbine [76]

Figure 32 illustrates the fault tree analysis of this system, and Figure 33 shows the event tree analysis of this system.

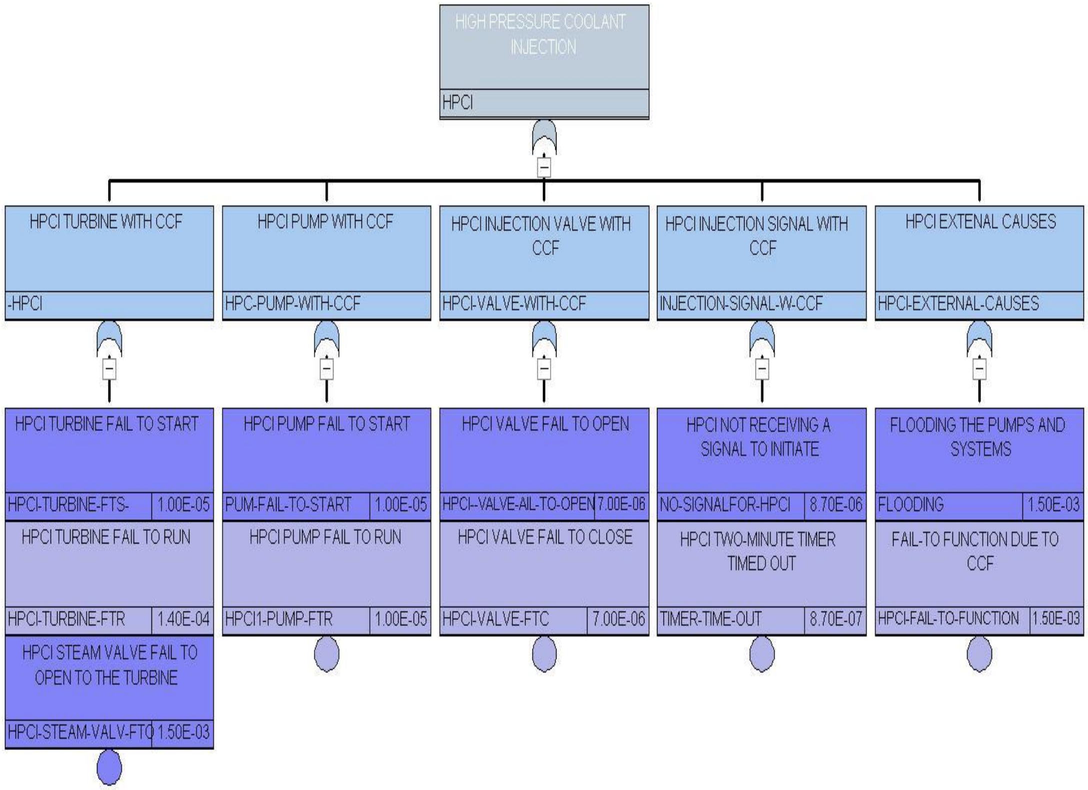

Figure 32. Fault tree analysis for the high pressure core injection system

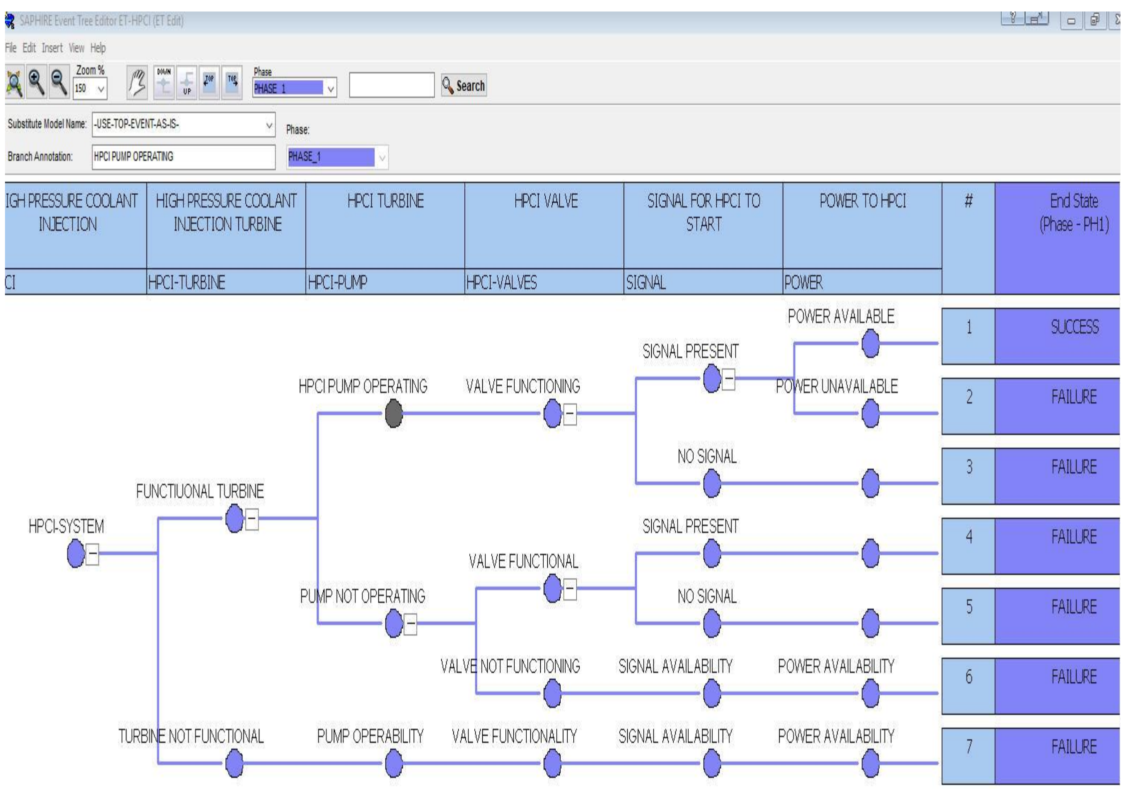

Figure 33. Event Tree Analysis for high-pressure coolant injection system.

FOR “AND GATE”: $F_T = F_1 * F_2 * F_3 * \ldots\ldots F_n$

FOR “OR GATE”: $F_T = F_1 + F_2 + F_3 + \ldots\ldots F_n$

$F_T = 0.00001+0.00001+0.000007+0.00000087+0.00000000007$

$=0.0000279=2.79*10^{-5}$

This implies that the core damage frequency due to high-pressure coolant injection failure is: **$2.79*10^{-5}$/HR.**

This implies that the high-pressure coolant injection turbine and pump, pose the greatest portion of the contribution to the core damage frequency, which is still acceptable per the NRC requirement.

6.12.6. Isolation condenser system

For a discussion of this system and its drawing, see Section 2.1.4.3. Figure 34 shows the fault tree analysis of this system, and Figure 35 shows the event tree analysis of this system.

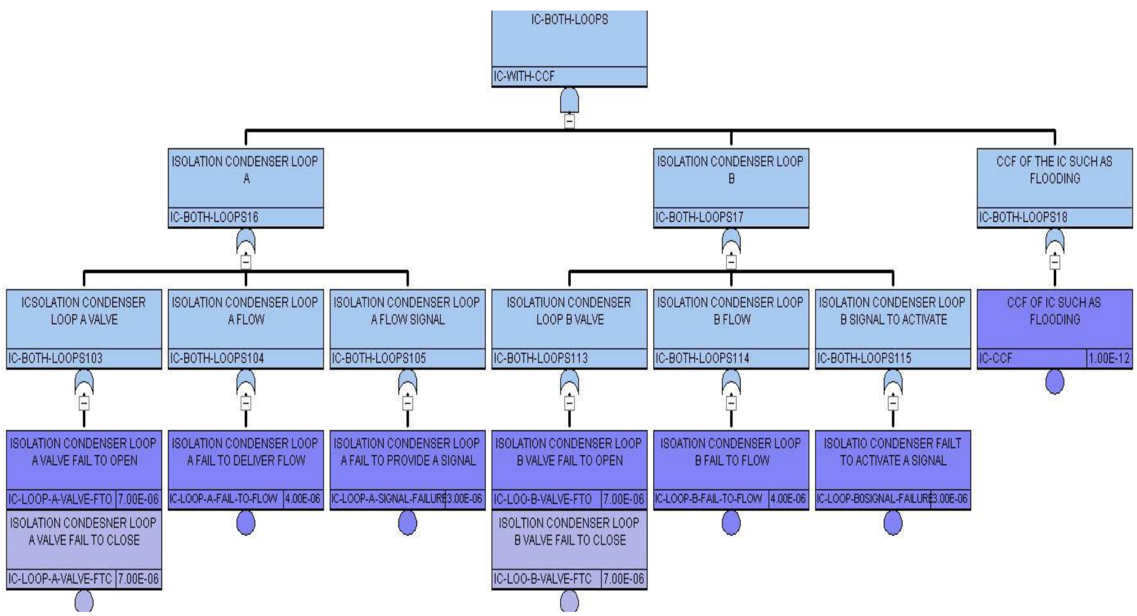

Figure 34. Fault tree analysis for isolation condenser system

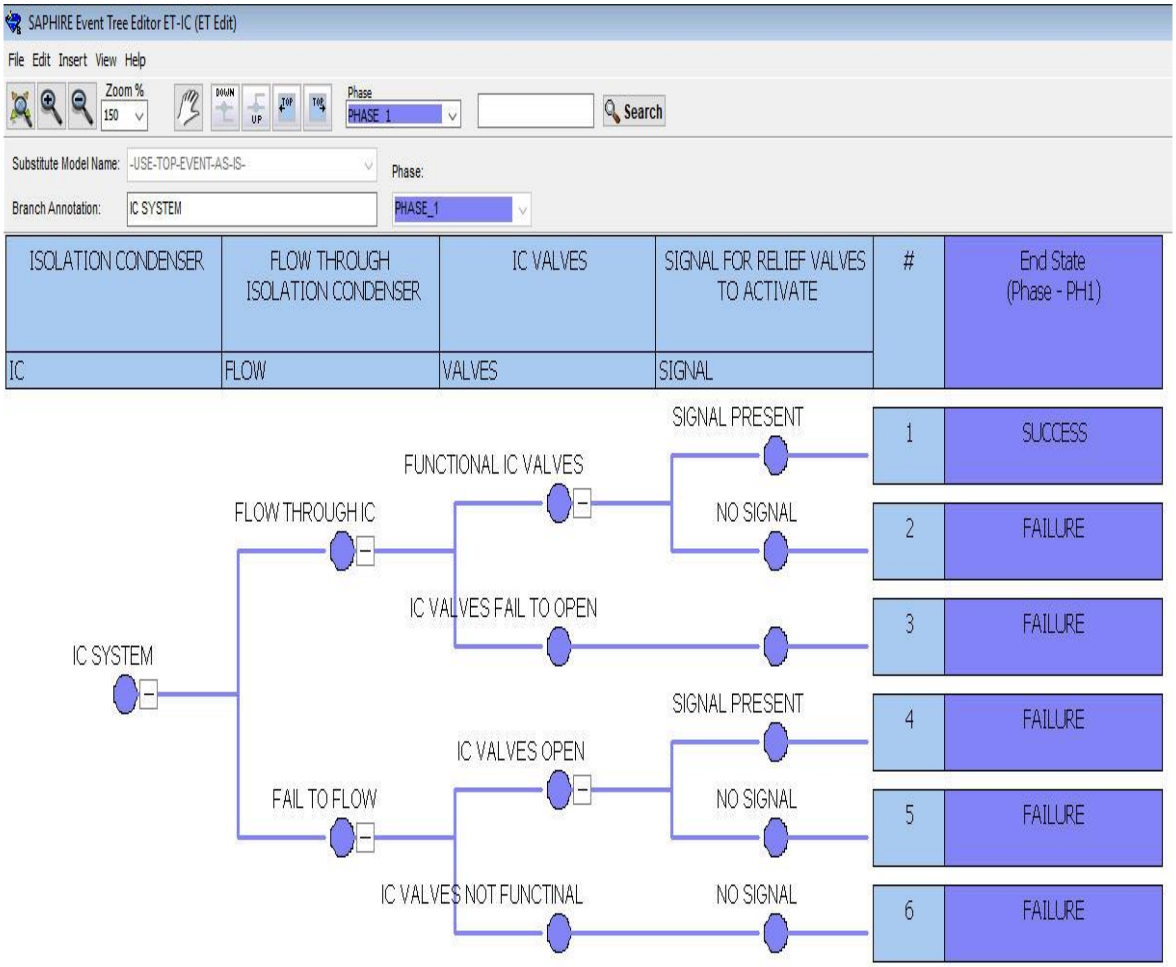

Figure 35. Event tree analysis for isolation condenser system

FOR “AND GATE”: $F_T = F_1 * F_2 * F_3 * \ldots\ldots F_n$

FOR “OR GATE”: $F_T = F_1 + F_2 + F_3 + \ldots\ldots F_n$

$F_{T(ic)}$=

$F_{FLOW} + F_{\mathbf{VALVE}} + F_{SIGNAL}$**+=**0.0000039+0.00000450+0.000000087+=0.00000849

=8.49x10^{-6}

This implies the core damage frequency due to isolation condenser failure is: 8.49x**10^{-6}/HR.**

This implies that the isolation condenser system failure does not pose an unreasonable risk to a LOCA condition. This core damage frequency is still acceptable per the NRC requirement. It must be noted that if all the other emergency core cooling systems fail and the isolation condenser system fails as well, the core damage frequency becomes equal to **1.0**

6.12.7. Auto depressurization system (ADS)

For a discussion of this system and its drawing, see Section 2.1.4.5. Figure 36 below shows the fault tree analysis of this system, and Figure 37 shows the event tree analysis of this system.

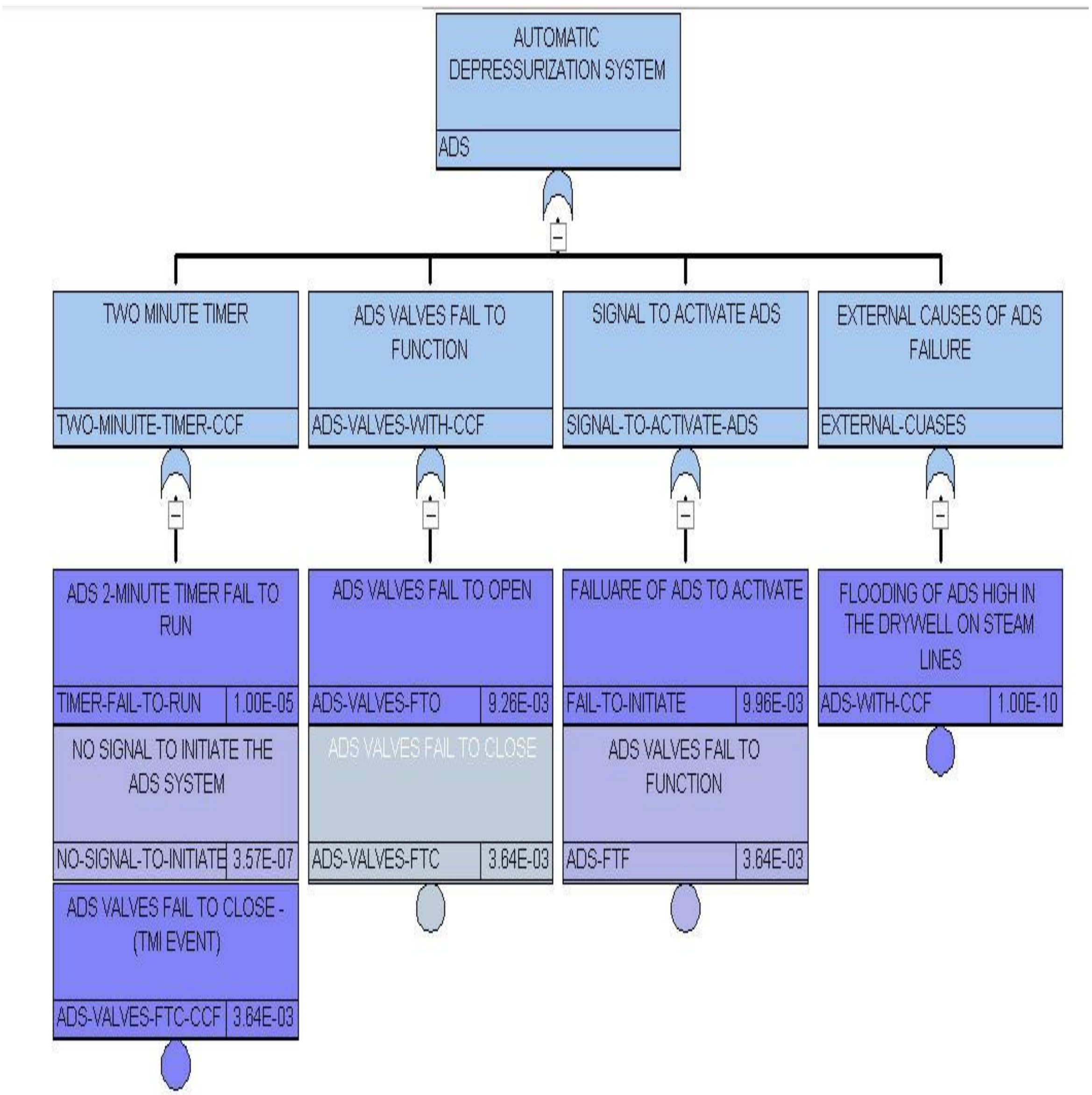

Figure 36. Fault tree analysis for the automatic depressurization system

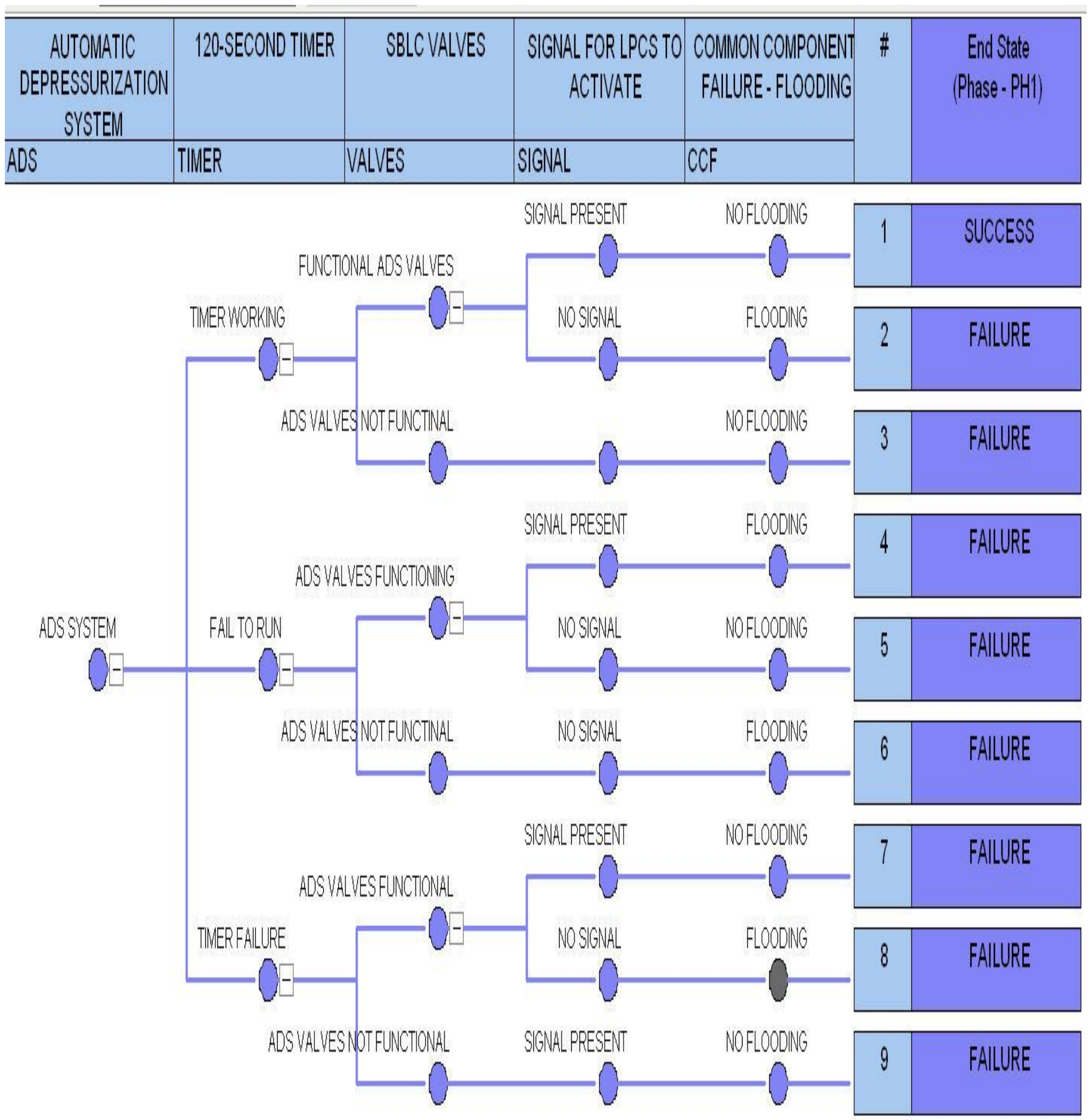

Figure 37..Event tree analysis for the automatic depressurization system

FOR “AND GATE”: $F_T = F_{1*}F_{2*}F_{3*}......F_n$
FOR “OR GATE”: $F_T = F_1+F_2+F_3+......F_n$
$F_{T(ADS)}$ = 0.0001+0.00001+0.00007 **=0.00018=1.8x10^{-4}**

This implies that the core damage frequency due to automatic depressurization system failure is: **1.8*10^{-4}/HR.**

This implies that for the ADS the failure of the valves poses the greatest portion of the contribution to the core damage frequency. However, if all the ADS valves become non-functional, and there is a signal available to open the valves, the core damage frequency becomes equal to 1.0, as happened in the Fukushima Daiichi event.

6.12.8. Safety relief valves (SRV) [18]

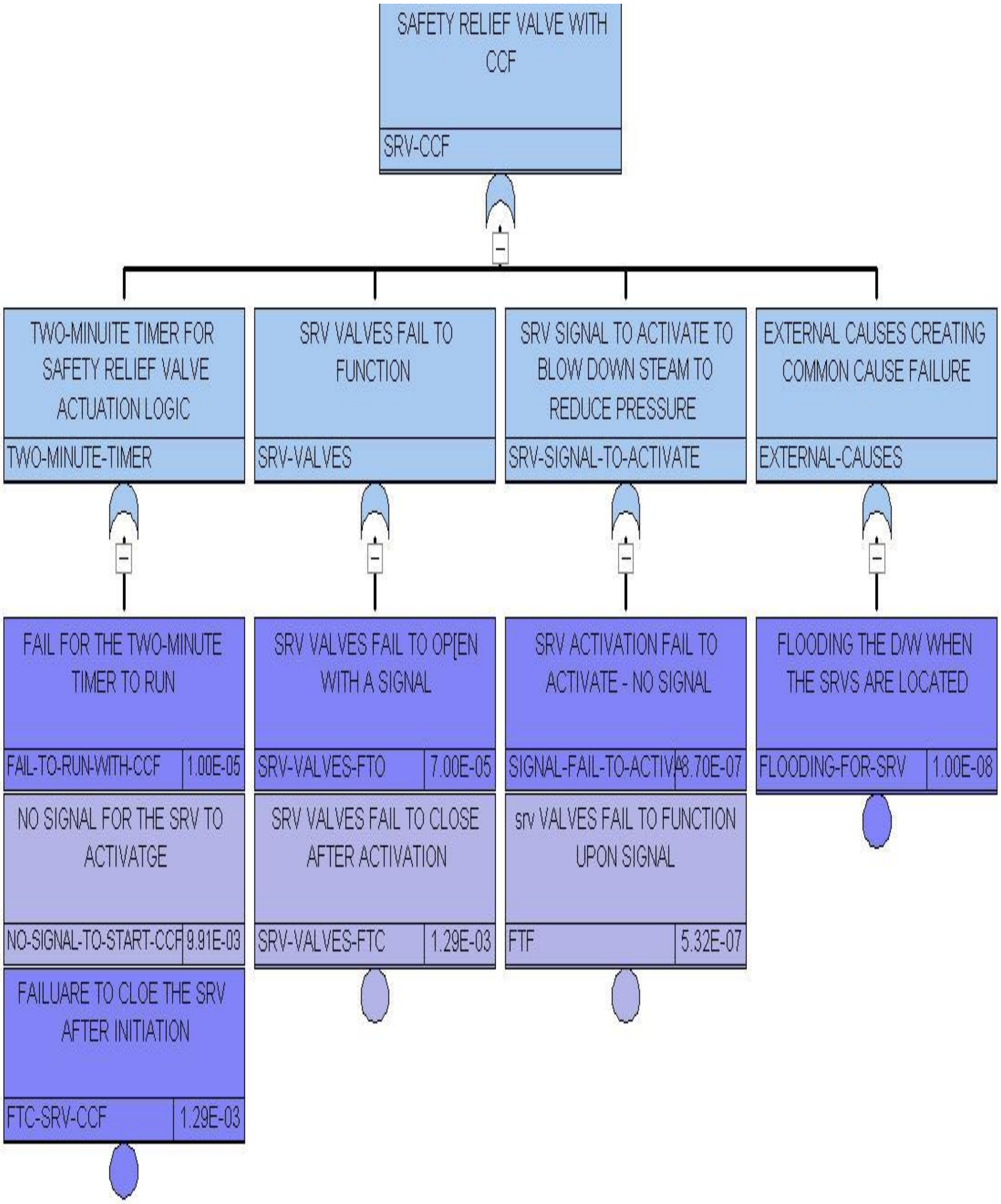

Figure 38. Fault tree analysis for safety relief valve

[18] CCF is an abbreviation for common cause failure. Flooding is one element of a common cause failure, which can render multiple systems inoperable.

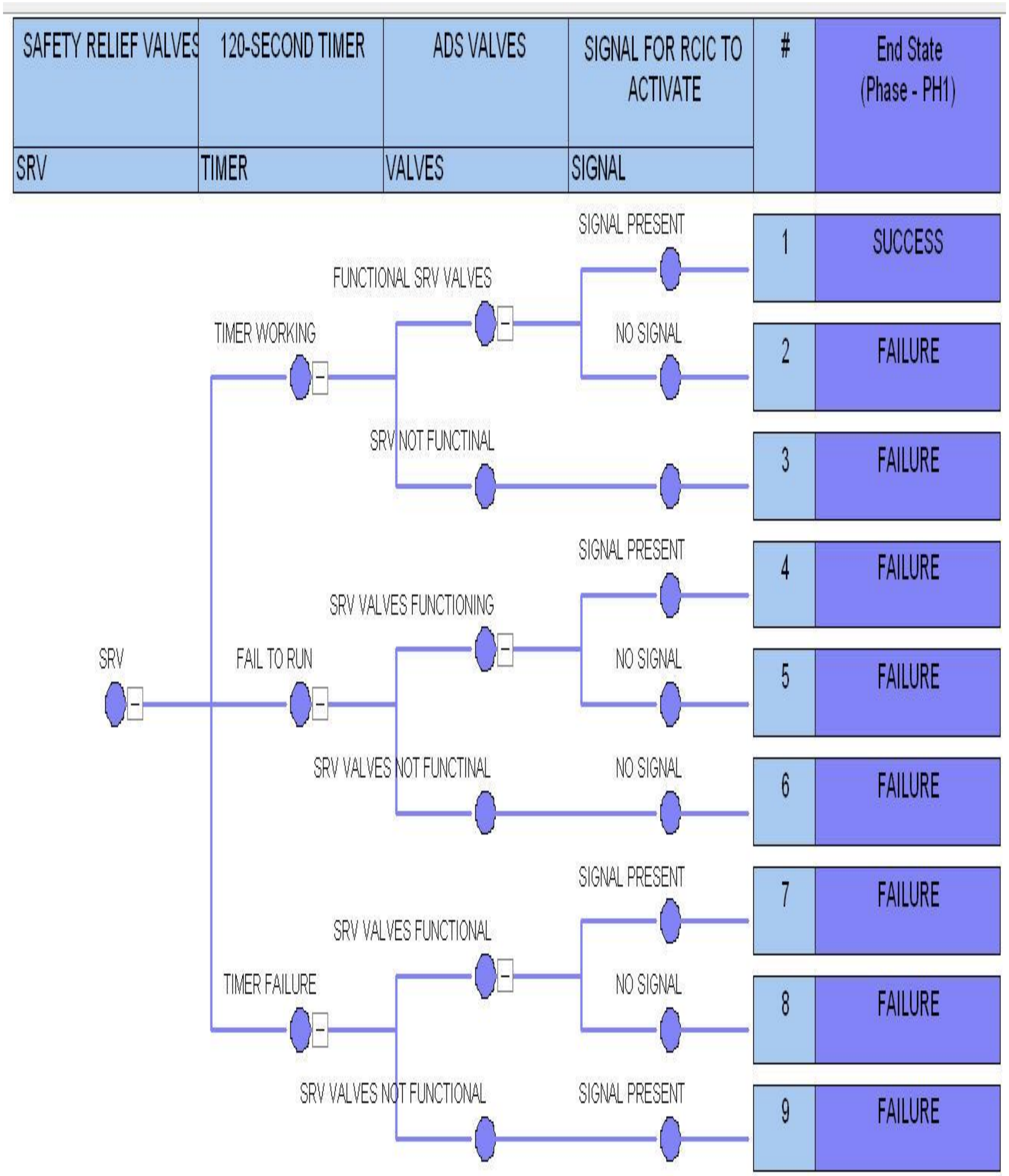

Figure 39. Event tree analysis for safety relief valves

Figures 38 and 39 show the fault tree and event trees of safety relief valves, and Figure 40 [77]shows a safety relief valve inside the containment in a tight *spot.*

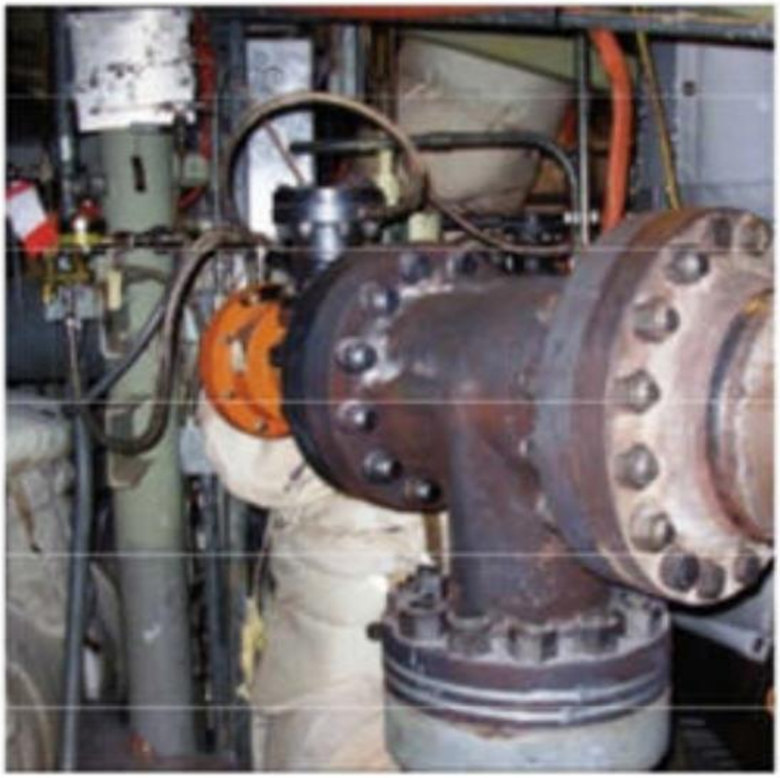

Figure 40. A safety relief valve Reactor core isolation cooling system

6.12.9. Reactor Core Isolation Cooling system (RCIC)

For a discussion of this system and its drawing, see Section 2.1.4.4. Figure 41 provides the fault tree analysis of the reactor core isolation cooling, whereas, Figure 42 provides the event tree analysis of the system..

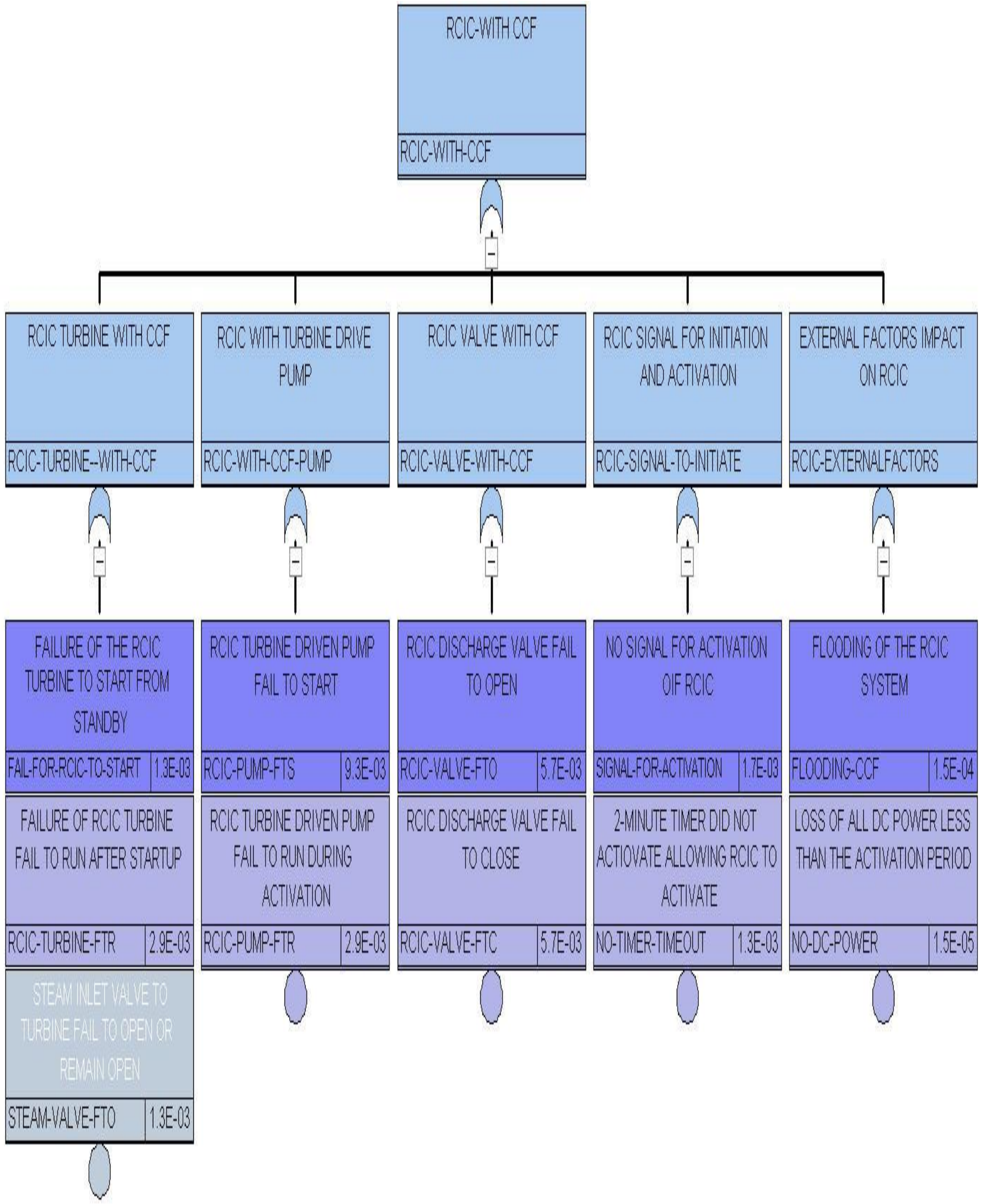

Figure 41. Fault tree analysis for reactor core isolation cooling system

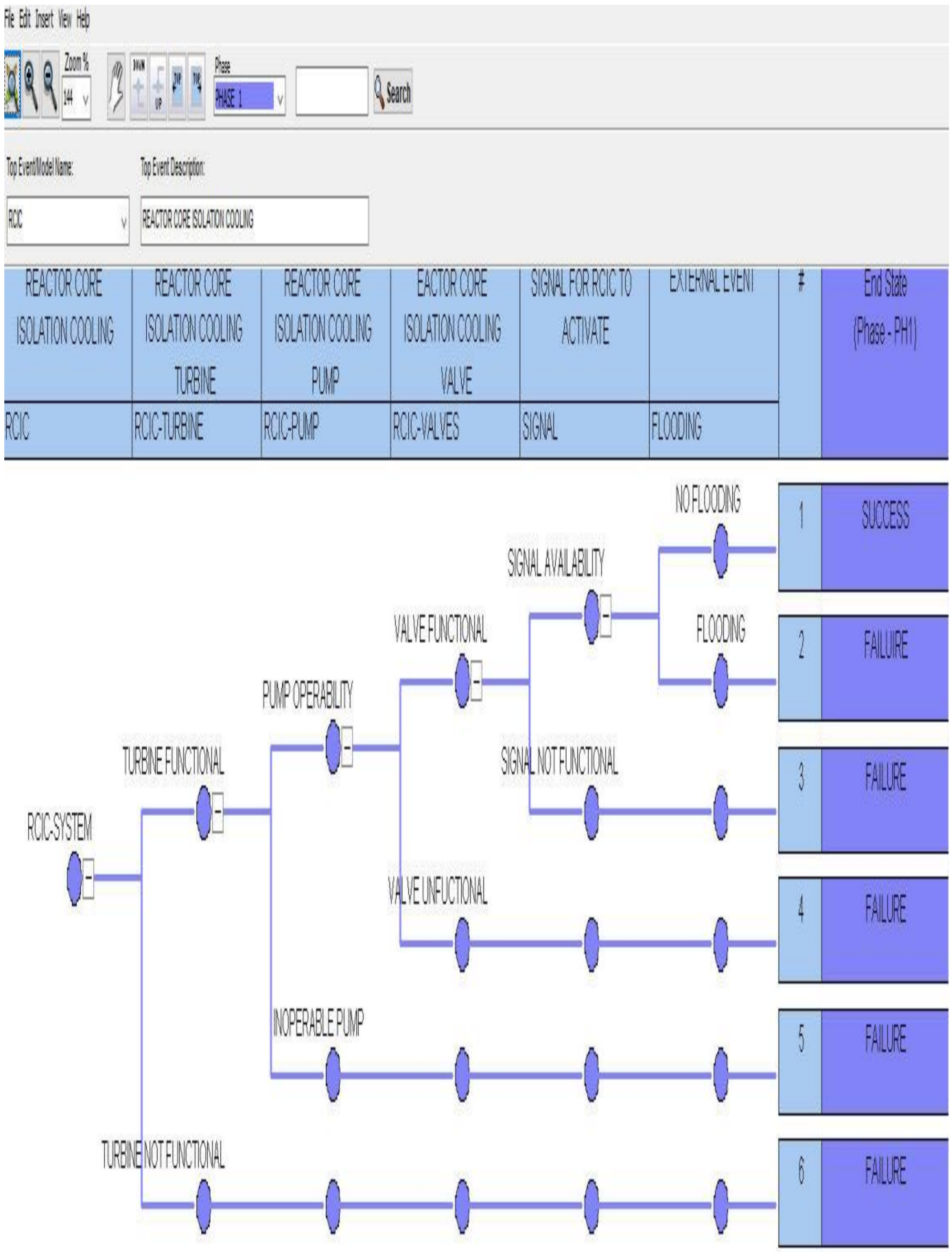

Figure 42. Event tree analysis for reactor core isolation cooling

FOR "AND GATE": $F_T = F_1 * F_2 * F_3 * \ldots\ldots F_n$
FOR "OR GATE": $F_T = F_1 + F_2 + F_3 + \ldots\ldots F_n$
$\mathbf{F_{T(RCIC)}} =0.00001+0.00001+0.000007+0.00000087+0.00007=0.0000279\mathbf{=9*10^{-5}}$

This implies that the core damage frequency due to reactor core isolation cooling system failure is: **$9*10^{-5}$/HR** when power is available for one day. Core damage frequency becomes 1, if the reactor core isolation cooling system has no power for longer than a day and no other emergency core cooling system is available due to lack of AC power.

This implies that the reactor core isolation cooling turbine, pump, and DC power pose the greatest portion of the contribution to the core damage frequency, which is still acceptable per the NRC requirement. However, if DC power becomes unavailable and there is a signal unavailable to run the reactor core cooling system, the core damage frequency becomes equal to 1.0, as happened in the Fukushima Daiichi accident.

6.12.10. Standby liquid control system

For a discussion of this system and its drawing, see Section 2.1.4.8. Figure 43 provides the fault tree analysis of the standby liquid control system, whereas, Figure 44 provides the event tree.analysis of the system.

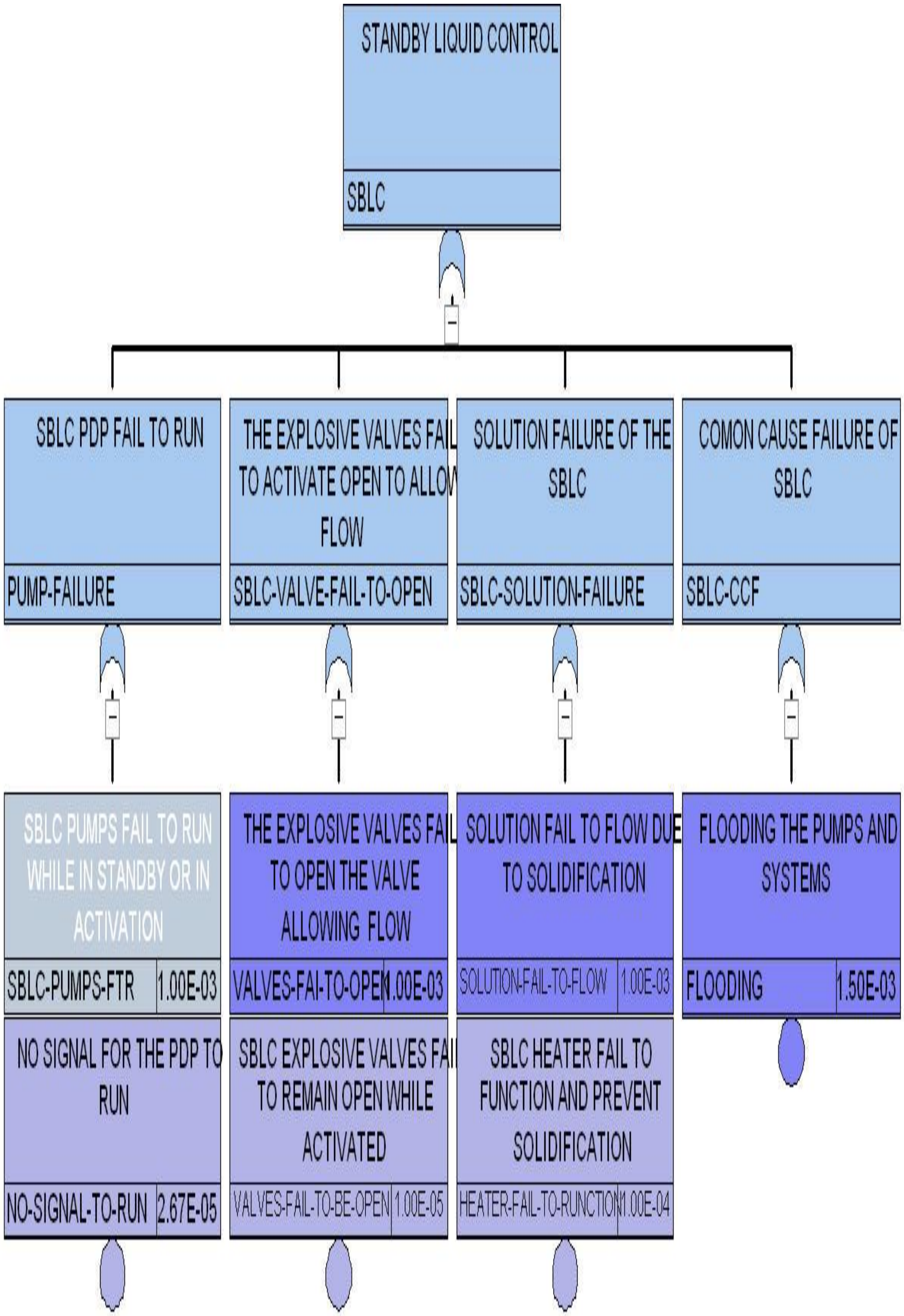

Figure 43. Fault tree analysis for standby liquid control system

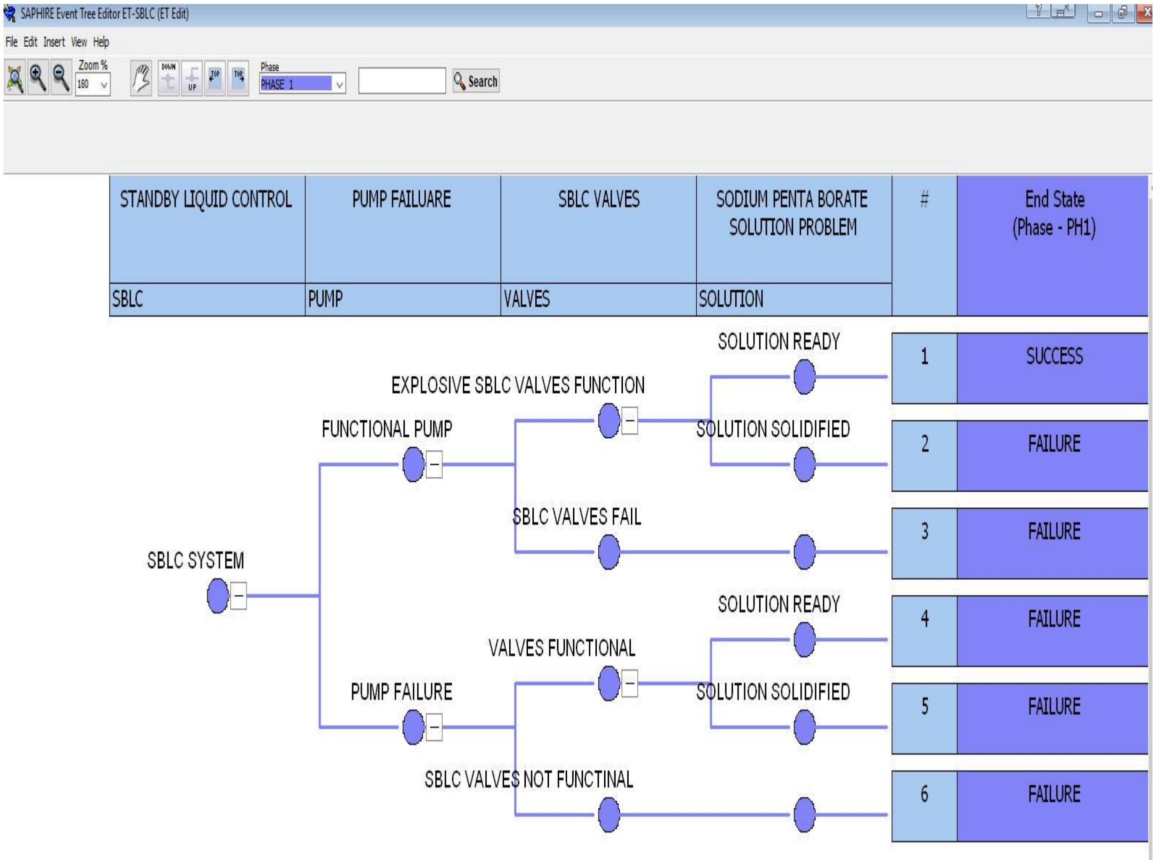

Figure 44. Event tree analysis for standby liquid control system

FOR “AND GATE”: $F_T = F_1 * F_2 * F_3 * \ldots\ldots F_n$

FOR “OR GATE”: $F_T = F_1 + F_2 + F_3 + \ldots\ldots F_n$

$F_{T(SBLC)} = 0.0001+0.00001+0.000007+0.0000016=0.00019=1.19*10^{-4}$

This implies that the core damage frequency due to all components of the standby liquid control failing is: **$1.19*10^{-4}$/hr.**

This implies that for all control rods, and the standby liquid control to fail, the two valves pose the greatest portion of the contribution to the core damage frequency. However, if one valve fails, there is one other valve available, the failure frequency will reduce 1/4 of that value ($1/2^2$), and that would become **$2.9x10^{-5}$**, which is smaller than **$1.0x10^{-5}$**, which is acceptable.

CHAPTER SEVEN - RESULTS OF THE RESEARCH, AND SUMMARY

7.1. Comparative analysis.

This part compares the results of analysis performed by the researcher and a utility that independently performed their own probabilistic risk assessment analysis. This utility, like most utilities use the computer aided fault tree analysis (CAFTA) developed by EPRI. This is a software similar to SAPHIRE, except one major difference, that being that it does not normally perform the event tree analysis. It calculates the core damage frequency independent of performing event tree analysis relying only on fault tree analysis. Because it does not rely on event tree analysis, it does not use individual components. The resulting core damage frequency is mostly a guide for determining the impact of making a change in a plant configuration on the core damage frequency. Whenever a utility that uses CAFTA wishes to remove a system out of service, they perform a probabilistic risk assessment using this software. They have developed a method that it would take little effort and time to get their results. Shift technical advisors on duty have been instructed to perform that task to perform this function. Another notable difference is that the utility's failure frequency data comes from the NRC's NUREG/CR 6928 whereas this research used the IAEA document that is referenced numerous times in this manuscript. Neither of these two software programs provides information as to how these software programs perform the calculation. One software is developed by the Idaho National Laboratory (SAPHIRE), and the other by EPRI (CAFTA). The method of calculation is proprietary information for these providers and the calculation methods are not released to a third party. The process requires as inputs the elements of the fault tree analysis and the values of core damage frequencies. It is the function of the software programs to provide the values of the core damage frequencies.

For a better understanding of the content of the NUREG /CR 6928, Table 7 provides failure modes and failure frequency values for 18 different components, some with more than one failure mode. The values of failure frequencies are given per unit time, some per day and others per hour. There is a difference between failure to operate (FTO) and failure to run (FTR). Failure to run assumes the unit starts but fails to continue to operate.

Table 7. Selective values of failure frequency from NUREG 6928

COMPONENT	FAILURE MODE[19]	FAILURE FREQUENCY	TIME ***
AIR OPERATED VALVE	FTC	3.0 E-6	H
AIR OPERATED VALVE	SIGNAL	1.11 E-7	D
BATTERY	FAIL TO OPERATE	1.86 E-7	H
BATTERY CHARGER	FAIL TO OPERATE	5.08 E-6	H
CIRCUIT BREAKER	FTC/FTFUNCTION	2.55 E-3	D
CHECK VALVE	FTO	1.30 E-5	D
CHECK VALVE	FTC	1.04 E-4	D
EMER. DIESEL GENERATOR	FTC /FTR	2.90 E-3	D
EDG	FTR (AFTER 1 HR)	4.53 E-3	D
EXPLOSIVE VALVE (SBLC)	FTO	1.07 E-3	D
MOTOR DRIVEN PUMP	FTR	4.54 E-6	H
MOTOR DRIVEN PUMP	FTR	2.23 E-3	D
MOV	FTC	3.0 E-6	H
POSITIVE DISPLACE. PUMP	FTR	8.32 E-6	D
PDP (SBLC)	FTS	3.3 E-3	D
RELIEF VALVE	FTC	1.09 E-3	H
RELIEF VALVE	FTO	4.63 E-3	D
SAFETY RELIEF VALVE	FTC	7.95 E-4	D
SRV	FTO	7.71 E-7	D
SRV	FT CLOSE	1.0 E-1*	D
WATER LEVEL SENSOR	FTC	8.15 E-4	D
SAFETY VALVE	FTC	6.76 E-5	D
SAFETY VALVE	FTO	2.47 E-3	D
TURBINE DRIVN PUMP (TDP)	FTR	2.64 E-3	H
TDP	FTS	2.22 E-2**	D
TANKS - LARGE	LEAKAGE	2.23 E-9	H

[19] FTC: Fail to close for valves or circuit breakers
FTS: Fail to start for pumps or motors
FTO: Fail to open for circuit breakers or for valves
FTO: Fail to operate for electrical component
FTL: Fail to load for generators
FT Function: fail to function for an electrical or a mechanical component
H: Time on hourly basis
D: Time on daily basis
The convention here is:E-x = 10^{-X}, * The lowest value of core damage frequency for the Safety Relief Valve fail to close, which is 1xE-1, whereas, for the same component, Fail to Open has a value of 7.71 E-7. The remnant of the Three Mile Island event has carried its presence here. The Safety Valve will open but the probability is low for its closure.
** The second lowest value of core damage frequencyfor the Positive Displacement Pump, is the pump for the SBLC.
*** The unit of component failure frequency in this table for some parameters is failure per day. When the values are given per hour, it needs to be converted to failure per day.

$1x10^{-5}$ or one in 100,000 operating reactor years. If a failure results in a core damage frequency equal to $1x10^{-4}$, it implies that the failure of a single component could result in core damage frequency of one core so that it becomes degraded in 10,000 reactor years. Within the design space of a nuclear power plant, such a failure frequency is unacceptable and the licensees must perform analyses to verify their reactor is safe, or take necessary steps to implement changes to the plant systems and components to ascertain plant safety. In January of 2018, the NRC issued the Reg Guide 1.174 to the licensees establishing acceptable values for core damage frequency [78]. This Reg Guide generated guidelines for the licensees to follow in case the probabilistic risk assessment resulted in an unacceptable value.

7.2. Combination of failures

This section discusses the different combination of failures, from a single component to multiple systems.

7.2.1. Single failure of a single component

The Impact of a single failure of a component within the emergency core cooling systems was determined. The results of the analysis verified that a failure of a single component within any of the emergency core cooling systems did not and would not alter the core damage frequency. This is a design basis requirement of any system in a nuclear power plant. Depending on the component, it can reduce the margin to a degraded core frequency without exceeding the threshold of allowable core damage frequency.

7.2.2. Failure of a single system, including multiple components in that system

Similarly, the failure of a single system within the emergency core cooling systems does not and would not result in a reduction of core damage frequency, per the design requirements. This includes multiple components within a system, without the presence of a common cause failure. As an example, we could hypothesize a failure of the turbine-driven pump of the high-pressure core injection system (HPCI), its valve (steam inlet and discharge to the reactor pressure vessel), HPCI turbine, and other components, thus rendering the high-pressure cooling injection system inoperable. If this system fails, there are backup systems to reduce reactor pressure, such as the safety relief valves, automatic depressurization system, and reactor core isolation cooling system to reduce reactor pressure until the low-pressure cooling systems can inject or spray coolant into the reactor pressure vessel. The design basis of these systems mandates that a single failure of a system will not result in core damage. This concept is referred to a single failure criterion.

7.2.3. Multiple component failures in multiple systems.

When more than one system fails, the scenario is outside the principle of single failure criterion. For this combination, failures of more than one component largely depends on the component itself. For example, if the turbine-driven pump of the high-pressure cooling injection system fails, rendering the high-pressure coolant injection system inoperable, a lower core damage frequency depends on how many safety relief valves fail concurrently. In some BWRs, such as the Fukushima Daiichi nuclear power station, there are 7 relief valves, and 5 safety valves. In addition to automatic actuation, these valves can be manually opened to reduce the reactor pressure via the switches placed in the main control room panels.

Since there are more than one high-pressure coolant injection systems (HPCI, RCIC, and HPCS) and more than one low-pressure coolant injection systems (LPCI, LPCS, SDC) it is permissible to lose one high-pressure coolant injection system and one low-pressure coolant injection system without causing any core damage. However, the failure of more than one high-pressure or more than one low-pressure coolant injection system may result in core damage.

7.2.4. Common cause failure

This is a reference to a condition that a single causal factor common to multiple safety and non safety-related systems can cause multiple systems to fail to function. Examples of common cause failure are: flooding, loss of AC power, loss of DC power, loss of all electrical and pneumatic air, human error, or other factors. Under such scenarios, it is highly likely that a core damage frequency of 1.0 or a 100% probability will ensue. This condition occurred in the Fukushima Daiichi, Chernobyl, and Three Mile Island accidents. At the Fukushima Daiichi nuclear power station, the absence of all core cooling systems, caused the reactor water level to drop below the bottom of the active fuel assemblies, thus resulting in a degraded core condition in about four hours. Three of the six reactors, which had just shut down from being at full power before the scram suffered massive core damage. One reactor had the benefit of its functional reactor core isolation cooling provided coolant to the reactor and avoided a degraded core condition. The remaining two reactors had been in cold shutdown and one of these two reactors had no fuel assemblies in the reactor pressure vessel, just prior to the accident.

7.3. Results of the research

The results of the research identified several key factors. As stated in Section 7.2, a failure of a single component will not result in a degraded core condition. It is quite useful to assimilate the results of the analyses in a single table. For this purpose, Table 10 is constructed with its four columns and 12 rows. The first column lists the systems that were analyzed. For the batteries, two rows are used, one for the station batteries and the other, the battery chargers. The second column identifies the input for each system for the analysis, and assumptions are reflected in the third column. The third column provides the assumptions for the systems evaluated in the results section. The fourth column provides the results of the analysis for each system. In Table 8 the results column (the column four on the right) refer to the values of core damage frequencies for various systems that are listed in column 1 on the left.

Table 8. Results of the analyses for various systems with inputs and assumptions

SYSTEMS	INPUT	ASSUMPTIONS	RESULTS
Diesel generators	• Fuel oil • Circuit breaker • Water heater • Diesels are functioning	• Fuel is available • Circuit breakers are intact • Connections are intact • No flooding	8.09E-02 (1 diesel 6.5E-5 (2 diesels)
Batteries	• Functioning batteries • Connections	• Battery room is dry • Connections are intact • Batteries are functional	2.15E-5
Battery charger	• Power from AC • Connections to batteries	• Battery room is dry • AC power is available • Intact connections	5.76E-5
Low-pressure-core injection	• Pumps functioning • Valves functioning • Signal • Motor functioning • AC power available • Piping	• AC power is available • Signal present • Low reactor pressure • Pumps &valves functioning • Piping intact • Signal present	1.39E-5
Low-pressure core spray	• Pumps functioning • Valves functioning • Signal present • Motor functioning • AC power available • Piping intact	• AC power is available • Signal present • Low reactor pressure • Pumps &valves functioning • Piping intact • Signal present	1.39E-5
High-pressure core injection	• Pumps and valves • Turbine • DC Power • Signal • Piping	• The turbine is functional • Pumps and valves are functional • The signal is present • DC power available • Steam exhaust available	2.79E-5
Isolation condenser	• Tanks • Valves • Piping	• Valves functioning • Lines intact • Tanks filled	8.49E-5
Automatic Depressurization system	• Valves • Signal • Timer	• Signal present • Valves functioning • The timer is functioning	1.9E-4
Safety relief valves	• Valves • Signal • Timer	• Signal present • Valves functioning • The timer is functioning	1.8E-5
Reactor core isolation cooling	• Pumps and valves • Turbine • DC Power • Signal	• Turbine functioning • Pump functioning • Valves functioning	9.0E-5
Standby liquid control	• Solution in the tank • Valves • Pump • Piping and injection line	• Tanks contain the solution • Solution in liquid phase • Pump functioning • Valves functioning • Pipes intact	1.33E-4 (1 valve) 2.9E-5 (2 valves)

This research identified conditions in which a common cause failure can result in a degraded core condition. Under such conditions, the research proposed recommendations to reduce the core damage frequency. Proposing recommendations is insufficient to determine their effectiveness. The proof of the validity of the recommendations was based on determining the value of the core damage frequency assuming implementation of one or more of the recommendations. For this purpose, new sets of fault tree analysis and event tree analysis were performed after assuming implementation of recommendation(s). The result of the analysis indicated that after implementing a recommendation on a typical emergency core cooling system, the value of core damage frequency improved. Two of the recommendations are presented below and they are based on performing the event tree analysis for one system, the reactor core isolation cooling system.

- Recommendation 1: Adding DC batteries, thereby extending the cooling capability of the reactor core isolation cooling system to one week. This duration is sufficient to handle the decay heat.
- Recommendation 2: Add an air operated valve in parallel with the existing motor-operated valves for critical valves on the emergency core cooling systems. The air operated valve can use air cylinders, thus making it independent of an electrical power source.

In the reactor core isolating cooling or high pressure core cooling injection systems, the flood will not render the system inoperable because the location of the turbine-driven pump and the valves are sufficiently high in the reactor or turbine buildings that they will not be submerged. Three images of the event tree analysis for the reactor core isolation cooling system are presented below (Figures 45-47).

Figure 45: Not implementing any recommendation.: Figure 46: is for Implementing installation of the additional batteries. Figure 47: is for implementing installation of the batteries and an air-operated valve in parallel with the motor operated valve. In these comparisons, one can note that as the impact of a negative factor is removed, the event tree analysis changes a result from a failure to success.

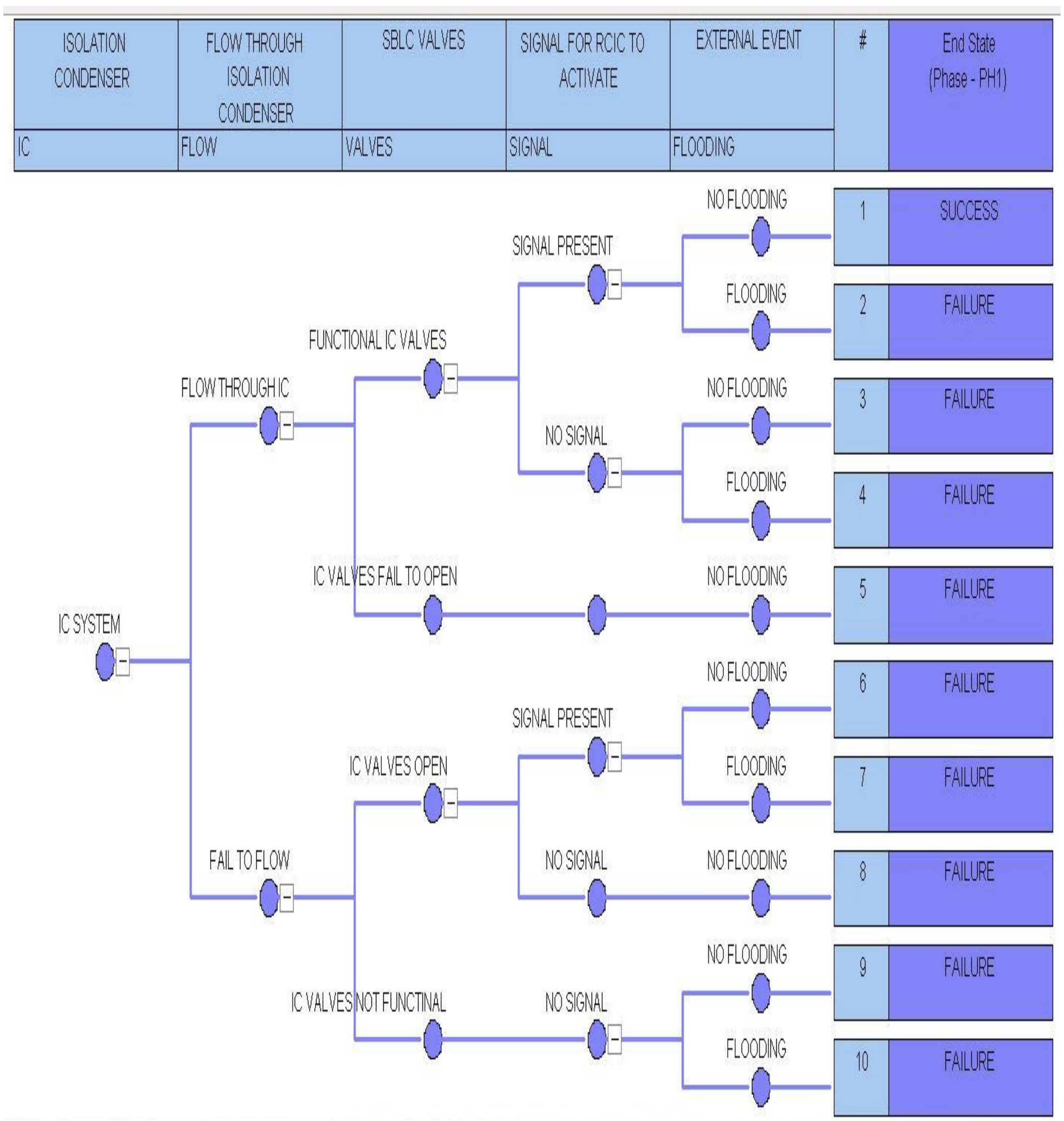

Figure 45. Event tree analysis for the RCIC with no recommendation

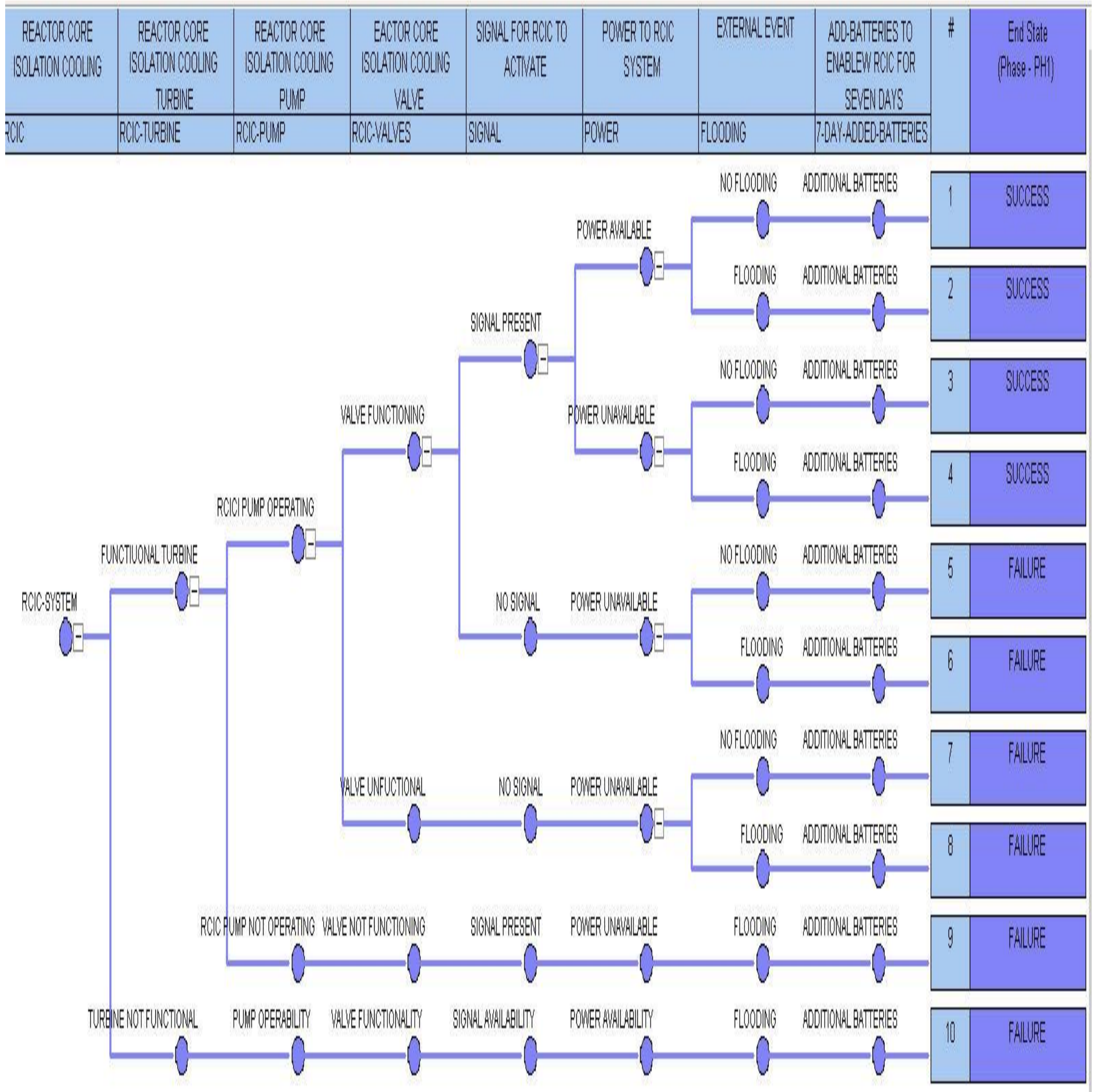

Figure 46. Event tree analysis for the RCIC system with extra batteries

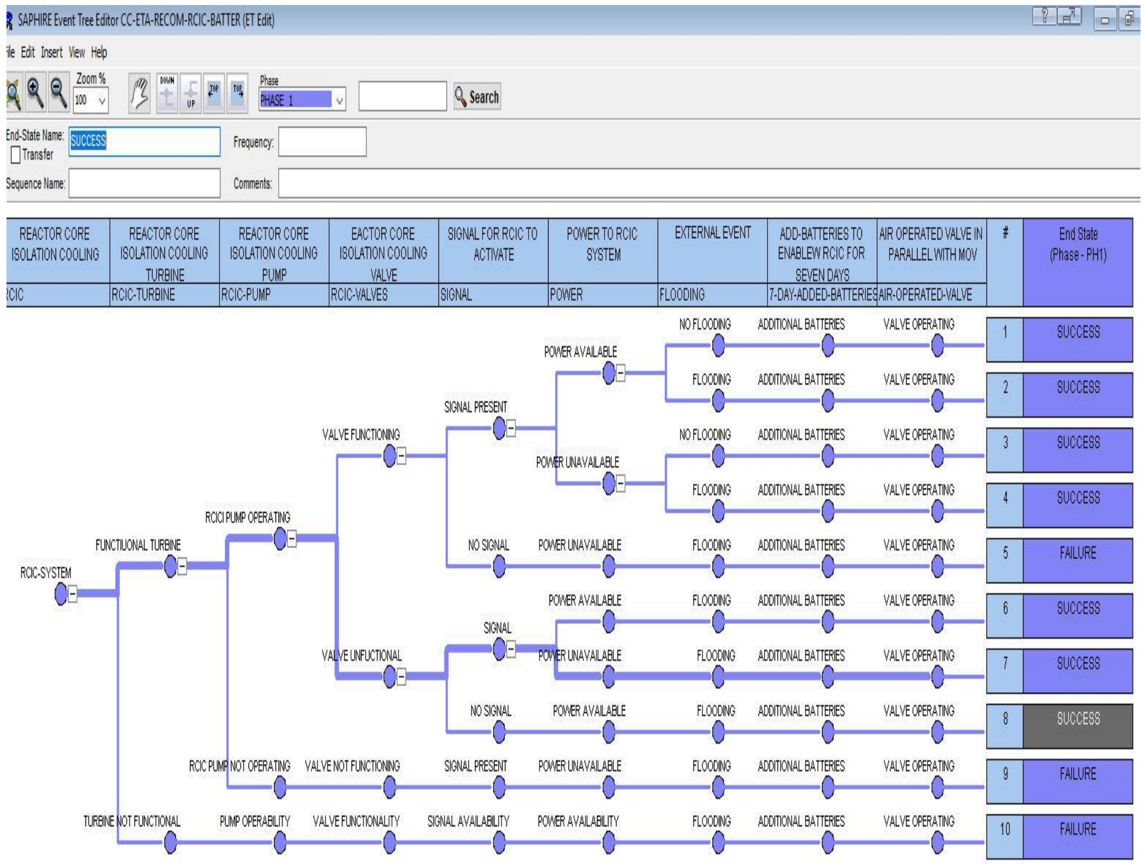

Figure 47. Event tree analysis for the RCIC with batteries and air-operated valve

For comparison basis, the results of these three cases are tabulated in Table 9. The values provide the numbers of successes and failures. As it can be easily deduced, implementation of the recommendations will increase the ratio of success to total cases. For this comparison, the number of cases is selected as 10. If the turbine does not function, or the pump does not function, variations in cases are excluded, since it will not result in functionality of the reactor core isolation cooling system. However, if extra sets of batteries are provided, the power is then available and those cases that branched out of “no power” become successes rather than failures. Similarly, those cases that were branched out of “unavailable valve” become successes when an air operated valve is placed in parallel to a motor-operated valve. These changes convert a failure to success. To demonstrate similar analysis to be done for different systems, will require this research to be extended. Table 9 shows that by adding the batteries, the success ratio will increase from two to four successes in ten trials, that is a 50 % increase in the success ratio. By adding an air-operated valve in addition to the batteries, e 4960% will convert from failure to success. The values of core damage frequency are calculated using these percentages.

Table 9. Comparative analysis for RCIC with and without recommendations

Parameters	No Recommendation	Batteries	Batteries and AOV
SUCCESS/TOTAL	2/10	4/10	7/10
CDF	$9x10^{-5}$	$5.4x10^{-5}$	$2.7x10^{-5}$

Table 10 shows the values of failure frequency for different systems with and without flooding. Further, it sows the difference and the percent improvement after implementing a flood wall of sufficient height preventing flooding .

Table 10. Comparing the results with and without flooding

	Systems	With the wall	W/O the wall	Difference	Improvement	Improvement in Percent
1	Diesel generators	0.0089000	0.0136000	0.00470000	0.346	34.558
2	Batteries	0.0000128	0.0000150	-0.00000220	0.147	14.667
3	Battery Chargers	0.0000058	0.0005700	-0.00056424	0.990	98.989
4	HPCI	0.0000139	0.0000279	-0.00001400	0.502	50.179
5	RCIC	0.0000279	0.0002790	-0.00025110	0.900	90.000
6	LPCI	0.0000139	0.0002790	-0.00026510	0.950	95.018
7	LPCS	0.0000139	0.0000638	-0.00004990	0.782	78.213
8	IC	0.0000085	0.0000140	-0.00000550	0.393	39.286
9	ADS	0.0000140	0.0001770	-0.00016300	0.921	92.090
10	SRV	0.0001800	0.0001770	0.00000300	0.017	1.695
11	SBLC	0.0001190	0.0001330	-0.00001400	0.105	10.526

7.4. Summary of the research

This research initially developed a blueprint for its activities and its progress. As initially conceived, it would have been a very large project. However, once the research began, the scope was reduced to a manageable and realistic scope. A key element of this research was the researcher's extensive experience in the commercial nuclear power industry in all of its facets. (For a detailed experience of the researcher's experience, see his curriculum vitae in Appendix C.)

After developing the framework and the structure of the research, the effort began the research activities. This subject was so interesting that it caused a large spectrum of people around the world to express their viewpoints and to conduct their own research in this area. The initiating step was to gather pertinent and available articles, books, and other written documents about the Fukushima Daiichi accident by other authors. In concert with this effort, the researcher met with the U.S. NRC and a member of a utility to understand their process in dealing with this issue. At the same time, there was a need to access a software program that performs probabilistic risk assessment. The researcher was able to acquire a single user license for a software program that was developed by the Idaho National Laboratory (INL) that was funded by the NRC. This is not the only software program that performs the probabilistic risk assessment, as there are other programs. However, there was no fee required for this license for the university students who were also in the nuclear engineering field. The key components of this research began to form. Initially the existing literature was examined. The next step was establishing the unique aspects of this research, and why this research was different than others. The next step of this research was to gather failure frequency data from a data source that is credible and acceptable. Since the Fukushima Daiichi was not a US nuclear power plant and was instead under the scrutiny of the IAEA, the research used the data from the IAEA data base.

The next step was initiating the fault tree and event tree analysis for 11 safety-related systems. These systems included various emergency core cooling systems and other safety-related systems that are essential for mitigating the consequence of the extensive core damages suffered by the Fukushima Daiichi nuclear power station. Once the fault tree analysis and event tree analyses were performed, it was noted the end result of the failure of each component under loss of power and flooding situations. The analysis focused on a single end result, that being core damage frequency.

One key question surfaced during this research process. There was interest in comparing this approach with the approach used by a utility. The researcher met with members of the probabilistic risk assessment group from a utility which owns multiple nuclear power plants. These two methods, one by this research and the utility's approach, were different in various key elements. These differences are delineated in Table 8 of this manuscript. For this reason, the utility staff was asked to perform a probabilistic risk assessment based on unavailability of one component within the low pressure cooling injection system using their process. The utility did that and provided their results being 3.0E-6.

This value compared with the research's result of being 1.39E-5. The difference between these two processes, came to a value of 1.09E-5. In other words, these two results compared within 78.4%. There appeared to be no reason to make comparisons with other systems and components. The additional scope of work would take a considerable amount of time, and the utility supervisor informed the researcher that they had prior commitments to support their plants.

When a common cause failure resulted in the failure of more than one component within a system, or a failure of more than a single system, resulting in a core damage frequency higher than the tolerated margin, the analysis applied a recommendation proposed by the research and redid the analysis. If the reanalysis showed the margin of the safety was restored or a higher margin was achieved, the effectiveness of the recommendation was established.

CHAPTER EIGHT - SPECIFIC RECOMMENDATIONS AND CONCLUSIONS

This chapter is dedicated to the specific recommendations made on the basis of this research and to provide the conclusion of this research.

8.1. Specific recommendations

This is the last phase of the research. Based on the review of the existing literature, the researcher reviewed the existing recommendations by various sources. These recommendations were evaluated for a typical boiling water reactor located on a coastal site susceptible to flooding. If the results of the probabilistic risk assessment analysis revealed a greater than acceptable risk of core damage frequency, specific recommendations were deemed necessary to reduce the core damage frequency to an acceptable value. However, if the probabilistic risk assessment analysis revealed a core damage frequency to fall within an acceptable range of values, no recommendations were deemed necessary.

The analysis focused on a flood event with sufficiently large wave(s) to cause the loss of all AC power, including the diesel generators. In this scenario, called ‘station blackout’, all AC power is unavailable and all emergency diesel generators are likewise unavailable for core cooling. Under such a scenario, all licensees must ascertain that their reactor will not get into a degraded core condition, that is the core damage frequency must not be below 1E-5. For those plants where the core damage frequency falls between 1E-4 and 1E-5, the utilities must implement modifications to the plant configurations and after the modification perform another analysis verifying that they reached a core damage frequency of 1E-5. A key assumption for the station blackout is that the plant has the availability of DC power for one day, and the AC power becomes available after one day. This assumption is based on the capacity of the reactor core isolation cooling system that has the ability to remove the decay heat for the first 24 hours. Once that time has passed, the station blackout assumes that AC power becomes available and other systems within the emergency core cooling systems, including the residual heat removal pumps, become fully functional.

Since the event of March 11, 2011, NRC has advised U.S. utilities to think beyond the design basis, in order to maintain reactors in safe conditions. Many recommendations have been made, including the acquisition of extra diesel generators that can be readily transferred via a helicopter to provide the necessary power for a plant facing the threat of a flood. The concept is to share the cost of the emergency diesel generators and their maintenance with other nuclear power plants that will become the beneficiary of such a device in a time of need. This concept is also being applied to cover concerns about a major geomagnetic event.

The researcher contends that there is a better solution. The flaw with the prevailing assumption in the station blackout analysis is the availability of AC power within one day. This assumption requires that the reactor core isolation cooling system, per its design requirements, removes the decay heat for 24 hours after a reactor shutdown. This limitation is based on the battery capacity. The quantity of batteries in a boiling water reactor plant is rather limited, only enough to provide power for one day. Because of this issue and concern, NRC generated NUREG/CR-7188 that wazs developed by the Brookhaven National Laboratory with its title BNL-NUREG-106282-2014.[79] This NUREG examines what is needed to extend the life of different battery cells to a maximum of 72 hours. In the research to generate the NUREG 7188, Brookhaven National Laboratory, in concert with different battery vendors, tested different battery types to make the batteries provide power for up to 3 days or 72 hours. While this research and this guideline established by the U.S. NRC is commendable, it is not the full answer. Nuclear physics tells us that decay heat continues to be produced for one week and longer, and that heat must be removed. What is not clear to the researcher is valid reasons for not doubling or tripling the number of batteries to ensure sufficient power to remove the decay heat for a full week. Three days or 72 hours does not equate to a whole week. Station batteries can be installed on the refueling floor or other large empty spaces inside the reactor building, or in other areas at high elevation in the reactor building, which is a Class 1 seismic structure. The extra batteries would meet the one week power requirements for the reactor core isolation cooling system to remove the decay heat and provide power to the critical instruments. This idea is far less costly and less uncertain than procuring extra diesel generators and transporting them it to a stricken plant. The added batteries can be kept in standby condition as are other station batteries. They can be activated one day after the shutdown has occurred and while there remains a need to remove the decay heat of the second day, the third day, and so on until the threatening level of decay heat is removed after seven days. This concept is not at all novel as everyone replaces batteries for devices that use batteries for a fixed period of time. In this case, the batteries take up to one day to become exhausted. After one day, fresh and different battery cells come into play, until they become exhausted. The switching is remarkably simple and it is easy to switch one row of cells to another row of cells, each with one day of capacity. This process can also be automated to reduce the role of a technician. Of course, without a degraded core, plant personnel are free to move about with no concern for radiation exposure.

The researcher has other recommendations for other actions. To combat the flood water, numerous ideas can be implemented. Naturally, each concept has its associated costs and complexities. The specific recommendations appear below:

A. Walls – It is never too late to erect walls surrounding facilities with sufficient height to preclude flood water from reaching the plant in general. If an extremely tall wall is unrealistic, then safety-related systems could be guarded by high walls and steps could be taken to make the equipment or component flood proof. Clearly, one must generate a cost benefit analysis for such a change. The key point with respect to cost is the[80] knowledge that the cost of the Fukushima Daiichi exceeded $630,000,000,000.[47] Further, the functionality of the plant was lost, forcing the utility, TEPCO, to shut down the plant. For illustration purposes, an aerial image of the Ft. Calhoun nuclear power plant speaks volume as the plant was inundated with flood water in June of 2011 (Figure 52).

Figure 48. A view of Ft. Calhoun plant inundated on June 14, 2011

B. Valves

There are thousands of valves in a typical nuclear power plant. Of key concern are the safety-related valves, which are motor driven. All motor-driven valves require electrical power, whether from AC power or DC power. When there is such a high demand for AC power in a nuclear power plant, one option could be to install different types of valves, such as air operated valves. There is a need for an air compressor to provide air to activate these valves. The air compressors require power for them to function. However, air cylinders with compressed air can be provided for emergency conditions to provide the air for the air operated valve actuators.

C. Pump Motors- Nearly all pumps require an electric power supply in order to operate, except for the turbine-driven pumps. There are two turbine-driven pumps in a boiling water reactor, in the reactor core isolation cooling system and the high pressure coolant injection system. Without AC or DC power, the pumps in a nuclear power plant will not function. The reactor core isolation cooling system needs the DC power in order to function. There is no option for a pump to function without a power source.

D. Water Storage Tanks- The standard condensate storage tank holds a large volume of water. It varies from 2000 to 30000 m^3 when the tanks are full. The volume of the tank at the Fukushima Daiichi nuclear power plant holds 550,000 gallons of water or 2083 m^3.[81] The tank design allows 150,000 gallons of water for the high-pressure coolant injection and the reactor core isolation cooling by having a suction line at the lowest level of the tank.[81] The design of the tanks is such that it provides adequate suction for the reactor core isolation cooling system to handle the decay heat for 28 hours, or at least 24 hours. However, prior to the water reaching a low level in the tank, it switches either manually or automatically to take suction from the suppression pool. The suppression pool has a volume of water equal to 115,585 ft^3 or 3,273 m^3 for MK I [82] and 134,000 ft^3 or 3794 m^3 for the MK II containment design. These values, convert to 864,635 and 1,002,269 gallons of water, for MK I and MK II, respectively. The researcher's recommendation is to install additional condensate storage tanks with the same capacity and the ability to provide water when the original source of water is depleted. Further, the researcher recommends that these tanks be installed at sufficiently high elevation to make them immune to flooding. The use of a pedestal or an enclosure for this purpose is a clear option. Another recommendation is to have these facilities filled at all times when a reactor is in operation. A high level sensor with an annunciation alarm must be installed to warn the operating staff when the water in the tanks is less than full. There is no sight glass on the condensate storage tanks to allow the operators see the water level in the tanks. Further, if a tank is less than full, it will have less capability to cool the core and a greater probability of a degraded core condition will follow. The condensate storage tanks at the Fukushima Daiichi station were 2/3 full. It must have been for this reason that the plant operators manually switched over from the condensate storage tank to the suppression pool.

E. Positive Displacement Pumps

Positive displacement pumps require a great deal of power and are typically not used for water applications. In a boiling water reactor or a pressurized water reactor, the standby liquid control pump needs to push sodium pentaborate fluid through the reactor. If this liquid becomes cool, it crystallizes and becomes a fluid with much higher viscosity. A fluid with higher viscosity requires a special pump to be able to push the fluid through the pump. Hence, a positive displacement pump is used.

In general, all pumps require power to deliver fluid from its source to its destination. Electric motors are normal for most pump types. Pumps can be driven by turbines, either by steam (such as high-pressure coolant injection and reactor core isolation cooling system) or by compressed air, by diesels or gasoline through an internal combustion engine process. The pumps can also be driven by turbines by water in dams and other areas. It is possible for critical pumps to have their own small generators driven by gasoline in special circumstances.

The researchers' recommendations and evaluation are based on the qualifications stated in Appendix C of this dissertation. The experience and technical qualification of the researcher uniquely justifies making these recommendations to mitigate similar consequences for another boiling water reactor at a coastal site of the U.S.

8.2. Conclusions of the research

This research had successful outcomes and met its objectives. It determined the consequences of failure of individual components within 11 different safety-related systems on the values of core damage frequency. Based on an analytic approach, using probabilistic risk assessment, this research determined that the failure of a single component, even to the extent of making the system unable to meet its intended safety function of other systems, did not result in a degraded core condition. The analysis shows that if all emergency core cooling systems do not fail to function, the degraded core condition will not take place. Even with the loss of all AC power, the degraded core condition will not take place, unless the DC system is unable to perform its function. Under such a scenario involving absence of both AC and DC power sources, the degraded core condition will have a failure probability of one, as happened at the Fukushima Daiichi station. The assumption that a plant loses all of its AC and the DC power (all batteries and battery chargers) for longer than one day is unlikely, but clearly possible. This scenario will invalidate the station blackout assumption, especially, if one assumes that the station will not incorporate NRC guidelines concerning the hydrogen recombiner, and the AC power is unavailable for longer than one day. Under this scenario, it yields only one outcome, a degraded core condition and an accident similar to the Fukushima Daiichi station.

The following factors form the basis of the conclusion that a similar event in the U.S. has a probability of less than 10^{-6} (one reactor event in 1,000,000 reactor years of operation.)

A. A wave of 35 meters high has never occurred at any of the reactors in the U.S.

B. All nuclear power plants in the US have installed some form of hydrogen recombiner for their containments. This factor alone will prevent hydrogen accumulation. There has been about 40 years of nuclear power plant operation with an average of 100 reactors per year. This indicates that not a single containment was breached in 4000 reactor years of operation in the U.S.

C. Unlike Japan, the regulatory agency, U.S. NRC, is not in a position to promote nuclear energy in the U.S. and based on its inspection and audit program, it has the means and the ability to impose notable sanctions against non-compliance and non-conformance to the established requirements. In the U.S., all of the utilities, except the TVA, are private companies and the regulator is part of the federal government. There is no conflict of interest.

D. In the U.S. there are numerous institutions that monitor the performance of the utilities and provide advice to the utility management. These entities are INPO, EPRI, NEI, and other professional associations, such as IEEE, ASME, ANSI, ACI, ASTM, ASHRAE, SAE, and others.

Despite all of these measures, the research demonstrated that there still can be improvements to further reduce the probability of degraded core conditions. These recommendations were discussed in Chapter 8, Subsection 8.1 of this dissertation. Of the various recommendations, the researcher strongly recommends that providing more battery and battery charger capacity to increase the capability of the reactor core isolation cooling system to remove the decay heat to one week will significantly mitigate the potential of having a degraded core. With decay heat removal capability of less than one week (current capacity), or even three-day capacity, being currently considered, the probability of the degraded core condition approaches 1.0 (assuming no AC and no DC, after a reactor shutdown.)

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APPENDIXES

APPENDIX A: TIMELINE OF THE FUKUSHIMA DAIICHI ACCIDENT

A summary of the impact of the damage to units 1-3 of the Fukushima Daiichi nuclear power station appears first. Table 10 is [83] taken from the Fukushima Accident report by the World-Nuclear Association located in London and the article was last updated in June of 2018. Please note that software mentioned in this table is a reference to modular accident analysis program (MAAP), developed and owned by EPRI. It is a licensed computer software, for the purpose of simulating and studying severe accidents for both primary systems and the containment during severe accidents.

Table 11. Event sequence of the units 1-3 at the Fukushima Daiichi station

	UNIT 1	**UNIT 2**	**UNIT 3**
LOSS OF AC POWER	+51 MIN	+54 MIN.	+52 MIN
LOSS OF COOLING	+1 HR	+70 HOURS	+36 HRS
WATER LEVEL DOWN TO TAF*!	+3 HRS.	+74 HOURS	+42 HRS
CORE DAMAGE STRATS*	+4 HRS.	+77 HOURS	+44 HRS
DAMAGE	+11 HRS.	UNCERTAIN	UNCERTAIN
FIRE PUMPS WITH FRESH WATER	+15 HRS.		+43 HRS.
HYDROGEN EXPLOSION (not confirmed for Unit 2)	+ 25 HRS. SERVICE FLOOR		+68 HRS. SERVICE FLOOR
FIRE PUMPS WITH SEAWATER	+28 HOUR		+46 HRS.
OFF-SITE ELECTRICAL SUPPLY	+ 11-15 DAYS		
FRESH WATER COOLING	+14-15 DAYS		

*: According to 2012 MAAP Analysis. TEPCO has decommissioned the reactors.

A detailed Timeline for the Fukushima plant appears in this Appendix. [84] However, "a select set of sequence of events highlighting [the role of] reactor core isolation cooling system appears prior to providing a detailed tabulated Timeline. The reference to RCIC sequence appears in bold. Because the entries are taken from a source.[85] the use of acronyms are left as is, The researcher modified some by entering his comments within a bracket.

HTTP://LIBRARY.THINKQUEST.ORG/25916/DATABASE/ALABAMA3.HTM "[85]

"3-11-11

14:46 AUTO SCRAM – SEISMIC

14:47 LOSS OF OFF-SITE POWER

14:47 TURBINE GENERATORS TRIP

14:47 EMERGENCY DIESEL GENERATORS START

14:47 MSIV'S CLOSE

14:52 SRV'S CONTROLLING PRESSURE IN AUTO

15:02 OPERATORS START REACTOR CORE ISOLATION COOLING SYSTEM

15:27 SERIES OF TSUNAMI WAVES BEGIN FLOODING THE TURBINE BUILDING AND THE REACTOR BUILDING

21:30 WORKERS BEGIN RUNNING TEMPORARY CABLE TO POWER [STANDBY LIQUID CONTROL] SLC PUMPS

3-12-11

02:55 REACTOR CORE ISOLATION COOLING SYSTEM VERIFIED IN SERVICE ON UNIT 2

04:00 OPERATORS SWITCH REACTOR CORE ISOLATION COOLING SYSTEM SUCTION TO TORUS [THE OPERATING PERSONNEL MUST HAVE HAD AN INDICATION OF THE CONDENSATE STORAGE TANK WAS UNAVAILABLE OR WAS NOT FUNCTIONAL]

3-13-11

02:42 REACTOR CORE ISOLATION COOLING SYSTEM MAINTAINING WATER LEVEL

3-14-11

11:01 BLOWOUT PANEL IN REACTOR BUILDING DISLODGED BY EXPLOSION IN UNIT 3

11:01 SECONDARY CONTAINMENT LOST [PRIMARY CONTAINMENT IS THE DRYWELL, WHEREAS SECONDARY CONTAINMENT IS THE REACTOR BUILDING ENCLOSING THE DRYWELL]

13:25 REACTOR CORE ISOLATION COOLING SYSTEM TRIPS RESULTING IN LOSS OF INJECTION INTO THE REACTOR

AT TIME OF [86] TRIP, INDICATED REACTOR WATER LEVEL WAS APPROX 95 INCHES (2400 MM) ABOVE THE TOP OF ACTIVE FUEL AND DRYWELL PRESSURE WAS 67 PSI (465 KPA)

17:17 INDICATED REACTOR PRESSURE VESSEL LEVEL BELOW TAF

18:00 OPERATORS [WERE] SUCCESSFUL IN OPENING AN SRV AND START[ED] TO DEPRESSURIZE THE REACTOR

18:22 REACTOR WATER LEVEL LOWERED BELOW THE BOTTOM OF THE INDICATING RANGE

19:20 WHILE TOURING TO CHECK STATUS OF FIRE ENGINES, WORKERS DISCOVERED THAT THE ENGINE HAD RUN OUT OF FUEL AND NO SEAWATER WAS BEING INJECTED INTO THE REACTOR

19:54 AFTER REFUELING AND STARTING A FIRE ENGINE, SEAWATER INJECTION COMMENCED INTO THE R[E]ACTOR VIA THE FIRE PROTECTION SYSTEM

23:00 BASED ON INCREASING REACTOR PRESSURE, OPERATORS SUSPECTED THAT THERE WAS NOT ENOUGH AIR LEFT TO OPEN THE SELECTED SRV. THE OPERATORS STARTED TO OPEN OTHER SRV SWITCHES IN AN ATTEMPT TO DEPRESSURIZE THE REACTOR

3-15-11

00:02 OPERATORS WORKED TO ALIGN THE CONTAINMENT VENT SYSTEM[,] HOWEVER, [86] CONTAINMENT PRESSURE REMAINED STABLE AT APPROX 102 PSI

00:22 OPERATORS CONTINUED CYCLING SRV CONTROL SWITCHES IN AN ATTEMPT TO DEPRESSURIZE THE REACTOR. REACTOR PRESSURE, HOWEVER, REMAINED ABOVE 160 PSIG

06:14 A loud noise was heard in the area around the torus. Operators in the unit 1-2 [main control room] MCR felt a shock – different than what they felt when unit 1 reactor building explo[a]ded. While suppression chamber pressure dropped to 0 psia indicating a potential instrument failure, drywell pressure remained high, indicating 105.9 psia, and reactor water level was 106 inches below TAF [Top of Activve Fuel: the fuel assemblies are 144 inches high. This implies that 106/144 inches of the fuel assemblies or 73.6% of the actual height of the fuel assemblies were dry]

UNIT 3 TIMELINE (REACTOR CORE ISOLATION COOLING SYSTEM EVENTS HIGHLIGHTED)

3-11-11

14:46 AUTO SCRAM – SEISMIC

14:47 LOSS OF OFF-SITE POWER

14:47 TURBINE GENERATORS TRIP

14:47 EMERGENCY DIESEL GENERATORS START

14:47 MSIV'S CLOSE

14:52 SRV'S CONTROLLING PRESSURE IN AUTO

15:06 OPERATORS START REACTOR CORE ISOLATION COOLING SYSTEM

15:26 REACTOR CORE ISOLATION COOLING SYSTEM TRIPS DUE TO HIGH REACTOR WATER LEVEL

15:27 SERIES OF TSUNAMIS [WAVES] BEGIN FLOODING IN TURBINE BUILDING AND REACTOR BUILDING

NOTE: POST TSUNAMI UNIT 3 HAD 125 VDC ON MAIN BUS PANELS A AND B

3-12-11

02:30 REACTOR CORE ISOLATION COOLING SYSTEM IN SERVICE, MAINTAINING THE REACTOR WATER LEVEL

11:36 REACTOR CORE ISOLATION COOLING SYSTEM MALFUNCTIONS, NO INJECTION INTO THE REACTOR VESSEL

12:35 OPERATORS START[ED] HIGH PRESSURE COOLANT INJECTION

3-13-11

02:42 OPERATORS SECURE HIGH-PRESSURE COOLANT INJECTION IN PREPARATIONS FOR OPENING A RELIEF VALVE AND INJECTING USING A DIESEL DRIVEN FIRE PUMP. THE RELIEF VALVE DOES NOT OPEN, AND REACTOR PRESSURE IS TOO HIGH TO INJECT WATER, RESULTING IN LOSS OF INJECTION INTO THE REACTOR

05:08 OPERATORS ATTEMPTED TO RESTART HIGH PRESSURE COOLANT INJECTION, THE STEAM STOP VALVE WOULD NOT REMAIN OPEN AND THE SYSTEM WOULD NOT START

09:08 OPERATORS OPEN[ED] A SRV ([SAFETY ~~STEAM]~~ RELIEF VALVE) TO DEPRESSURIZE THE REACTOR

3-14-11

01:10 INJECTION INTO THE REACTOR STOPPED BECAUSE OF A LACK OF WATER IN THE SEAWATER PIT

03:20 WORKERS MOVED THE FIRE ENGINE AROUND ALLOWING THE HOSE TO DROP DEEPER INTO THE SEAWATER VALVE PIT AND SEAWATER INJECTION INTO THE REACTOR WAS RESTORED USING A FIRE ENGINE

06:00 WORKERS BEGAN INJECTING BORIC ACID INTO THE UNIT 3 BACK WASH VALVE PIT

11:01 HYDROGEN EXPLOSION DESTROYING SECONDARY CONTAINMENT

11:01 11 WORKERS INJURED

11:01 DEBRIS DAMAGE[D] PORTABLE GENERATORS AND TEMPORARY POWER CABLING

REFERENCES:[85]

NRC – BOILING WATER REACTOR SYSTEMS

HTTP://WWW.NRC.GOV/READING-RM/BASIC-REF/TEACHERS/03.PDF

ANS – SAFETY SYSTEM DESCRIPTIONS FOR STATION BLACKOUT MITIGATION: ISOLATION CONDENSER, REACTOR CORE ISOLATION COOLING, AND HIGH-PRESSURE COOLANT INJECTION

HTTP://FUKUSHIMA.ANS.ORG/INC/FUKUSHIMA_APPENDIX_F.PDF

INPO – FUKUSHIMA DAIICHI ACCIDENT

HTTP://NAS-SITES.ORG/FUKUSHIMA/FILES/2012/08/INPO-MENG-AUGUST-2012-NAS-FINAL.PDF

“FUKUSHIMA FAILURE BY DESIGN

HTTP://BRAINMINDINST.BLOGSPOT.COM/2011/07/FUKUSHIMA-FAILURE-BY-DESIGN.HTML

PEACH BOTTOM REACTOR STATISTICS

HTTP://LIBRARY.THINKQUEST.ORG/25916/DATABASE/PENNSYLVANIA5.HTM

BROWNS FERRY REACTOR STATISTICS

HTTP://LIBRARY.THINKQUEST.ORG/25916/DATABASE/ALABAMA3.HTM “[85]

Fukushima Daiichi Unit 1 Accident Timeline

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature ºC	Containment Pressure, MPa	Source	Comment
3/11	14:46	0:00	Reactor SCRAM (large earthquake acceleration)					"Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety—The Accident at TEPCO's Fukushima Nuclear Power Stations," Government of Japan, June 2011 (Japanese Government Report)	
	14:47	0:01	All CR were fully confirmed inserted					Japanese Government Report	
			Turbine trip					Japanese Government Report	
			Loss of external power supply					Japanese Government Report	
			EDG start-up					Japanese Government Report	
			MSIV close					Japanese Government Report	
	14:52	0:06	IC automatic start-up (high RPV pressure)					Japanese Government Report	
	15:03	0:17	Reactor subcriticality confirmed					"Fukushima Daiichi Nuclear Power Station, Response After the Earthquake," Summary Report of Interviews of Plant Operators (TEPCO Operators Report), 8/10 update	
	~15:03	~0:17	IC shutdown					Japanese Government Report	Excessive RPV cooldown rate
	15:07 - 15:10	0:21 - 0:24	Reactor containment spray system pumps A and B were started up to cool the suppression chamber					Japanese Government Report	
	15:17	0:31	IC(A) restarted					Japanese Government Report, Attachment IV-1	
	15:19	0:33	IC(A) stopped					Japanese Government Report, Attachment IV-1	
	15:24	0:38	IC(A) restarted					Japanese Government Report, Attachment IV-1	
	15:26	0:40	IC(A) stopped					Japanese Government Report, Attachment IV-1	
	15:27	0:41	First tsunami wave arrives					TEPCO Operators Report	
	15:32	0:46	IC(A) restarted					Japanese Government Report, Attachment IV-1	
	15:34	0:48	IC(A) stopped					Japanese Government Report, Attachment IV-1	
			During this time after earthquake and before tsunami no HPCI start, as no L-L level					Japanese Government Report	
	15:35	0:49	Second tsunami wave arrives					TEPCO Operators Report	
	15:37	0:51	All AC power supplies lost. Water level indication lost, UHS lost					Japanese Government Report	

UNIT -1 TIMELINE 1

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature ºC	Containment Pressure, MPa	Source	Comment
			125V DC batteries flooded—no I&C					"IAEA International Fact Finding Expert Mission of the Fukushima Dai-ichi NPP Accident Following the Great East Japan Earthquake and Tsunami," (IAEA Mission Report), p. 30	15:50, according to TEPCO Operators Report
	15:42	0:56	TEPCO notification NEPA Article 10 (loss of all AC)					Japanese Government Report	
			HPCI determined to be inoperable due to loss of DC					TEPCO Operators Report	
	16:36	1:50	TEPCO notification NEPA Article 15 (loss of all ECCS) because water level cannot be confirmed					Japanese Government Report	
	16:45	1:59	Decided to cancel Article 15 notification because water level was confirmed					TEPCO Operators Report	
	17:07	2:21	Confirmation of water level lost—Article 15 notification again					TEPCO Operators Report	
	17:12	2:26	Site superintendant orders investigation of ways to inject water using fire system and fire trucks					TEPCO Operators Report, 8/10 update	
	17:30	2:44	Diesel-powered fire pump activated (standby)					TEPCO Operators Report	
	18:18	3:32	Manual opening of IC(A) supply isolation valve and return valve. Steam generation was observed, but the degree of IC function was not known.					IAEA Mission Report, p. 30	18:10, according to Japanese Government Report, Attachment IV-1. Also, if IC function was truly lost for over 2.5 hours, then core uncovery should have happened. So probably, IC was working, at least partially.
	18:25	3:39	IC(A) return valve closed					TEPCO Operators Report; and Japanese Government Report, Attachment IV-1	TEPCO Operators Report says plant operator closed the valve, but no reason is given
	20:00	5:14	Reactor pressure measurement	6.9				IAEA Mission Report, p. 31	
	20:00	5:14	HPCI determined to be inoperable due to loss of DC					TEPCO 6/16 presentation	Actually lost when tsunami hit because DC batteries are in the turbine building basement—see 15:42 item above
	20:30	5:44	MCR lighting temporarily restored					Japanese Government Report	20:49, according to TEPCO Operators Report and 8/10 report update
	21:19	6:33	Diesel fire pump lined up to add water to IC shell		200			Japanese Government Report	Level from TEPCO Operators Report
	21:23	6:37	Prime minister orders evacuation from within 3 km of Unit 1					Nuclear and Industrial Safety Agency, "Seismic Damage Information (the 81st release) (As of 16:00 April 8th, 2011)" (NISA Release #81)	
	21:30	6:44	IC(A) return valve manually opened; steam generation observed					Japanese Government Report	3 hours with no IC? Core should have melted. Maybe too little, too late.
	21:35	6:49	Diesel fire pump injecting water to IC shell					Japanese Government Report	
	21:51	7:05	High radiation level in reactor building: 290 mSv/hour					TEPCO Operators Report	

UNIT -1 TIMELINE 2

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature °C	Containment Pressure, MPa	Source	Comment
	22:00	7:14	RPV water level TAF +550 mm		550			TEPCO Operators Report	If this were so, then no radiation should have been observed, so reading is suspect. High DW temperature may give false high reading.
	23:00	8:14	Radiation rising in TB					Japanese Government Report	
3/12	0:30	9:44	Water being supplied to IC(A) shell side					Japanese Government Report	
	0:49	10:03	Possibility that DW pressure exceeded 600 kPa. TEPCO determined NEPA Article 15 (abnormal rise in containment vessel pressure occurred).				0.6	Japanese Government Report	It should have taken 20 hours to get to this pressure, unless a large amount of H_2 was generated. "Additional Report of the Japanese Government to the IAEA," September 2011 (Additional Japanese Government Report) says this was done by 23:50, 3/11; TEPCO Operators Report 8/10 update says 0:06, 3/12. Site superintendent orders preparations for venting.
	1:30	10:44	Proposed to government and obtained agreement to vent					TEPCO Operators Report	
	1:48	11:02	D/D FP trouble, shutdown					Japanese Government Report; and TEPCO Operators Report	
	2:30	11:44	DW pressure 840 kPa, RPV level 1300/530; RPV pressure ~840 kPa	0.84	900?		0.84	Japanese Government Report; and INPO 11-005 (for RPV pressure)	Suspect level. High DW temp; reference leg may have boiled. If RPV pressure = PCV pressure, then PCV pressure must be reduced to allow fire pump injection.
	2:45	11:59	Reactor pressure read using car batteries	0.8				IAEA Mission Report, p. 31	No mention of how RPV was depressurized, so pressure boundary may have been compromised
	3:58	13:12	A large aftershock was felt at the plant					TEPCO press release, March 12, 2011, 4AM update	
	4:15	13:29	DW pressure 840 kPa				0.84	Japanese Government Report	
	5:09	14:23	DW pressure 770 kPa				0.77	Japanese Government Report	Possibility of leak according to text
	5:14	14:28	Site radiation level rising and DW pressure decreasing					Japanese Government Report	
	5:44	14:58	Prime minister orders evacuation from within 10 km of Unit 1					NISA Release #81	
	5:46	15:00	Truck FP injection of freshwater to RPV					IAEA Mission Report, p. 31; and TEPCO Operators Report, 8/10 update	
	5:52	15:06	1 m^3 (~250 gal) water injected via CS line					TEPCO Operators Report, 8/10 update	
	6:30	15:44	2 m^3 (~500 gal) water injected via CS line					Japanese Government Report	
	6:50	16:04	Ministry order to implement vent					TEPCO Operators Report	
	7:11	16:25	Prime minister arrives at site					TEPCO Operators Report, 8/10 update	
	7:55	17:09	RPV level near TAF, 3 m^3 (~800 gal) injected		-100			Japanese Government Report	
	8:04	17:18	Prime minister departs site					TEPCO Operators Report, 8/10 update	Not mentioned in 1F1 timeline, but in 1F2 timeline

UNIT 1 TIME LINE 3

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature °C	Containment Pressure, MPa	Source	Comment
	8:15	17:29	4 m^3 (~1050 gal) water injected via CS line					TEPCO Operators Report, 8/10 update	
	8:30	17:44	5 m^3 (~1300 gal) water injected via CS line					Japanese Government Report	
	9:03	18:17	Evacuation of Okuma town confirmed					Additional Japanese Government Report, pg 6	
	9:04	18:18	Start of attempt to vent					Japanese Government Report	
	9:15	18:29	6 m^3 (~1600 gal) water injected via CS line					Japanese Government Report	Average rate 7 gpm. Need 56 gpm to keep up with decay heat. Must be near shutoff head of fire pumps.
	~9:15	~18:29	SC vent MO valve opened 25%					Japanese Government Report	Per procedure. 25% open should be enough. 9:24, according to TEPCO Operators Report.
	~9:30	~18:44	SC second AO valve open attempt stopped due to high radiation					Japanese Government Report	
	9:40	18:54	21 m^3 (~5550 gal) water injected					Japanese Government Report	Rate up to 160 gpm
	10:17	19:31	SC second AO valve open attempt from control room					Japanese Government Report	This attempt seems to have been successful
	10:40	19:54	As radiation level at main gate increased, judged that vent was working					TEPCO Operators Report	
	11:15	20:29	Radiation levels falling, so venting may have stopped					TEPCO Operators Report, 8/10 update	
	12:55	22:09	RPV level -1700/-1500; DW pressure 750 kPa		-1600		0.75	Japanese Government Report	How can DW pressure stay at ~0.8 MPa for an additional 10 hours after first getting there? There has to be a leak.
	~14:00	~23:14	Additional local attempt to open second AO valve					Japanese Government Report	Using compressor, per Additional Japanese Government Report
	14:30	23:44	Operators confirmed decrease in DW pressure					IAEA Mission Report, p. 32; and TEPCO Operators Report	
	14:50	24:04	DW pressure 0.58 MPa				0.58	TEPCO Operators Report	
	14:53	24:07	80 m^3 (~21,000 gal) water injected. Freshwater ran out.					Japanese Government Report	Average 46 gpm
	15:18-15:36	24:32-24:50	Working on restoration of SLCS to pump water, using truck power supply					TEPCO Operators Report, 8/10 update	
	15:36	24:50	Aftershock					TEPCO press release, March 12, 2011, 5PM update	
	~15:36	~24:50	Hydrogen explosion[1]					IAEA Mission Report, p. 32	Coincidence, or cause of explosion?
	16:17	25:31	Article 15 : radiation levels at the site boundary exceed limits					TEPCO press release: "Occurrence of a Specific Incident Stipulated in Article 15, Clause 1 of the Act on Special Measures Concerning Nuclear Emergency Preparedness (Extraordinary increase of radiation dose at	16:27, according to TEPCO Operators Report 8/10 update

[1] As used in this timeline, the term "explosion" could mean either a "deflagration" or a "detonation." Which of these rapid hydrogen combustion events actually took place is still under study.

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature °C	Containment Pressure, MPa	Source	Comment
								site boundary)," March 12, 2011	
	18:25	27:39	Prime minister orders evacuation from within 20 km of Fukushima Daiichi NPS					NISA Release #81	
	17:20-18:30	26:34-27:44	Investigating the condition of fire trucks, SLCS power supply, hoses, etc., and found unusable due to explosion					TEPCO Operators Report, 8/10 update	
	19:04	28:18	Injection of seawater by fire trucks started					Japanese Government Report	
	20:45	29:59	Injection of boric acid mixed with seawater started					Japanese Government Report	
3/13	3:38	36:52	Injection of seawater via fire line					Japanese Government Report	

UNIT - 1 TIMELINE 5

Fukushima Daiichi Unit 2 Accident Timeline

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature ºC	Containment Pressure, MPa	Source	Comment
3/11	14:47	0:00	Reactor SCRAM (large earthquake acceleration)					"Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety—The Accident at TEPCO's Fukushima Nuclear Power Stations," Government of Japan, June 2011 (Japanese Government Report)	
			All CR were fully confirmed inserted					Japanese Government Report	
			Turbine trip					Japanese Government Report	
			Loss of external power supply					Japanese Government Report	
			EDG start-up					Japanese Government Report	
			MSIV close					Japanese Government Report	
	14:50	0:03	RCIC was manually started up					Japanese Government Report	
	14:51	0:04	RCIC high RPV level trip (L-8)					Japanese Government Report	
	15:00	0:13	RHR pumps started up for SP cooling					Japanese Government Report	
	15:01	0:14	Reactor subcriticality confirmed					"Fukushima Daiichi Nuclear Power Station, Response After the Earthquake," Summary Report of Interviews of Plant Operators (TEPCO Operators Report), 8/10 update	
	15:02	0:15	RCIC was manually started up					Japanese Government Report	
	15:07	0:20	RHR pumps were ended sequentially					Japanese Government Report	Some confusion between table and text. Text says RHR pumps ran until tsunami arrived.
	15:27	0:40	First tsunami wave arrives					TEPCO Operators Report	
	15:28	0:41	RCIC trip (L-8)					Japanese Government Report	
	15:31	0:44	HPCI seems to be inoperable due to loss of DC					TEPCO 6/16 presentation	Timing? The tsunami must have arrived first.
	15:35	0:48	Second tsunami wave arrives					TEPCO Operators Report	
	15:39	0:52	RCIC was manually started up					Japanese Government Report	
	15:41	0:54	All AC power supplies lost					Japanese Government Report	I&C lost also, meaning DC bus also failed
	15:42	0:55	TEPCO notification NEPA Article 10 (loss of all AC)					Japanese Government Report	
	16:36	1:49	TEPCO notification NEPA Article 15 (loss of all ECCS)					Japanese Government Report	
	17:12	2:25	Start of planning for water injection via fire pump. RHR valves aligned to permit low-pressure injection to RPV when RPV depressurized.					TEPCO Operators Report	

UNIT 2 TIMELINE 1

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature °C	Containment Pressure, MPa	Source	Comment
	20:30	5:43	RCIC under shutdown					Japanese Government Report	Some conflict in sources. According to "IAEA International Fact Finding Expert Mission of the Fukushima Dai-ichi NPP Accident Following the Great East Japan Earthquake and Tsunami" p. 31, the RCIC ran continuously for 3 days although status could not be confirmed in the control room. Unless RCIC operation was made manual, it should have tripped several times due to high water level and restarted at low water level.
			MCR lighting temporarily restored					Japanese Government Report	20:49, according to TEPCO Operators Report
	21:02	6:15	Reported risk of water level reaching TAF because of uncertainty in water level					TEPCO Operators Report	
	21:23	6:36	Prime minister orders evacuation from within 3 km of Unit 1					Nuclear and Industrial Safety Agency, "Seismic Damage Information (the 81st release) (As of 16:00 April 8th, 2011)" (NISA Release #81); and TEPCO Operators Report, 8/10 update	
	22:00	7:13	RPV water level TAF +3400 mm		3400			Japanese Government Report	21:50, according to TEPCO Operators Report, 8/10 update
	22:47	8:00	RCIC operation cannot be confirmed					Japanese Government Report	Confusion—see comment at 20:30, above
	23:25	8:38	RPV pressure 6.3 MPa	6.3				Japanese Government Report	
	23:55	9:08	Drywell pressure 40 kPa				0.14	Japanese Government Report	After 9 hours of isolation? Must be a leak. Must mean 40 kPag.
3/12	0:00	9:13	RPV water level at 3500 mm		3500			Japanese Government Report	
	0:30	9:43	RCIC under shutdown				0.14	Japanese Government Report	Confusion—see comment at 20:30, 3/11, above
	1:30	10:43	Proposed to government and obtained agreement to vent					TEPCO Operators Report	
	2:55	12:08	RCIC start-up state was checked					Japanese Government Report	Found operational per TEPCO Operators Report
	3:00	12:13	Evacuation ordered for 3-km radius from plant					TEPCO press release, March 12, 2011, 3AM update	
	3:58	13:11	A large aftershock was felt at the plant					TEPCO press release, March 12, 2011, 4AM update	
	4:20 - 5:00	13:33 - 14:13	RCIC water supply was switched from CST to SC					Japanese Government Report	To avoid high water level in SP, according to text
	4:55	14:08	Rise in radiation level within station grounds					TEPCO Operators Report, 8/10 update	The 1F1 timeline says 05:14
	5:44	14:57	Prime minister orders evacuation from within 10 km of Unit 1					NISA Release #81	
	6:50	16:03	Ministry order to implement vent					TEPCO Operators Report	
	7:11	16:24	Prime minister arrives at site					TEPCO Operators Report, 8/10 update	

UNIT - 2 TIMELINE 2

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature °C	Containment Pressure, MPa	Source	Comment
	8:04	17:17	Prime minister departs site					TEPCO Operators Report, 8/10 update	
	15:36	24:49	Aftershock					TEPCO press release, March 12, 2011, 5PM update	
	16:17	25:30	Article 15 : radiation levels at the site boundary exceed limits					TEPCO press release: "Occurrence of a Specific Incident Stipulated in Article 15, Clause 1 of the Act on Special Measures Concerning Nuclear Emergency Preparedness (Extraordinary increase of radiation dose at site boundary)," March 12, 2011	16:27, according to TEPCO Operators Report, 8/10 update
	17:30	26:43	Station general manager orders venting preparation				0.2	TEPCO Operators Report	
	18:25	27:38	Prime minister orders evacuation from within 20 km of Fukushima Daiichi NPS					NISA Release #81	
3/13	3:00	36:13	DW pressure rises to 315 kPa				0.315	Japanese Government Report	Hard to believe it takes 36 hours to get containment to this pressure if it is the only heat sink, unless there is already a leak in the containment boundary
	8:10	41:23	PCV vent valve (MOV) opened 25%					TEPCO Operators Report	
	8:56, and again at 14:23 on 3/13 and at 4:24, 5:37, 8:00, and 9:34 on 3/14	42:09	Article 15 : radiation levels at the site boundary exceed limits					TEPCO Operators Report.	Times reported by NISA Release #81 are a little different
	11:00	44:13	Second valve set to "open" for venting					Japanese Government Report	No venting will occur until rupture disk setpoint of 0.549 MPa is reached. Note that this is above the design pressure of the PCV (0.53 MPa).
	12:05	45:18	Plant general manager orders preparation for seawater injection					TEPCO Operators Report	
3/14	11:01	68:14	Could not confirm that the SC side valve was open. Prepared water injection line not available.					Japanese Government Report	Due to explosion in Unit 3, per TEPCO Operators Report
	>11:00	>68:13	Blowout panel in RB opened by explosion in Unit 3					NISA Release #81	
	12:00	69:13	SC temperature 147ºC and SC pressure 485 kPa and increasing. Water level decreasing.		3400	147	0.485	Japanese Government Report	Text also says RCIC was running at this time
	12:30	69:43	RPV level 2950 mm (A), 3000 mm (B)		2950			Japanese Government Report	
	13:05	70:18	Reconfigured seawater injection line, including fire truck					TEPCO Operators Report, 8/10 update	
	13:25	70:38	RCIC shutdown (assumed)					Japanese Government Report	Must be, as RPV level is decreasing, per TEPCO Operators Report, 8/10 update

UNIT 2 TIMELINE 3

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature ºC	Containment Pressure, MPa	Source	Comment
			TEPCO notification NEPA Article 15 (loss of reactor cooling)					Japanese Government Report	
	15:00	72:13	RCIC operation state was being checked					Japanese Government Report	
	15:28	72:41	Authorities notified that TAF expected by 16:30					TEPCO Operators Report, 8/10 update	
	16:00	73:13	Started operation to open SC side valve		300			Japanese Government Report; and INPO 11-005 (for RPV water level)	
	16:20	73:33	Confirmed that the SC side valve was closed					Japanese Government Report	
	16:34	73:47	Depressurization of RPV was started using SRV, and seawater injection was started using fire engine lines					Japanese Government Report	
	17:17	74:30	Water level reached TAF		0			Japanese Government Report	Text says 16:20, and this is more consistent with boiloff rate
	~18:00	~75:13	Reactor pressure decrease was observed. Problems with air pressure for SRV and excitation of the admitting solenoid, so SRV seemed to be closed, as RPV pressure increased.	5.4				Japanese Government Report	This is a key problem. Inability to depressurize the RPV is why no low-pressure water can be added.
	18:22	75:35	RPV level is TAF -3700 (BAF)		-3700			Japanese Government Report	5 hours to boil off water to BAF consistent with no makeup
	19:03	76:16	RPV pressure	0.63				Japanese Government Report	
	19:20	76:33	Fire pumps for seawater injection stopped due to lack of fuel					Japanese Government Report	
	19:54	77:07	Seawater injection started—first fire pump started up					Japanese Government Report	No injection is going to occur until RPV pressure is low
	19:57	77:10	Second fire pump started up					Japanese Government Report	
	21:00	78:13	Operation of opening SC side small valve success unknown				0.42	Japanese Government Report; and INPO 11-005 (for containment pressure)	TEPCO Operators Report says it worked, but rupture disk pressure not reached.
	21:03	78:16	RPV pressure decreased	1.418				Japanese Government Report	Must mean relative to rated pressure—already reported as 0.63 MPa at 19:03. This is above head of fire pumps.
	21:20	78:33	By opening 2 SRV, RPV depressurization and water level restoration were confirmed. Thereafter, due to problems including air pressure for driving SRV and maintaining excitation of solenoid valve controlling air supply, the opening and closing operation of SRV seemed to be performed.					Japanese Government Report	From Figure IV-5-5, when RPV pressure spikes up to 3 MPa, fire pumps cannot add water
	~21:20	~78:33	It was observed that RPV water level tended to recover		-3000			Japanese Government Report	

UNIT 2 TIMELINE 4

Date	Time	Δ Time from Begin	Item	RPV Pressure, MPa	RPV Level Above TAF, mm	SP Temperature °C	Containment Pressure, MPa	Source	Comment
	22:14	79:27	Reactor water level recovered to -1800 mm. Core damage thought to be 5% or less		-1800			Japanese Government Report	
	22:50	80:03	DW pressure exceeded design pressure. Operator determined NEPA Article 15 event (abnormal increase in containment pressure).				0.54	Japanese Government Report	There is no way containment can last 80 hours as a heat sink unless some venting is occurring
	23:35	80:48	Decide to open small DW vent path, since DW pressure higher than WW pressure					TEPCO Operators Report	
	23:44	80:57	Measurements				0.75	INPO 11-005	Drywell pressure
3/15	0:00	81:13	CAMS reading went up by 3 to 4 orders of magnitude					Japanese Government Report	Means fuel is melting
	0:02	81:15	Valve set to open for drywell venting					Japanese Government Report	
	0:45	81:58	Reactor pressure at 1823 kPa	1.823				Japanese Government Report	
	3:00	84:13	DW pressure at 750 kPa				0.75	Japanese Government Report	
			Since DW pressure exceeded design pressure, RPV depressurization was begun to allow injection into the reactor, but RPV not sufficiently depressurized					Japanese Government Report	With high PCV pressure, not possible to depressurize RPV fully; 0.5 MPa PCV pressure will only allow RPV to depressurize to 3 MPa (430 psia) with some SRV designs
	5:00	86:13	RPV pressure decreased	0.626				Japanese Government Report	
	~6:00 - 6:10	~87:13 - 87:23	Explosion,[2] thought to be hydrogen, came from near the SC. All personnel evacuated except those necessary for operation. SC pressure unknown.	unknown	-2800		0.73	Japanese Government Report	There are second thoughts about whether it was a hydrogen explosion. (Source: "No explosion at No. 2 reactor. TEPCO: Only 3 hydrogen blasts occurred at Fukushima N-plant," *Daily Yomiuri*, October 3, 2011.) Also, hydrogen not mentioned in TEPCO Operators report, 8/10 update. Maybe the containment leak is at this location.
	7:00, and again at 8:36, 16:22, and 23:20 on 3/15	88:13	Article 15 : radiation levels at the site boundary exceed limits					TEPCO Operators Report	Times reported by NISA Release #81 are a little different
	8:25	89:38	White smoke (seemed to be steam) observed near the fifth floor of RB					Japanese Government Report	
	11:00	92:13	Prime minister directs indoor refuge for people living within 20 to 30 km from the site					TEPCO Operators Report, 8/10 update	
	15:25	96:38	Reactor pressure lower than DW pressure	0.119			0.174	Japanese Government Report	Means no leak in RPV boundary?
	15:30	96:43	Core damage estimate changed from14% to 35%					Japanese Government Report	

UNIT 2 TIMELINE 5

APPENDIX B: LIST OF U.S. COASTAL NUCLEAR POWER PLANTS PRONE TO FLOODING

This subsection covers the 15 nuclear power plants that are susceptible to tsunamis.

1. Diablo Canyon, San Luis Obispo, CA

Diablo Canyon is located in the State of California around San Luis Obispo (between Los Angeles and San Francisco). This plant with two pressurized water reactor reactors [87] was built with full consideration of seismic potential as it is close to the San Andreas fault. It is also located at high elevation and there is little chance of being affected by tsunamis that have historically been experienced in that location. NRC requires all nuclear power plants for their safety-related systems and components to be designed, operated and maintained to and tsunamis with full capability of performing their intended safety functions. In all designs, they consider the worst event in the past 100 years and add 10% safety margin on top of that. For instance, during a major hurricane, the containments must be able to withstand firm when a telephone poll carrying maximum velocity impinges on the containment building horizontally at 90-degree angle. The sheer force at other angles is considerably less.

The design basis of the Diablo Canyon plant is to be able to sustain a maximum oceanic flood level of combined tsunami, storm wave, high tide, and storm surge of 32 feet above sea level [87]. The plant's intake structure was designed with an elevated air intake so that the design Class 1 (Class 1 design is a standard classification for safety-related systems and components, and all

Class 1 structures and material are referred to as such), such that saltwater pumps can take suction and operate during the tsunami wave hitting the facility as high as +48 feet above minimum low-low water level [87]. The pump motors are placed in watertight rooms within the plant intake structure. In addition to these design considerations, there are numerous other features that make the plant less susceptible to adverse effect of a tsunami. They are:

> "The diesel generators are at 85' above sea level in a turbine building.
> Diesel fuel storage tanks 85' above sea level west of the turbine building.
> Batteries are located at the 119' above sea level in an auxiliary building.
> The power block is on a cliff 85' above sea level enclosed in concrete.
> Reservoirs and dry cask storage are located 310' above sea level.
> The Spent Fuel Pool is at 140' elevation of the Fuel Handling Building." [87]

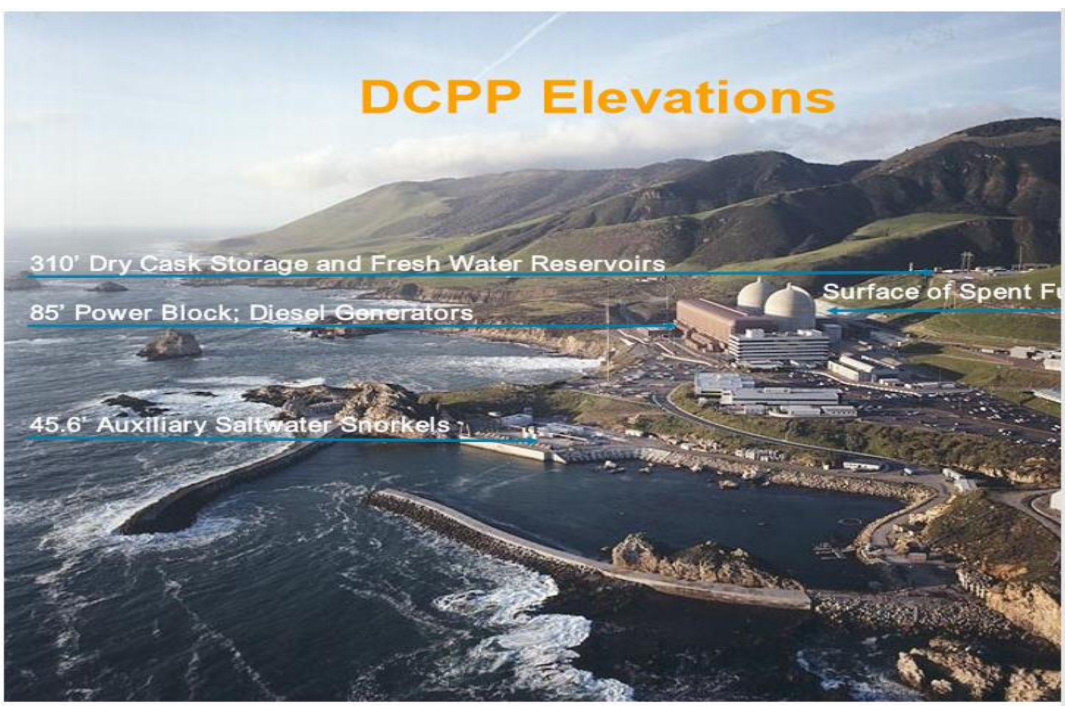


Figure B - 1. DIABLO CANYON [87]

2. Saint Lucie, Jensen Beach, FL

Saint Lucie contains two pressurized water reactors located on Hutchinson Island, near Port St. Lucie in St. Lucie County, Florida [88]. The architect engineering firm designing the plant was Combustion Engineering for Florida Power & Light, and Westinghouse manufactured its turbine generators. [88] Unlike many nuclear power plants that use hyperbolic cooling towers, this plant uses the ocean water for cooling the condensers. This plant is licensed to operate with its extension to 2036 and 2043 for Units 1 and 2, respectively. Because this site takes suction from the ocean for cooling purposes, and its proximity to the sea level, it is susceptible to a tsunami.

Figure B - 2. SAINT LUCIE [88]

3. Turkey Point, Homestead, FL

The Turkey Point Nuclear Generating Station contains two pressurized water reactors located two miles east of Homestead, Florida, which is about 25 miles south of Miami, Florida.[89] This plant is owned and operated by Florida Power and Light, the same utility that runs the St. Lucie nuclear power plant. The utility plans to build two additional pressurized water reactor reactors manufactured by Westinghouse (AP1000)[89]. The two additional reactors approved by the state are scheduled to begin construction in 2017 with a total capacity of 3330 MWe, including an oil/natural gas station and a Combined Cycle gas-fired unit. This station has experienced a sea level rise near the plant, hurricanes and storm surges. In the early morning of August 24, 1992, Hurricane Andrew came ashore just eight miles from the plant [89]. The storm was one of the worst to hit Florida's eastern shore in history, and when it made landfall, it sent 175 mile/hour wind across the plant and pushed a 16-foot surge of water towards the reactors [89] . One notable difference between hurricanes and the tsunamis is worth mentioning. The plant staff are aware of the impending hurricanes and have time to trip the reactor and to bring to hot standby. In tsunamis, the luxury of early warning does not exist.

Figure B - 3. TURKEY POINT [89]

The staff at Turkey Point started the shutdown process 12 hours before Hurricane Andrew's arrival. As in the Fukushima plant, the most challenging event was the loss of off-site power. However, unlike the Fukushima where the off-site power took months to restore, it took five days at the Turkey Point, which had the diesel generators providing power in the five-day period to maintain core cooling. In 1993, NRC in the review of Turkey Point stated that: "Hurricane Andrew is historic because this is the first time that a hurricane significantly affected a commercial nuclear power plant" [89]. It took six months to repair the damages Andrew caused before Turkey Point plant resumed operation.

4. Waterford 3, Killona, LA

When Hurricane Katrina hit Louisiana on August 29, 2005, the Waterford 3 plant did not sustain notable damage [90]. Waterford 3 is about 20 miles west of New Orleans, and is a pressurized water reactor built by Combustion Engineering, and its turbine generator was manufactured by Westinghouse. It started operation in 1985, and its license expires in 2024. It receives its cooling water from the Mississippi River and is about 25 miles from the Atlantic Ocean. It is located in Killona, LA at about 20 feet above sea level [90].

Figure B - 4. Waterford 3 [90]

5. Pilgrim, Plymouth, MA

The Pilgrim is a single boiling water reactor 3 reactor, the only nuclear power plant in Massachusetts. It is located in the Manomet section of Plymouth on Cape Cod Bay, near Priscilla Beach [91]. It is on the Atlantic Ocean near the Plymouth Rock. Bechtel Power Company built this General Electric boiling water reactor Mark I containment design for a low cost of $231 million. It was sold in 1999 to Entergy Corporation, [91] which keeps its irradiated fuel in an on-site wet fuel pool. The plant was licensed to operate until 2012; however, its license was extended by 20 years to 2032. In 2015, Entergy announced that it would not operate the plant past 2019 due to the costs associated with its retrofits to meet safety requirements.

On August 22, 2013, at 98% power, all three main feedwater pumps tripped, causing a notable drop in the reactor water level. Normally two feedwater pumps are operating and the third is kept on hot standby in case one pump trips. As a consequence of the tripping of all three pumps, there was not water feeding the pressure vessel and the result was a sudden drop in the water level, which is one of many signal inputs to scram the reactor. After water level dropped to -46 inches, the emergency core cooling systems automatically activated and made up water to the reactor pressure vessel. The reactor core isolation cooling system and high-pressure coolant injection system promptly restored the reactor water to normal level.

Figure B - 5. Pilgrim [91]

6. Salem, Lower Alloways Creek, NJ

Salem and Hope Creek Nuclear Power Station share artificial island in the Delaware Bay in Wilmington, New Jersey. Salem has two pressurized water reactors built by Westinghouse. The reactors started their commercial operation in 1977 (Unit 1) and 1981 (Unit 2) [92]. Each has an electrical capacity of 1137MWe, providing a total capacity of 2,275 MWe. Unit 1 has a license to operate until August 13, 2036, and Unit 2 until April 18, 2040 [92]. In 2009, the owner applied to the NRC for a 20-year extension, which was granted in 2011.

MSNBC listed the risks associated with earthquakes for all nuclear power plants, including Salem. It identified both plants and ranked them 62 and 63 on the earthquake list [92]. The same list identifies Indian Point in New York and Pilgrim as number 1 and number 2 highest risk. The risks for the Salem plant, as performed by the NRC, estimates an earthquake with sufficient magnitude to cause core damage is 1 in 90,909. Salem 1 and 2 use Delaware Bay for cooling the condensers, using once-through cooling with no cooling tower. The heat from the condensers is exhausted to the Bay resulting in a temperature increase of 1°C in summer months and to 2°C the rest of the year [92] .

Figure B -6. SALEM [92]

7. Hope Creek, Hancocks Bridge, NJ

Hope Creek is a single unit boiling water reactor plant with total generating electrical power of 1,176 MWe, located next to the Salem plant. The construction began in 1974 with commercial operation in 1986. With its power uprate addition, its end of life is extended to 2046. This is another nuclear power plant owned and operated by Public Service Electric and Gas (PSEG). The other nuclear units are Oyster Creek (boiling water reactor) and Salem (pressurized water reactor). The NRC estimates, published in August 2010, the risks of an earthquake of a significant magnitude to cause a core damage is 1 in 357,143 [93]. There is a natural-draft cooling tower at Hope Creek, which can be seen from many miles away. The tower serves only Hope Creek's single reactor. The plant has a cylindrical reactor building with a dome, making its shape similar to a pressurized water reactor plant. However, this similarity is only in appearance, and like other boiling water reactors, the actual containment vessel for the reactor is a separate drywell/torus structure enclosed within the reactor building. The building is the secondary containment and includes many of the reactor's safety systems and components [94].

Figure B - 7. HOPE CREEK [94]

8. Indian Point, Buchanan, NY

Indian Point has three pressurized water reactor reactors located in Buchanan, south of Peekskill, New York on the east bank of the Hudson River. It is 36 miles north of Manhattan[95]. The plant generates 2000 MWe of electrical power that was owned and operated by Consolidated Edison (Con Ed), now part of the Entergy Corporation who bought the plant from Con Ed. Of the three reactors, Unit 1 is permanently shut down. The 40-year operating licenses for

Units 2 and 3 expired in September 2013 and December 2015, respectively. However, Entergy applied for license extensions, and NRC is moving toward granting a 20-year extension for each reactor. As of this writing, both are operating at 100%, and the final Commission decision is yet to be determined. The license application has gone through two Environmental Impact Statement (EIS) processes. Despite the governor of the state and other environmentalists opposition to the NCR granting license renewal, chances are high that NRC will grant the Unit 2 and Unit 3 a 20-year extension [95].

Entergy's Indian Point Energy Center (IPEC) seen from across the Hudson River

Figure B - 8. Indian Point [95]

9. South Texas Project, Wadsworth, TX

South Texas Nuclear Generating Station is located in Bay City, Texas. It contains two pressurized water reactor reactors on the Colorado River, about 90 miles from Houson, Texas. The reactors are cooled by a 7,000-acre reservoir, which does not necessitate cooling towers. The initial construction company was Brown and Root, and the project became delayed for two years with a large cost overrun. Finally, the Houston Light and Power selected Ebasco and Bechtel Power to replace Brown and Root. For a $1M initial construction cost, they finished it for $4.4-4.8 B. The Unit 1 reached criticality on March 8, 1988, and went commercial on August 25, 1988. Unit 2 reached initial criticality on March 12, 1989, and went commercial on June 19, 1989. In 2016, the NRC issued a construction license for two additional Advanced boiling water reactor giving the plant output of 3260 MWe. The NRC estimates the risk of a substantial earthquake damaging the core to be 1 in 158,630[95] .

South Texas Nuclear Project, Units 1 & 2 (NRC image)

Figure B - 9. SOUTH TEXAS PROJECT [96]

10. North Anna, Mineral, VA

North Anna located in Louisa County, Mineral, Virginia, has two pressurized water reactor reactors manufactured by Westinghouse; Unit 1 going commercial in 1978 and Unit 2 in 1980. The two reactors provide 1790 MWe and they are operated by the Dominion Generation Company. In March 2003, NRC approved life extension for the reactors for an additional 20 years. The reactors are cooled by a man-made lake, Lake Anna, on the North Anna River; the reservoir provides cooling water for the units' condensers. The NRC estimates the risk of an earthquake sufficiently high to do core damage to be 1 in 22,727 [96]. This probability is primarily due to the presence of two distinct seismic zones in Virginia: one near Central Virginia, and the other in Giles County [96] . Both zones produce small recurrent earthquakes at least every few years. The plant located 40 miles northwest of Richmond lies within the Central Virginia seismic zone.

Figure B - 10. NORTH ANNA [97]

11. Surry, Surry, VA

Surry is located in Surry County in southeastern Virginia, in the south Atlantic Ocean. The power station is adjacent to the James River across from Jamestown. It is operated by Dominion Generation, but owned by Dominion Resources, Inc. It has two pressurized water reactor reactors, each 800 MWe, manufactured by Westinghouse and became commercial in 1972 and 1973 [98] The plant is similar in appearance and design to the North Anna Power Station. The reactors are cooled by the James River.

In 2003 the NRC extended the plant's license for an additional 20 years. Further, in 2016, the utility asked the NRC for an additional 20-year extension, which would make it 80 years. No other plants in the US have operated or received a license to operate for 80 years. With this extension, the license will be extended to 2052 and 2053 [98]. There has been an extensive opposition to this request. There are also many environmentalists who are avid fans of the use of the nuclear energy in Virginia. These two reactors have not provided radioactive effluents to the environment, and have significantly reduced carbon emission[98]. For the past decade, the Surry and North Anna plants have had a capacity factor above 90% and are amongst the highest capacity factor plants in the US.

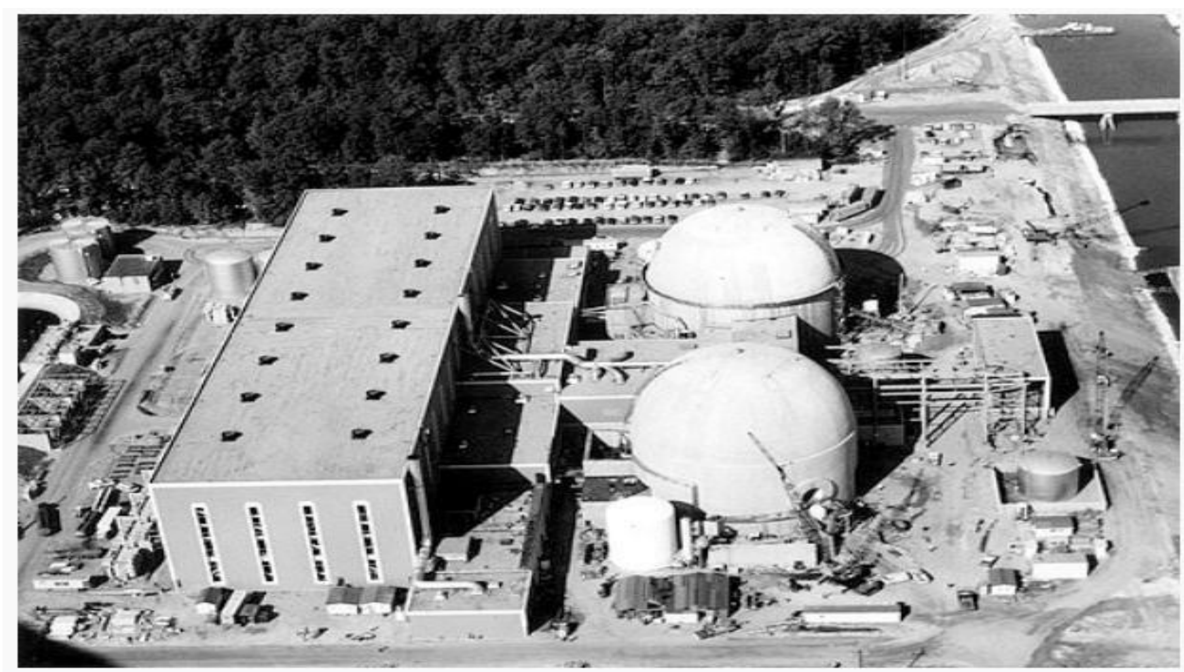


Figure B - 11. SURRY [98]

12. Brunswick, South Port, NC

This plant consisting of two General Electric boiling water reactors is named because of its location in the Brunswick County, in North Carolina. The site covers 1,200 acres (490 hectare) adjacent to the town of Southport and to wetlands and woodlands going commercial in 1975 [99]. The plant is cooled by the Cape Fear River for its intake and discharges into the Atlantic Ocean. The plant is primarily (81.7%) owned by the Duke Energy, and the remaining portion is owned by the North Carolina Eastern Municipal Power Agency. The 2010 U.S. census gave the population within 10 miles of the Brunswick site as 36,413, an increase of 105.3 percent in a decade, and the same census provided the population within 50 miles to be 468,953, an increase of 39.6 % since 2000. [99]

Cities within 50 miles include Wilmington which is 18 miles to the city center. The NRC's estimate of the risk each year of an earthquake intense enough to cause core damage at Brunswick was 1 in 66,667, in the study NRC published in August 2010.

Figure B - 12. BRUNSWICK [99]

13. Oyster Creek, Forked River, NJ

This plant is a single unit 636- MWe boiling water reactor power plant located on an 800-acre (320 hectare) site adjacent to the town of Oyster Creek in

Ocean County, New Jersey, and is owned and operated by Exelon [58]. This is the oldest operating commercial nuclear power plant in the US [100].when in December 1, 1969, became commercial. Although it is licensed to operate until April 9, 2029, the plant is scheduled to be permanently shut down by December 31, 2019. The plant discharges into the Atlantic Ocean, and as such, it is susceptible to a tsunami.

Figure B - 13. OYSTER CREEK [100]

14. Millstone, Waterford, CT

This plant is the only nuclear power plant in Connecticut which initially was designed to have three reactors, Unit 1, 2 and 3. After a Time magazine cover story on safety issues at Millstone, all three units were shut down, however, Units 2 and 3 were restarted and are operating at a combined output of 2020 MWe. Unit 1 did not restart, permanently ceasing operations in July 1998. Unit 2 and 3 are pressurized water reactor types, one made by Combustion Engineering, and the other by Westinghouse. The plant uses the Atlantic Ocean for cooling purposes and the site covers about 500 acres ($2km^2$) [101].

In 1999 Northeast Utilities, the plant's operator at the time, agreed to pay $10 million in fines for 25 counts of lying to federal investigators and for having falsified environmental reports. Its subsidiary, Northeast Nuclear Energy Company, paid an additional $5 million for having made 19 false statements to federal regulators regarding the promotion of unqualified plant operators between 1992 and 1996 [101]. Millstone Units 2 and 3, were sold to Dominion Resources by Northeast Utilities in 2000 and continue to operate [101]. On November 28, 2005, after a 22-month application and evaluation process, Millstone was granted a 20-year license extension for both Units 2 and 3 by the NRC [102].

Figure B - 14. MILLSTONE [102]

15. Seabrook, Seabrook, NH

The plant is located in Seabrook, New Hampshire, approximately 40 miles north of Boston and 10 miles south of Portsmouth [101] . Two pressurized water reactor reactors were initially planned, however, due to construction delays, cost overruns and troubles obtaining financing, the second reactor was never completed [101]. In 1976 the construction permit was granted and was completed in 1986 with full power operation in 1990. It provides 1,244 MWe electrical output, being the largest individual electrical nuclear power plant in the New England power grid, and is the second largest nuclear power plant in New England [101].

The construction was completed ten years later than expected, with a cost approaching $7 billion. The large debt resulted in the bankruptcy of Seabrook's major owner, Public Service Company of New Hampshire The plant was originally owned by more than 10 utility companies serving five New England states [101]. In 2002, most of the utilities sold their shares to Florida Power and Light Energy.

Figure B - 15. SEABROOK [101]

APPENDIX C: THE RESEARCHER'S CURRICULUM VITAE

The researcher with 27 years of commercial nuclear power experience is expressing judgment in this dissertation. The judgement expressed herein, is based on his experience and expertise in the nuclear engineering discipline. The experience includes, certification as a nuclear power plant engineer at the (QCNPS)[20], the SRO[21] license holder at QCNPS (an identical plant as the Fukushima Daiichi), a plant design engineer (at Bechtel Power), a mid level section manager at GE Nuclear for a decade, and a federal reactor inspector (at U.S. NRC with detailed knowledge of the core and fuel design.

The researcher traveled to all boiling water reactors (BWR) in the US and international sites except for the Japanese BWR plants. Critical training courses such as the degraded core conditions, transient analysis, and emergency procedure guideline were essential for the licensees staff and engineers. In the degraded core condition training course that the researcher developed for the utilities, students were informed of the consequences of lack of hydrogen control, use of the emergency procedure guideline procedures, venting of the containment, instrumentation response, and other topics that would evolve in degraded core conditions. As part of selling a plant, GE offered BWR buyers to receive numerous credits for training at the San Jose facilities (GE Nuclear's headquarters). It was imperative for GE to have their customers treat their

[20] QCNPS is the name of the Quad Cities Nuclear Power Station, located in Cordova, Illinois. It is a BWR/3 MK I containment, where the researcher obtained his SRO license.

[21] There are two types of operating license for a nuclear power plant. One type is Reactor Operator (RO) license and the other is Senior Reactor Operator (SRO) license. To supervise operators, one has to have an SRO license and to operate a reactor,one has to have an RO license. It is much more difficult and encompassing to acquire an SRO license than the RO license. Each license is valid for two years. After that, the applicants must requalify to maintain their license.

warrantied nuclear fuel with adequate knowledge and information. There would have been no registration fee for these credited courses. A typical registration fee for an on-site five-week course was around $75,000 when the researcher was managing GE's nuclear engineering training program. A typical fee for a single attendee for the same five-week course in San Jose was about $14,000. This did not include the cost of travel to and from a utility site site to San Jose. It can be argued that the staff at Fukushima received their training from the Hitachi or Toshiba Corporations. It is certain that Hitachi or Toshiba did not and could not have access to the BWR owners group and the generation of the Emergency Procedures Guideline training prior to June of 2007. It is uncertain if Hitachi/Toshiba developed and delivered Degraded Core Coditions training course to TEPCO personnel.

a. The researcher worked at a commercial nuclear power plant in the US for three years, where he was the responsible nuclear engineer for the Unit 1 reactor of a large BWR station that is identical to the Fukushima station. During this period, he received an SRO license issued by the NRC at that facility. Prior to receiving the license, he certified as an SRO at the Dresden Nuclear Power Station using the only available nuclear power plant simulator in the US at that time. This license was issued by the GE. It was equivalent to the NRC license, except that it was certified by GE rather than the NRC.

b. The researcher worked for Bechtel Power as a Nuclear Engineering Group Supervisor for a year, designing a large BWR at Taiwan's Kuosheng plant. He was responsible for maintaining the Q-list and approve the design features.

He examined the licensing issues, using the NRC guidelines and industry codes and standards. Different engineering disciplines needed their design features approved by the nuclear engineering group, as this group was responsible for all features of the plant to meet the appropriate regulatory requirements, codes, and standards. Other groups included electrical, mechanical, civil-structural, instrumentation, and architectural features that were designed by various disciplines within Bechtel's design group.

c. The researcher was a mid-level manager at GE Nuclear for a decade, teaching advanced nuclear engineering courses in the Engineering Training and Technology Transfer organization. He was a subsection manager of 60 engineers with unit managers as his direct reports. He was instrumental in the development and delivery of teaching courses and related services for managers, engineers, technicians and operators in the transient analysis course, GE's emergency procedure guidelines, degraded core conditions, and probabilistic risk assessment. He delivered numerous courses to GE's customers in the US, Mexico, Europe, and Taiwan. He co-authored and approved the publication of the three-volume Station Nuclear Engineering manual for all BWRs. A copy of this document was maintained at the NRC's Emergency Response Center in Rockville, Maryland. He developed and delivered numerous times the degraded core conditions and transient analysis courses, which were developed as a consequence of the Three Mile Island accident. The transient analysis and degraded core conditions courses were essential requirements for someone conducting this research. Evaluating transients and events that form the basis of accidents, requires deep understanding of the events to be analyzed in this research. Even though the Fukushima Daiichi nuclear power station was manufactured by GE, the Japanese staff did not attend GE's courses; however, the Taiwanese, Mexicans, and European staff attended. Japanese personnel working for TEPCO developed their own training programs and courses.

d. In addition to managing the training and related programs, the researcher periodically traveled to GE's training simulator (Illinois) and conducted certification programs for utility and GE engineers and operators. This certification included creating and conducting written exams for half a day, plant walk-through for half a day, and the plant simulator exam for another half a day. For the simulator certification exam, the researcher created various transients on the control room simulator and monitored the candidates' responses manipulating the controls and switches to prevent a nuclear accident. This was essential in determining the students' ability to control the systems and components to certify as a reactor operator or as a senior reactor operator.

e. Next, the researcher joined the NRC for a period of seven years as a reactor inspector and a project manager. He conducted many inspections both as a single inspector, or as a member of a team inspection, at BWRs and PWR facilities in the US. As a staff member of the NRC, he was attached to Region III, whose mission was to regulate the nuclear power program responsible for the midwestern parts of the US. As an inspector, he inspected GE and Westinghouse fuel fabrication facilities. Because the researcher's specialty was fuel design, he was among a handful of NRC staff qualified to conduct these inspections. All inspections, whether at the plants or at fuel fabrication facilities, included a review of documents in advance of the site trips, site inspection for one week, office evaluation of the collected data for one week, and site inspection for another week. The inspections always concluded with an exit meeting, where senior members of the licensee attended and heard the results of the inspection findings. In earlier years, these meetings were open to the public, and the end results of the inspections were published in an inspection report. All inspections conducted by the NRC include publication of the inspection report. Except for the result of security inspections, all reports are available for public information without pursuing the Freedom of Information Act (FOIA) process. Invariably, the inspection reports include the positive and negative aspects of the inspection findings, including the Notice of Violation(s) (NOV).

f. After leaving the NRC, the researcher joined Exelon for a year as the Engineering Assurance supervisor of ComEd's (later changing to Exelon) six nuclear power plants (Dresden, Quad Cities, La Salle County, Byron, Braidwood, and Zion; the first three are BWRs and the last three are PWRs). In this position, the researcher was responsible for coordinating engineering activities and engineering changes to these plants to ascertain uniformity and consistency. The NRC had faulted Exelon who owned and operated these six nuclear power plants for inconsistency in implementing design changes for these plants. Often, one improvement was not implemented at another plant, and there were little interfaces. Therefore, in response, Exelon created an engineering assurance organization in their headquarters, and the researcher's responsibilities included the creation of effective communication among the plant managers regarding the engineering activities of the six plants. Also included, was the generation of monthly reports covering events at all six plants and progress in engineering activities, for the benefit of Exelon's senior executives. Pareto charts for each plant reflected the trend of progress in the engineering activities, and were of paramount importance to their senior executives. The reports formed the basis for their decisions concerning these plants in a manner that, if there were an improvement in one plant, then it would be considered for implementation in the remaining plants. This, of course, included applicability of the targeted plants. Some changes in a BWR cannot be incorporated in a PWR, or vice versa.

g. The researcher received his MS in Nuclear Engineering from the University of Illinois in 1971, an MBA from the Pepperdine University in1983 and a Professional Engineers license in the State of California in the nuclear engineering discipline. The researcher is a PhD candidate at the University of North Carolina-Charlotte.

APPENDIX D: SYTEM COMPONENTS AND FAILURE MODE

This list was developed at the outset of the research identifying various systems and their components. The failure mode was stipulated for each component. In case of a failure of the component or a system, the backup system was also identified. This type of information is a required knowledge for a person holding a supervisory license to operate a commercial nuclear power plant, in this case, an identical plant to the Fukushima Daiichi nuclear power station. This list was generated by the researcher.

	SYSTEM	FAILURE TYPE	COMPONENTS	CONSEQUENCE	IMPLICATIONS	
1	HPCI	MECHANICAL	VALVES	FAIL TO INJECT	RCIC/OTHER BACKUP	YES
2	HPCI	ELECTRICAL	MOTOR	FAIL TO INJECT	RCIC/OTHER BACKUP	YES
3	HPCI	MECHANICAL	PUMP	FAIL TO INJECT	RCIC/OTHER BACKUP	YES
4	HPCI	MECHANICAL	PIPING	FAIL TO INJECT	RCIC/OTHER BACKUP	YES
5	HPCI	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO INJECT	RCIC/OTHER BACKUP	YES
6	HPCS	MECHANICAL	VALVES	FAIL TO INJECT	HPCS/RCIC /OTHER	YES
7	HPCS	ELECTRICAL	MOTOR	FAIL TO INJECT	HPCS/RCIC /OTHER	YES
8	HPCS	MECHANICAL	PUMP	FAIL TO INJECT	HPCS/RCIC /OTHER	YES
9	HPCS	MECHANICAL	PIPING	FAIL TO INJECT	HPCS/RCIC /OTHER	YES
10	HPCS	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO INJECT	HPCS/RCIC /OTHER	YES
11	LPCI	MECHANICAL	VALVES	FAIL TO INJECT	RHR/LPCI/OTHER LOOP	YES
12	LPCI	ELECTRICAL	MOTOR	FAIL TO INJECT	RHR/LPCI/OTHER LOOP	YES
13	LPCI	MECHANICAL	PUMP	FAIL TO INJECT	RHR/LPCI/OTHER LOOP	YES
14	LPCI	MECHANICAL	PIPING	FAIL TO INJECT	RHR/LPCI/OTHER LOOP	YES
15	LPCI	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO INJECT	RHR/LPCI/OTHER LOOP	YES
16	RHR	MECHANICAL	VALVES	FAIL TO INJECT	LPCI/OTHER LOOP	YES
17	RHR	ELECTRICAL	MOTOR	FAIL TO INJECT	LPCI/OTHER LOOP	YES
18	RHR	MECHANICAL	PUMP	FAIL TO INJECT	LPCI/OTHER LOOP	YES

19	RHR	MECHANICAL	PIPING	FAIL TO INJECT	LPCI/OTHER LOOP	YES
20	RHR	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO INJECT	LPCI/OTHER LOOP	YES
21	CONT.SP	MECHANICAL	VALVES	FAIL TO SPRAY	OTHER LOOP	YES
22	CONT.SP	ELECTRICAL	MOTOR	FAIL TO SPRAY	OTHER LOOP	YES
23	CONT.SP	MECHANICAL	PUMP	FAIL TO SPRAY	OTHER LOOP	YES
24	CONT.SP	MECHANICAL	PIPING	FAIL TO SPRAY	OTHER LOOP	YES
25	CONT.SP	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO SPRAY	OTHER LOOP	YES
26	ABD	MECHANICAL	VALVES	NO REDUCE PRESS	HPCS, SAFETY/RELIEF VALVES	YES
27	ABD	ELECTRICAL	MOTOR	NO REDUCE PRESS	HPCS, SAFETY/RELIEF VALVES	YES
28	ABD	MECHANICAL	PUMP	NO REDUCE PRESS	HPCS, SAFETY/RELIEF VALVES	YES
29	ABD	MECHANICAL	PIPING	NO REDUCE PRESS	HPCS, SAFETY/RELIEF VALVES	YES
30	ABD	ELECTRICAL	CABLES AND INSTRUMENTATION	NO REDUCE PRESS	HPCS, SAFETY/RELIEF VALVES	YES
31	RCIC	MECHANICAL	VALVES	FAIL TO INJECT	HPCS/HPCI	YES
32	RCIC	ELECTRICAL	MOTOR	FAIL TO INJECT	HPCS/HPCI	YES
33	RCIC	MECHANICAL	PUMP	FAIL TO INJECT	HPCS/HPCI	YES
34	RCIC	MECHANICAL	PIPING	FAIL TO INJECT	HPCS/HPCI	YES
35	RCIC	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO INJECT	HPCS/HPCI	YES
36	IC	MECHANICAL	VALVES	FAIL TO INJECT	LPCI	YES
37	IC	ELECTRICAL	N/A	N/A	LPCI	YES
38	IC	MECHANICAL	N/A	N/A	LPCI	YES
39	IC	MECHANICAL	PIPING	FAIL TO INJECT	LPCI	YES
40	IC	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO INJECT	LPCI	YES
41	DG	MECHANICAL	VALVES	FAIL TO FUNCTION	OTHER D/G	YES
42	DG	ELECTRICAL	MOTOR	FAIL TO FUNCTION	OTHER D/G	YES
43	DG	MECHANICAL	N/A	N/A	OTHER D/G	YES
44	DG	MECHANICAL	N/A	N/A	OTHER D/G	YES
45	DG	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO FUNCTION	OTHER D/G	YES
46	SBLC	MECHANICAL	VALVES	FAIL TO INJECT	IT IS A BACKUP TO CRD	YES
47	SBLC	ELECTRICAL	MOTOR	FAIL TO INJECT	IT IS A BACKUP TO CRD	YES

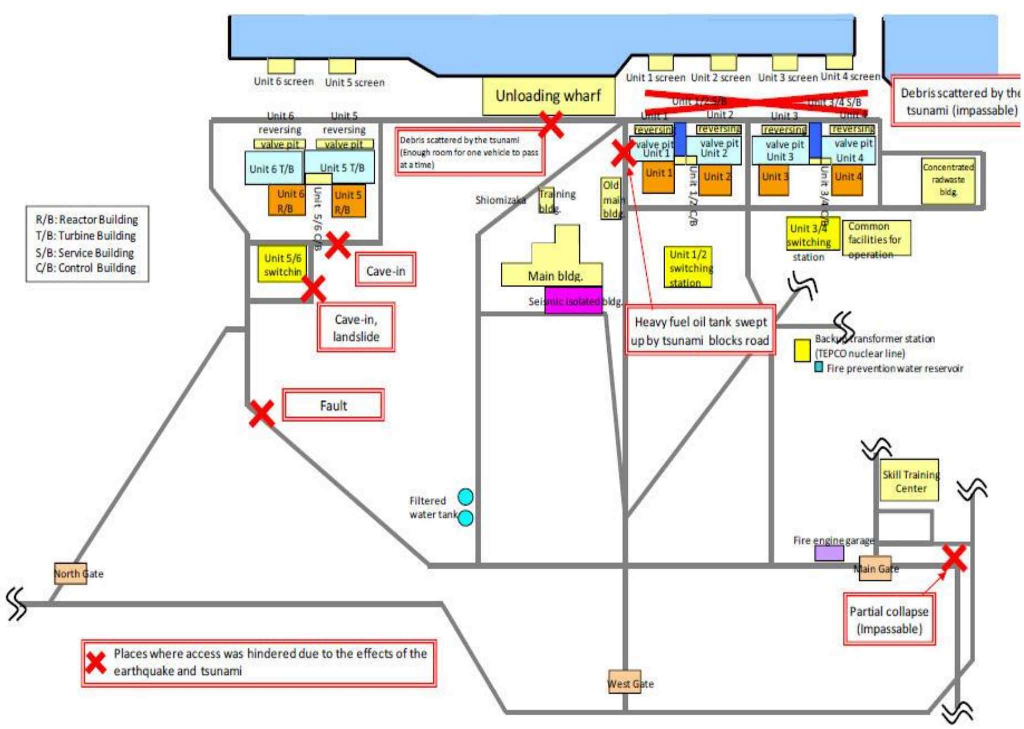
48	SBLC	MECHANICAL	PUMP	FAIL TO INJECT	IT IS A BACKUP TO CRD	YES
49	SBLC	MECHANICAL	PIPING	FAIL TO INJECT	IT IS A BACKUP TO CRD	YES
50	SBLC	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO INSERT	IT IS A BACKUP TO CRD	YES
51	CRD	MECHANICAL	VALVES	FAIL TO INSERT	SBLC	YES
52	CRD	ELECTRICAL	N/A	N/A	SBLC	YES
53	CRD	MECHANICAL	N/A	N/A	SBLC	YES
54	CRD	MECHANICAL	N/A	N/A	SBLC	YES
55	CRD	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO INSERT	SBLC	YES
56	PR.SUPP.	MECHANICAL	VALVES	FAIL TO INJECT	RIVER, LAKE OCEAN	YES
57	PR.SUPP.	ELECTRICAL	N/A	N/A	RIVER, LAKE OCEAN	YES
58	PR.SUPP.	MECHANICAL	N/A	N/A	RIVER, LAKE OCEAN	YES
59	PR.SUPP.	MECHANICAL	PIPING	FAIL TO INJECT	RIVER, LAKE OCEAN	YES
60	PR.SUPP.	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO INJECT	RIVER, LAKE OCEAN	YES
61	CONTAIN	MECHANICAL	VALVES	FAIL TO CONTAIN	SECONDARY	YES
62	CONTAIN	ELECTRICAL	MOTOR	FAIL TO CONTAIN	SECONDARY	YES
63	CONTAIN	MECHANICAL	PUMP	FAIL TO CONTAIN	SECONDARY	YES
64	CONTAIN	MECHANICAL	PIPING	FAIL TO CONTAIN	SECONDARY	YES
65	CONTAIN	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO CONTAIN	SECONDARY	YES
66	HYDROGE N	MECHANICAL	VALVES	NOT COMBINE	H2 EXPLOSION	YES
67	HYDROGE N	ELECTRICAL	MOTOR	NOT COMBINE	H2 EXPLOSION	YES
68	HYDROGE N	MECHANICAL	PUMP	NOT COMBINE	H2 EXPLOSION	YES
69	HYDROGE N	MECHANICAL	PIPING	NOT COMBINE	H2 EXPLOSION	YES
70	HYDROGE N	ELECTRICAL	CABLES AND INSTRUMENTATION	NOT COMBINE	H2 EXPLOSION	YES
71	SEC.CONT	MECHANICAL	NONE	FAIL TO CONTAIN	NONE	YES
72	SEC.CONT	ELECTRICAL	NONE	FAIL TO CONTAIN	NONE	YES
73	SEC.CONT	MECHANICAL	NONE	FAIL TO CONTAIN	NONE	YES
74	SEC.CONT	MECHANICAL	TUBING	FAIL TO CONTAIN	NONE	YES

75	SEC.CONT	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO CONTAIN	NONE	YES
76	RPV	MECHANICAL	VALVES	INTEGRITY	NONE	YES
77	RPV	ELECTRICAL	N/A	N/A	NONE	YES
78	RPV	MECHANICAL	N/A	INTEGRITY	NONE	YES
79	RPV	MECHANICAL	N/A	INTEGRITY	NONE	YES
80	RPV	ELECTRICAL	CABLES AND INSTRUMENTATION	INTEGRITY	NONE	YES
81	SBGT	MECHANICAL	VALVES	FAIL TO RELEASE	NONE	YES
82	SBGT	ELECTRICAL	MOTOR	FAIL TO RELEASE	NONE	YES
83	SBGT	MECHANICAL	PUMP	FAIL TO RELEASE	NONE	YES
84	SBGT	MECHANICAL	PIPING	FAIL TO RELEASE	NONE	YES
85	SBGT	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO RELEASE	NONE	YES
86	INSTR.	MECHANICAL	VALVES	FAIL TO CONTROL	BACKUP	YES
87	INSTR.	ELECTRICAL	N/A	FAIL TO CONTROL	BACKUP	YES
88	INSTR.	MECHANICAL	N/A	FAIL TO CONTROL	BACKUP	YES
89	INSTR.	MECHANICAL	PIPING	FAIL TO CONTROL	BACKUP	YES
90	INSTR.	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO CONTROL	BACKUP	YES
91	RPS	MECHANICAL	N/A	FAIL TO SHUTDN	NONE	YES
92	RPS	ELECTRICAL	N/A	FAIL TO SHUTDN	NONE	YES
93	RPS	MECHANICAL	N/A	FAIL TO SHUTDN	NONE	YES
94	RPS	MECHANICAL	N/A	FAIL TO SHUTDN	NONE	YES
95	RPS	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO SHUTDN	NONE	YES
96	CST	MECHANICAL	VALVES	NO H2O SUPPLY	SUPPRESSION POOL	YES
97	CST	ELECTRICAL	MOTOR	NO H2O SUPPLY	SUPPRESSION POOL	YES
98	CST	MECHANICAL	PUMP	NO H2O SUPPLY	SUPPRESSION POOL	YES
99	CST	MECHANICAL	PIPING	NO H2O SUPPLY	SUPPRESSION POOL	YES
100	CST	ELECTRICAL	CABLES AND INSTRUMENTATION	NO H2O SUPPLY	SUPPRESSION POOL	YES
101	FUEL POOL	MECHANICAL	VALVES	FAIL TO COOL FUEL	NONE	YES
10	FUEL	ELECTRICAL	MOTOR	FAIL TO COOL	NONE	YES

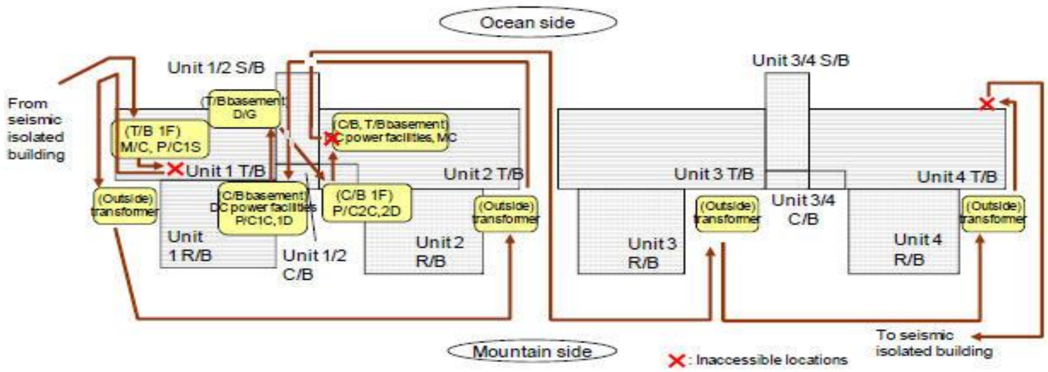
2	POOL			FUEL		
10 3	FUEL POOL	MECHANICAL	PUMP	FAIL TO COOL FUEL	NONE	YES
10 4	FUEL POOL	MECHANICAL	PIPING	FAIL TO COOL FUEL	NONE	YES
10 5	FUEL POOL	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO COOL FUEL	NONE	YES
10 6	FUEL	MECHANICAL	VALVES	FAIL TO GET COOLED	NONE	YES
10 7	FUEL	ELECTRICAL	N/A	N/A	NONE	YES
10 8	FUEL	MECHANICAL	N/A	N/A	NONE	YES
10 9	FUEL	MECHANICAL	N/A	N/A	NONE	YES
11 0	FUEL	ELECTRICAL	CABLES AND INSTRUMENTATION	FAIL TO GET COOLED	NONE	YES

APPENDIX E: IMAGES OF THE FUKUSHIMA DAIICHI PLANT

All the images in this Appendix are from various TEPCO documents provided to the researcher for the purpose of conducting the research. The list of these files appear in Figure 11 (Page 43) of this dissertation. The owner of these images and the corresponding documents is Tokyo Electric Power Company. The researcher is grateful to TEPCO for their generosity. Because these images come from different documents, they have different numbering in the caption of the images. The researcher did not alter the number and captured the title of the images as is. They are not cited either as they are from different sources and they are for information for the readers.



Layout of the Fukushima Daiichi Nuclear Power Station



Overview of Inspection

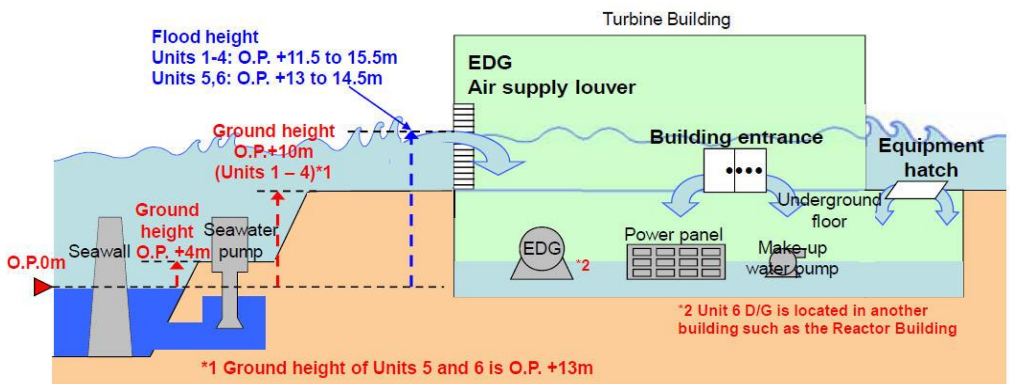


Figure 4: Path of inundation of major buildings

Tsunami Spray of approximately 50m

Unit 4 damage (on right side of photo)
(Unit 2 is on left side of photo)

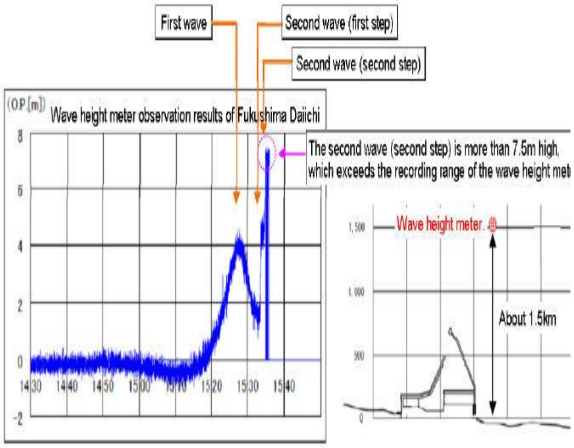
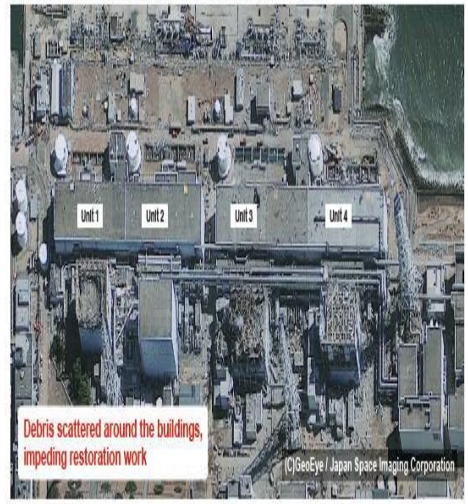


Figure Wave height meter observation results and the positional relation between the Fukushima Daiichi Nuclear Power Station site and the wave height meter

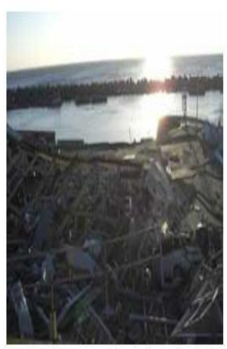
The damage at Fukushima Daiichi Nuclear Power Station Units 1-4, photographed March 19, 2011



Tsunami destroys 10m high seawall and pushes inland

Units 1 to 4 flooded by tsunami

Tank deformed by the tsunami (Same tank shown in picture above)



Conditions on sea side after tsunami arrival

17 (08:10)

18 (08:20)

19 (08:38)

Attachment Earthquake-tsunami-1-9

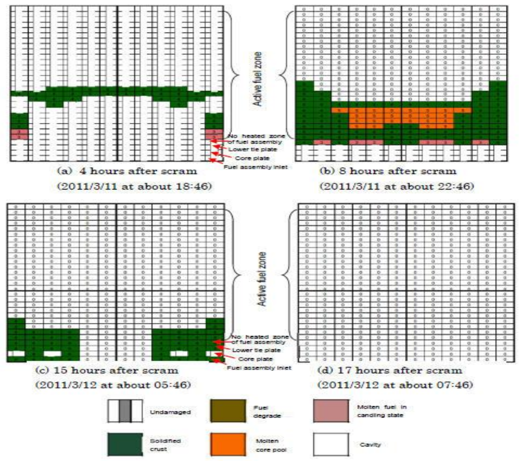


Figure 4.1 Conditions of damaged core (Unit-1)

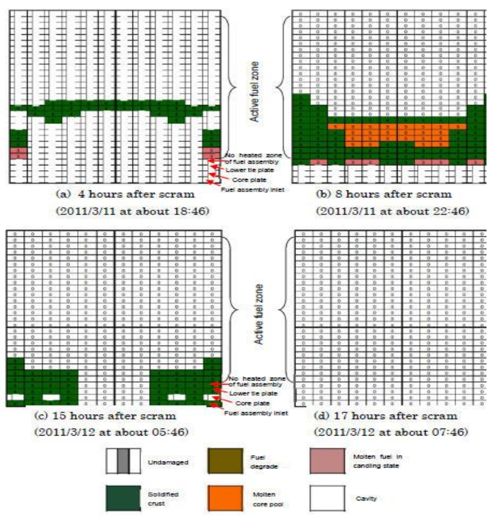


Figure 4.1 Conditions of damaged core (Unit-1)

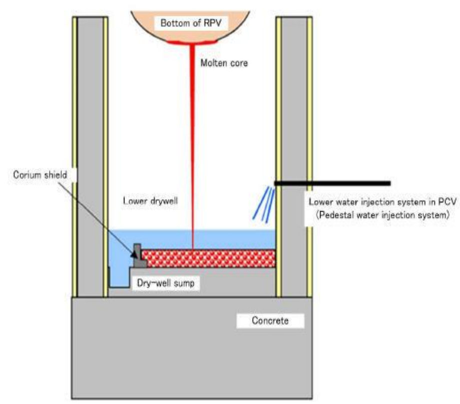


Figure 10 Schematic drawing of the structure to prevent molten fuel from flowing into the dry-well sump

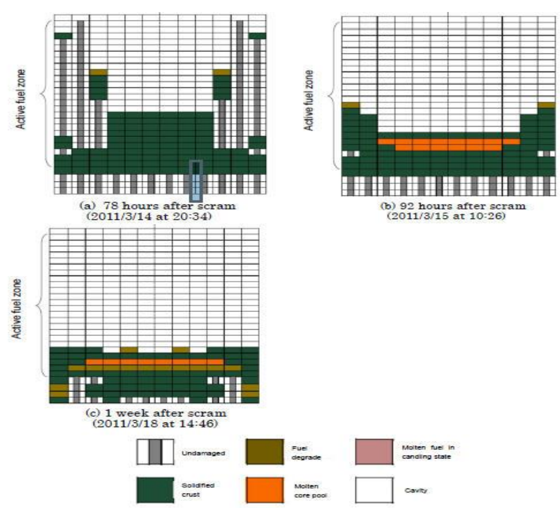


Figure 4.2 Conditions of damaged core (Unit-2)