Generic Level 2 Probabilistic Safety Analysis on Pressurized Water Reactors

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Acknowledgement

After six-month struggles with the challenges faced in the research, finally I have reached the last stop of my thesis journey. At this special moment, I would like to express my deep gratitude to everyone who has helped me during this unforgettable journey.

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Executive Summary

Research Background

In 1954, the first nuclear power station was put into operation for electricity generation. Since then, nuclear power plants have been widely applied around the world for electricity supply to the power grid. As a clean and sustainable energy source, nuclear energy has non-carbon dioxide emissions during energy production, which is perceived as an effective way to slow down global warming. Moreover, the nuclear fuel source for fission reactions, uranium, is largely stored in the earth. It can guarantee the energy supply security for the long-term run. Although nuclear energy could bring a lot of benefits to society, the public still holds back on the large-scale implementation of nuclear plants out of safety concerns.

In general, the electricity production process in nuclear power plants can be divided into two stages: 1. Capturing the heat from fission reactions to produce the steam. 2. Using the produced steam to power the turbine and generate electricity. However, some fission products are very toxic and unstable, once they got released to the environment, the accident consequence is unbearable. For example, the environmental damage and health impact caused by nuclear accidents such as Chernobyl and Fukushima Daiichi are unaccountable.

In order to prevent these nuclear accidents occurring, probabilistic safety analysis (PSA) was required by most power plants for their safety assessment. Among three different levels of PSA, level 2 PSA is designed to investigate the containment response during the severe accidents. Since 1975, WASH-1400 has introduced the level 2 PSA with very large containment event tree to represent different release paths. Later on, some improvements have been made to shrink the size of the whole event tree but to add multiple small event trees below the sub-events for detailed investigation. Also, currently the level 2 PSA studies become more and more detailed and plant-specific, which made the results obtained from these studies very difficult for other reactors to study. Although some plants did share the similar design and systems, each individual reactor still has their own complex level 2 PSA analysis, which made it difficult to figure out what safety functions should be improved in general for that same type of reactors. Therefore, a generic level 2 PSA is needed in order to provide a general overview of potential accidents and their consequences for certain design type of reactors. It can function as the reference for the precursor analysis and it also can be tailored based on specific plant requirement for detailed risk assessment.

Methodology

The main methodology for generic level 2 PSA development follows the general rules of level 2 PSA establishment. These steps are plant familiarization, plant damage state identification, accident progression analysis, release category identification, and source term analysis. They are recommended by the International Atomic Energy Agency (IAEA). However, different from other level 2 PSA studies, in this research, a generic containment event tree was constructed through the detailed analysis in previous steps. By using this generic containment event tree model, the release paths were obtained and the validity of release consequences got checked through the comparison with other level 2 PSA results.

Before performing level 2 PSA, pressurized water reactor was selected as the reference reactor type because it is the dominant reactor type in the world so that the research results can benefit more operating plants. In plant familiarization step, different generations of pressurized water reactors and their containment safety systems were classified and summarized in order to get a common design features of pressurized water reactor. During the plant damage state identification, key attributes and initiating events for severe accidents are spotted so that the further analysis of containment response can be performed. For the accident progression analysis, a generic containment event tree was built based on the grouped plant damage states. By taking containment safety systems' response into consideration, different release paths were created, which could be used for release consequence calculation. In the release category identification, different release categories were proposed, including early containment failure, late containment failure, isolation failure, containment bypass, and containment intact. Based on the characteristics of containment event tree end states, those end states were further grouped into these categories in order to estimate the release consequences. The last step is source term analysis, in which fission inventory fractions for each release category are calculated based on the data calibrated from TMI Unit 1 level 2 PSA. The release frequencies were also calculated according to the generic containment event tree.

In the end, the release frequency got compared with the results from WASH-1400, French 1300Mwe level 2 PSA, and TMI Unit 1 level 2 PSA. The results showed the frequency obtained from the generic model well complied with others.

Final Deliverables

All in all, two main deliverables are provided by this research, which can be viewed as below:

A conceptual review for pressurized water reactors and their containment safety systems.

In this research, different generations of pressurized water reactors have been discussed and the general operation mechanism of pressurized water reactors has been analyzed. Besides, containment safety systems of pressurized water reactors have been categorized based on their different functionalities. The classified containment safety systems are the system for temperature and pressure control, the hydrogen mitigation system as well as radioactive release control system. Based on the above classification, the implementation situation of these systems has been discussed by comparing different pressurized water reactors throughout the world. It was found most of the European pressurized water reactors have been back-fitted with hydrogen mitigation and radioactive control systems. Also, most of the advanced pressurized water reactors like generation III+ types were also equipped. In the United States, only reactors with ice condensers have implemented hydrogen mitigation systems and the implementation of radioactive control systems is still under development.

A generic level 2 PSA for pressurized water reactors

A generic level 2 PSA has been performed in the research. By following pressurize water reactor operation and safety systems analysis, the plant damage state logic diagram was developed, which was used to select the severe accidents for containment response analysis. In the accident progression analysis, a generic containment event tree has been developed based on those severe accident states identified through the plant damage state analysis. Therefore, in the generic containment event tree, all potential release paths are already presumed to have core damage before the containment response. By using the generic containment event tree, the accident progression under different release scenarios was studied. In the research, 22 containment end-state events were spotted, which were further grouped into 5 release categories for release consequence analysis and the estimated release frequencies were cross-compared with other level 2 PSA studies to see the approximation of the analysis.

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Chapter 1 Introduction

In order to better mitigate climate change, a transition towards clean energy supply has been gradually developed in recent years. As one of the most efficient and carbon-free emission energy source, nuclear energy has been widely applied for electricity production all over the world because it can not only reduce carbon dioxide emission but also improve energy supply security. At the end of 2016, there are 448 units of nuclear power plants in operation with total amount of 2476.2 TWh electricity supply to the world energy market (IAEA, 2017). By using nuclear energy instead of coal to generate this equivalent amount of electricity, it may help to reduce the carbon dioxide emissions by 1 million tons. For OECD countries, according to the latest data published by Nuclear Energy Agency (NEA) in 2018, the nuclear electricity generation took up 17.9 % of their total electricity generation (NEA, 2018). It is also estimated by IAEA that the nuclear electricity production will continuously increase in East Europe and Asia pacific regions in its both high and low projections (IAEA, 2017). Therefore, nuclear power production not only plays an important role in energy transition but also impacts a lot on people's daily life.

The general electricity production process for most nuclear power plants is through a well-controlled heavy element fission reaction. Nuclear fission reaction is the process that heavy element atom nuclei collides with neutrons and releases the heat and neutrons. By capturing the heat released from reaction, steam can be produced to power corresponding turbines in nuclear power plants in order to continuously generate electricity (NEA, 2012). In the reaction, uranium is perceived as the most widely used fission fuel and Uranium largely exists in the earth crust. Therefore, nuclear technology for power production was viewed as one of the most sustainable energy production method in the future. However, one of the biggest downsides of this technology is nuclear safety issues. The fission products are very unstable. Once they are released outside the containment, it will result in a huge health and environmental impact. In general, the most critical fission products are iodine, caesium, cromine, kryptom, etc. Therefore, when it comes to the commercial application of electricity generation through nuclear plants, the associated risk should be well considered.

1.1 Nuclear Safety & Societal Challenges

Nuclear safety issues are always perceived as a top priority during the operation. Once the radioactive substances got released, the damage to the environment and human beings is unbearable and its impacts always last for decades. Normally when radioactive substances released to the environment, there were severe accidents happened within the plants. Although the occurrence probability of severe accidents is very low, there are still several severe accidents that remind people of the importance of nuclear safety. These severe accidents include Three Mile Island (TMI) accident in 1979, Chernobyl accident in 1986 and Fukushima Daiichi accident in 2011(Gharari, et al., 2018). TMI accident happened in March of 1979, which resulted in hydrogen combustion within the containment and caused around 8% of hydrogen and noble gases escaped to the environment (Croucher, 1981). Later on, the U.S. nuclear regulatory commission (USNRC) revised its regulations on hydrogen control in order to prevent hydrogen explosions in severe accidents (Gharari, et al., 2018). After the TMI accident, Chernobyl is another well-known catastrophe in which a large amount of radioactive substances were released to the environment. Around 4760 Km^2 nearby areas were evacuated and detrimental effects are recorded in wildlife (Beresford, Scott, & Copplestone, 2019). After that catastrophe, a lot of improvements in safety systems of nuclear plants have been made and these improvements made the failure rate for accidents with loss over 20M USD drop to 0.003 (Wheatley, Sovacool, & Sornette, 2017).

However, there is still a lot of social anxiety towards nuclear safety that provoked continued social conflicts, which greatly impacted the economic development and societal progress. The accident happened to Fukushima nuclear plant has again brought a lot of social panic on nuclear technology (Wheatley, Sovacool, & Sornette, 2016). After the accident, over 7 years have passed but still around 60000 evacuated residents live in temporary houses and they are not allowed to go back to their original hometown (Suzuki, 2019). Nuclear safety is closely related to people’s daily life and its impact largely influences societal development and surrounding environments. Beyond those technology improvements, a global governance of nuclear risk management is also needed because different countries have different capabilities of nuclear risk management (Taebi & Mayer, 2017). Taebi (2012) also pointed out nuclear risk is not only related to the present generation but also future generations and the emotions of the public need to be well considered. Therefore, a continuously improved risk analysis and corresponding prevention measures on nuclear plants should be well performed in order to ensure nuclear safety and sustainable development of nuclear technology.

The figure 1 below shows the general overview of risk governance of nuclear risks including industrial risk management and social risk management. However, if nuclear accidents could be well controlled through industrial risk management, the induced social risk can be further eliminated. Therefore, in this research, the main focus is on how to improve current industrial risk analysis in order to better reduce the probability of potential accident occurrence and mitigate its consequences.

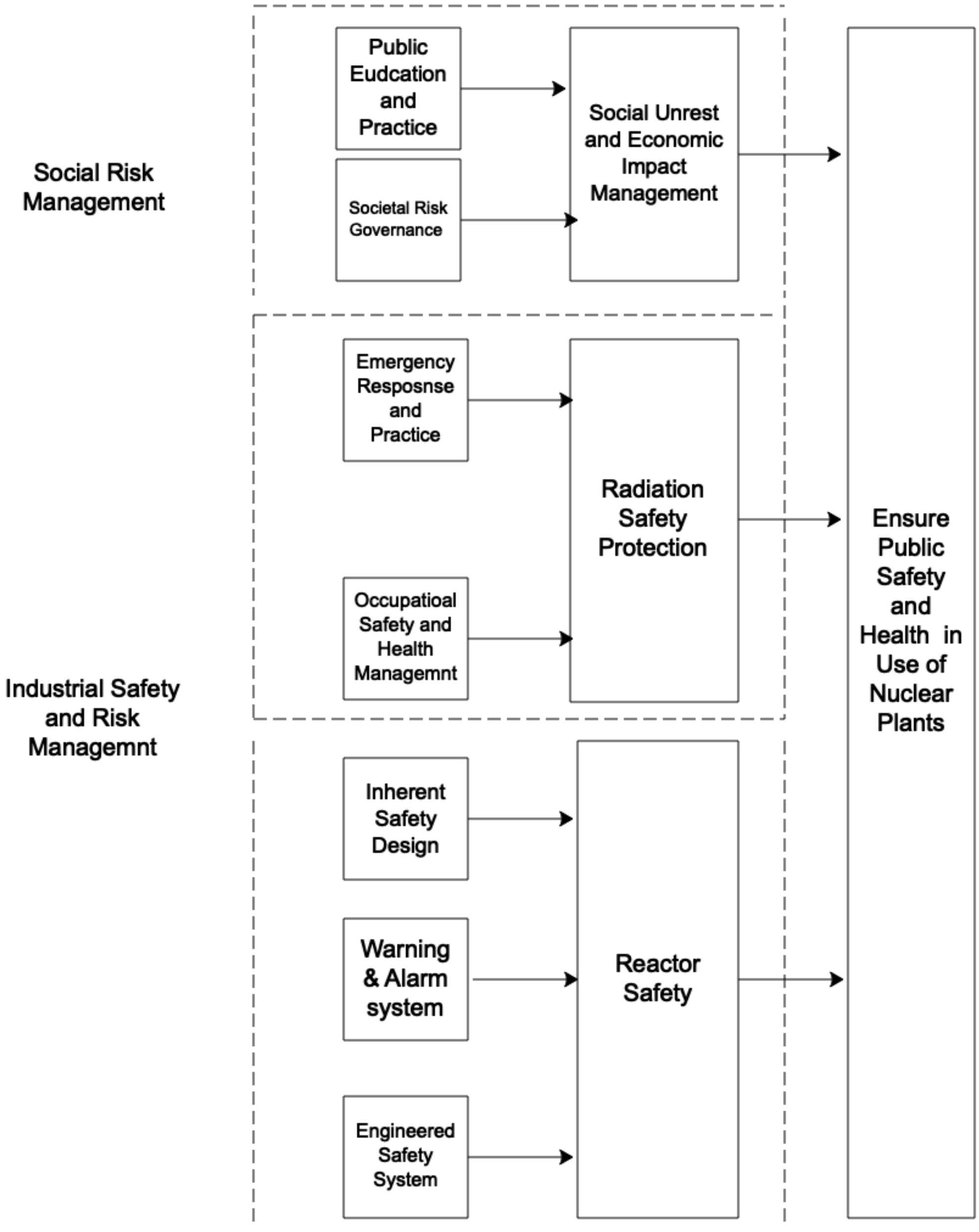

Figure 1. The general overview of risk management required in nuclear plants

1.2 Nuclear Safety Management

As mentioned in previous sections, in order to eliminate the potential social risks that may arise following the nuclear plant accidents, good industrial safety and risk management is strongly needed. For nuclear safety management, two strategies are widely applied to prevent radioactive release. The first one is to provide multiple leak-tight barriers between the public and the radioactive substances, which includes the fuel cladding, the primary system, and the containment. The second one is called the defense-in-depth strategy, which should be applied both in design period and operation period in order to provide good protection against various accident scenarios (NEA, 1992). According to IAEA, the objective of defense-in-depth strategy can be summarized as below: to mitigate the impact caused by component failure or human errors; to keep the safety barriers and systems effective in order to timely stop the accidents; and to prevent the potential accident from harming the public once the safety system failed (IAEA, 1996). It means the first priority is to prevent the accident from happening. But if the prevention measures failed, the mitigation measures should be available to control the accident progression and reduce the accident consequence level to as low as acceptable by the public. In general, the defense-in-depth strategy consists of five layers of prevention and mitigation approaches. Once the previous level failed, the following one will function to protect against potential accidents. The five levels of defense-in-depth concept were summarized as below:

1. First level- to prevent abnormal operation and system failure.
2. Second level- to detect the failure and control the abnormal operation.
3. Third level- to ensure safety functions and activate the specific safety systems and features.
4. Forth level- to mitigate the accident progression and mitigate severe accident consequence.
5. Fifth level- to mitigate the radiological consequence.

Figure 2 illustrates in what way safety systems and risk management measures get involved in these different levels of the defense-in-depth strategy. From the inner layer to the outer layer, the accident consequence mitigation effectiveness degraded. The optimal solution is to have inherent safety design, which means to design highly reliable systems in order to avoid potential loss.

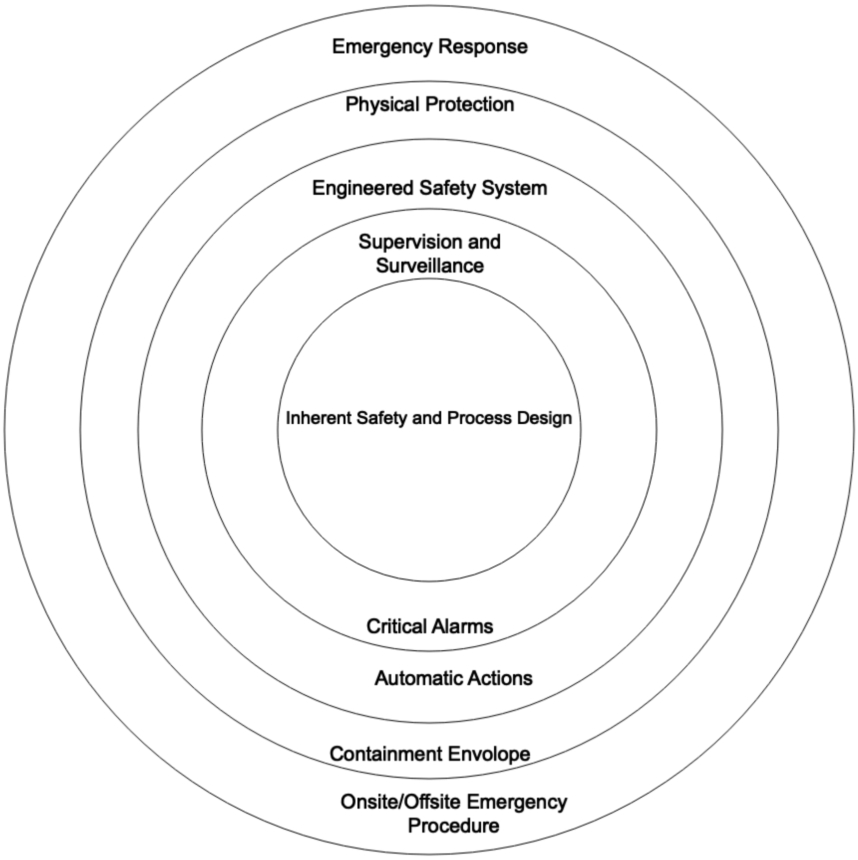

Figure 2. The simplified overview of defense-in-depth strategy (NEA, 1992)

Although deterministic approach plays an important role in nuclear safety management, Probabilistic Safety Analysis (PSA) is also a good example to apply those two strategies in practical nuclear safety management. It is widely performed on nuclear reactors around the world during their initial design and operation stages. The main objective of performing PSA is to ensure various plausible scenarios of nuclear accidents can be taken into account. Through the PSA development, the accident consequences can be well estimated, which could be used as a base reference for risk management improvement. The detailed discussion regarding PSA can be viewed in the following sections.

1.3 Probabilistic Safety Analysis (PSA)

PSA is developed to assess the risk of nuclear plants aiming at better identification of potential accidents. For nuclear plants, the risk is normally evaluated by its radioactive release magnitude and release frequency. As a top-down analysis approach, it is widely applied in industry and it basically consists of three-level analyses. In general, he level 1 PSA investigates a sequence of events that may lead to core damage. It offered an opportunity to look into those safety related systems and analyze their characteristics to understand how they will react to prevent the core damage (IAEA, 2010). If there is a severe core damage, the level 2 PSA is applied to examine how radioactive substances could release through the containment system and their corresponding magnitude. By taking potential failure into consideration, different release scenarios can be generated, which would help Level 3 analysis for evaluating release consequences to the environment and exposure risk for public health. The general steps for performing PSA can be viewed as below:

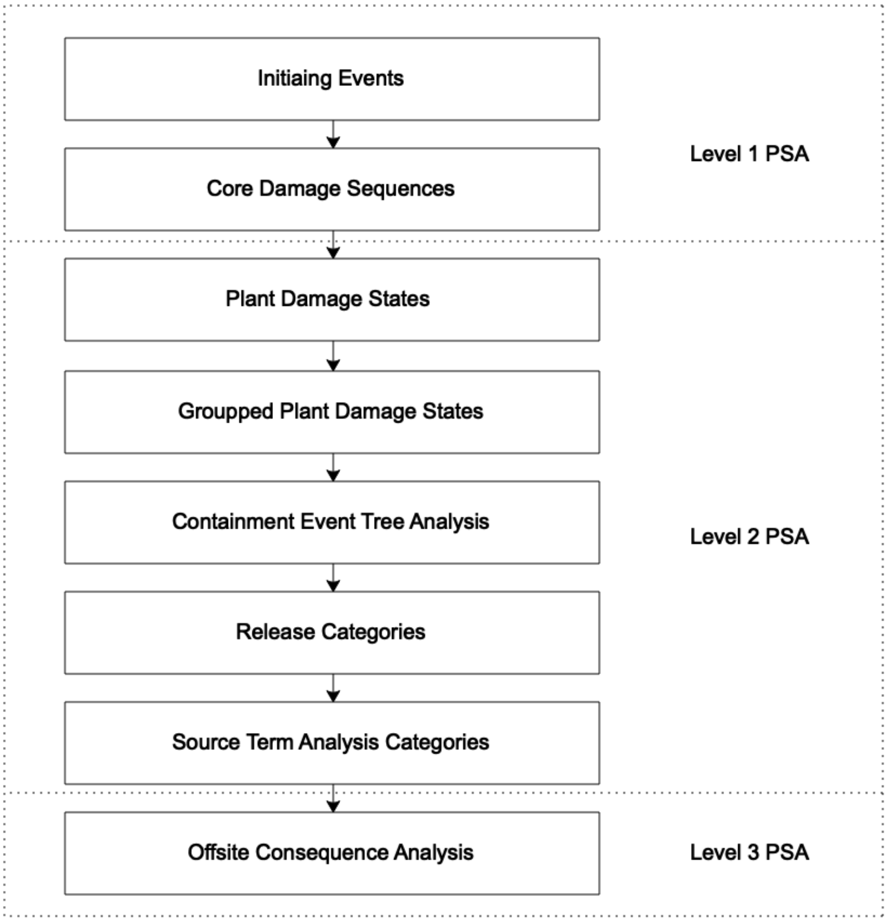

Figure 3. The general overview of the development of PSA (IAEA, 2010)

According to IAEA (2010), the PSA development process does not vary a lot, the general guideline is recommended. Therefore, as the key component of risk analysis, it is quite important to investigate how level 2 PSA could be better performed in order to ensure its effectiveness and applicability for a wide range of nuclear plants.

Chapter 2 Research Definition

As discussed in the first chapter, the level 2 PSA is one of the key components of probabilistic safety analysis (PSA) for nuclear plant risk management. If level 2 PSA models can be improved from status quo, it will greatly contribute to the development of nuclear safety management. Therefore, in this chapter, the research scope is defined to investigate current level 2 PSA studies so that the proposed approach can better improve the nuclear risk management. Through the comparison among level 2 PSA studies, the research motivation for developing generic level 2 PSA was identified and research questions were formulated. By using research question and its sub-questions as the guideline, the research methodology flow was structured.

2.1 Level 2 PSA Introduction

The level 2 PSA is designed to analyze different accident scenarios and potential radioactive material release pathways through the containment system, which includes the accident progression and containment response. In order to analyze accident consequences and quantify frequencies, several steps are suggested by IAEA, which includes familiarization with plants, plant damage states definition, severe accident analysis, containment performance analysis, source term analysis, and quantification (IAEA, 2010). Following a core damage, level 2 PSA is applied to model the accident progression, especially those which may cause potential containment failure and result in the fission products release to the public. The safety design feature of plants and severe accident management effectiveness are considered in the analysis so that it can estimate the frequency of potential release accidents as well as the radioactive release magnitude. Containment event trees (CET) are used to analyze and represent accident progressions and containment response in order to better facilitate source term analysis. Therefore, the establishment approach of CET is very important for the whole analysis. Due to different approaches of building CET, level 2 PSA also got developed and expanded since it was first introduced for plant risk assessment.

The level 2 PSA was first brought out by WASH 1400, in the study of plant risk assessment, which was published in 1975. Since then, the level 2 PSA methodology and plant-specific application have been developing very fast. In 1981, the severe accident scenarios of Zion and Indian Point nuclear power plants were assessed by using code MARCH during level 2 PSA.

Comparing with WASH-1400, it was more expanded and organized. In the early 80s, the Level 2 PSA performed by the USA and Europe were mainly based on the above method (NEA, 2007). Later on, the German PSA studies took Biblis B nuclear plant as the study object in order to provide a better knowledge of severe accidents. After that, NUREG-1150 was perceived as the most comprehensive report for level 2 PSA study in the 1990s. In NUREG-1150, accident progression event tree (APET) approach was introduced to analyze the accident progression for 5 American nuclear plants including PWR Surry and Zion (NEA, 2007). However, these event trees normally have over a hundred events and each event has multiple branches, which made it impossible to graphically represent. Additionally, numerous end states generated from those event trees made them very difficult to understand if no computer-based reduction was applied. Although large event trees could provide a model for a relatively complete estimate of accident progression phenomena, it was still very difficult to figure out the specific event occurrence logic due to its large number of details.

After that, NUERG-5602 was proposed by using simplified containment event trees (SCETs) to analyze the Sequoyah containment. By limiting the number of top events below 20, it improved the understanding of the approach used by NUERG-1150 (USNRC, 1990). However, even though it succeeded in simplifying the event trees and was able to reproduce the results of APET approach, it still has a large number of detailed end states to understand. After that, NEA (1997) summarized the performance and application of level 2 PSA among 10 existing pressurized water reactors (PWRs) around the world. Among those studies, only 2 of them applied large event tree methods and rest used small event trees for analysis for accident progression analysis. The small event tree, also known as the approach using decomposition event trees (DETs) for detailed inspection of top events. It was used to figure out the important sub-events which are useful for CET quantification.

In recent years, other developments regarding physical and chemical phenomena assessment of level 2 PSA has been made as well, and Level 2 PSAs have led to the development of severe accident management based on specific plant types (NEA, 2004). Kujal has performed containment behavior analysis on VVER-1000 based on 5 different severe phenomena with multiple event trees (NEA, 2007). The hydraulic and thermal conditions of French 1300 MW PWR fleets were also studied by IRSN using ASTEC V1.3 code. During the analysis, they took 60 different calculated scenarios in full power situation and over 20 scenarios in shutdown states to generate different event tress during PSA level 2 analysis (Tregoures et al., 2010).

Later on, Zvoncek, et al. (2017) investigated multi-states and different hazards of KKL nuclear plant with 36 release categories. Also, Hwang, et al. (2018) applied DET approach to perform the level 2 PSA analysis of APR1400 nuclear plants by investigating different sequence events. Gunhyo & Moosung (2018) used Westinghouse 3-loop system as the reference to build different scenario event trees for severe accident management.

2.2 Research Motivation in Level 2 PSA

Most of the level 2 PSA studies mentioned above are very detailed, which made it difficult for non-experts to evaluate the containment response behaviors under hundreds of accident scenarios. The current research trend for level 2 PSA is going as detailed as possible, which made all these studies difficult to be implemented for future nuclear plant analysis. Additionally, although similar safety systems are generally utilized at different plant designs, due to their uniqueness of the analysis, it is hard to use these studies as references for other plants analysis. One of the good examples is that after the Fukushima accident, the revision work for PSA model of Japanese domestic nuclear reactors has been conducted. However, because of different details in reactor models, it was very hard to technically standardize PSA model for these domestic plants in a consistent way, which made it insufficient for further discussion on improvements of safety design. Therefore, it is imperative to develop a generic analysis model for level 2 PSA because a generic model can be further used as the reference for precursor analysis. It could benefit a large variety of nuclear plant risk assessments not only in industrial practices but also in academic development.

2.3 Research Question

Based on the analysis and discussion in previous sections, the research question can be summarized as:
What kind of generic level 2 PSA model can be developed in order to better facilitate risk analysis of power plants in nuclear industry?

In order to provide a deep analysis of the research question and design the optimal research scope of the analysis, there are also several sub-questions proposed as follows:

1. *What type of nuclear plants should be selected as the reference target?*
2. *How to perform level 2 Probabilistic Safety Analysis for reference reactor?*
3. *Why is it creative to develop a generic model of level 2 PSA?*

4. *How does a generic level 2 PSA model contribute to the risk analysis of nuclear reactors?*

These listed sub-questions pointed out what should be considered in order to develop a generic level 2 PSA model and why this research work is useful for risk analysis of the nuclear industry. The first sub-question helps to outline the research scope aiming at finding the most critical nuclear type to perform the analysis in order to better facilitate nuclear risk analysis development. Because different types of nuclear reactors may have different design features and structures, it is important to find the most common and widely applied reactor type to perform the analysis.

The second sub-question pointed out the steps that researchers should follow in order to build a generic level 2 PSA model, which also functioned as the guideline to better investigate the main research question. The third sub-question is designed to help understand what the creative points are in generic level 2 PSA compared with other level 2 PSA models. It also strengthens the importance of necessity to build generic level 2 PSA model.

The last sub-question aims to support the main research question to find out in what way generic level 2 PSA could better facilitate the risk analysis of nuclear plants. These four sub-questions along with main research question worked as the guidance for the whole research development.

2.4 Research Flow

The overall research structure can be viewed in figure 4. Along with the research development, research questions get solved step by step following the analysis. In total, there are four chapters in this report and the primary methodology to build generic level 2 PSA model is presented in Chapter 3, which was developed based on the general guideline from IAEA. In conclusion chapter, it summarized the creative points of the research, its limitations and future works.

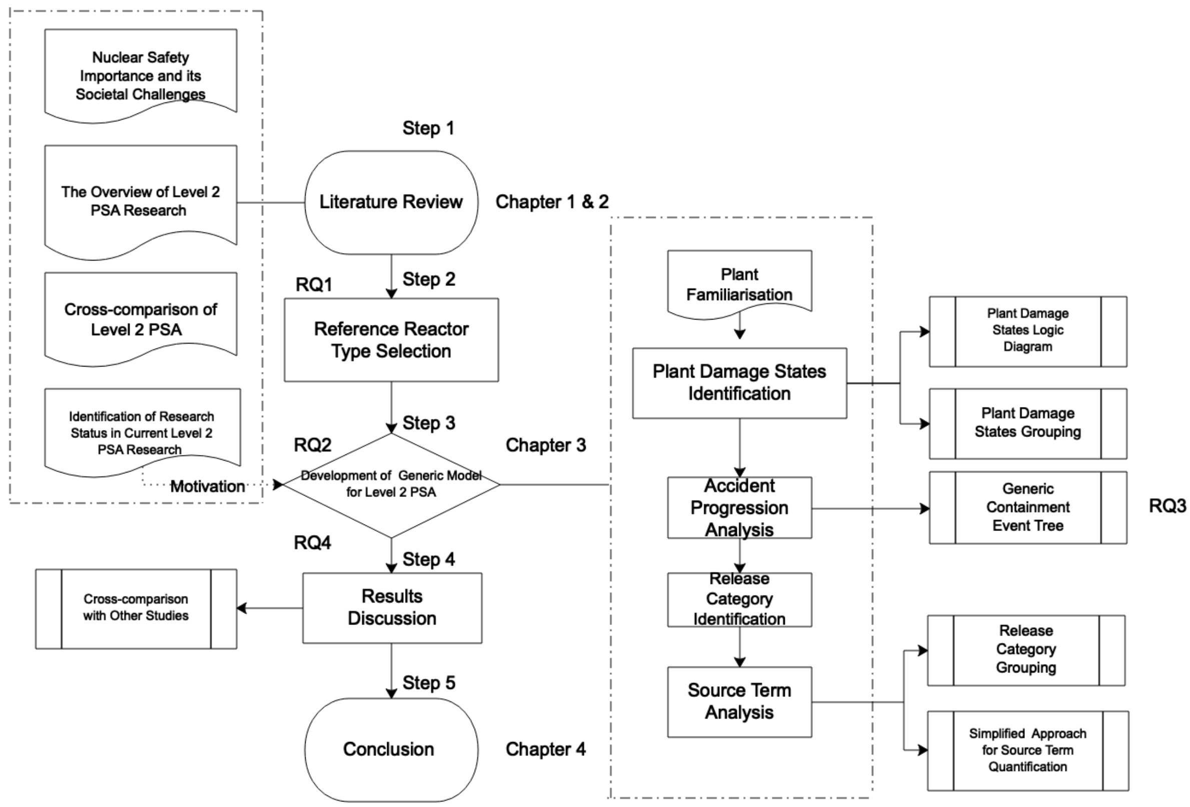

Figure 4. Level 2 PSA generic model development flow

Chapter 3 Generic Level 2 PSA Development

3.1 Reference Reactor Type Selection

At the end of 2017, around 82% of nuclear power plants in operation are light water reactors (Sornette, Wolfgang & Spencer, 2017). In addition, among the light water reactors, 80% reactors are pressurized water reactors (PWRs). In other words, the majority of current nuclear plants in operation are PWRs. Therefore, if a generic PSA level 2 model can be performed on pressurized water reactors, it will benefit a wide range of nuclear power plants which are in operation and also speed up the precursor analysis process for the nuclear industry.

3.1.1 Pressurized Water Reactor

Together with Boiling Water Reactors (BWRs), Pressurized Water Reactors (PWRs) are typical design types of Light Water Reactors (LWRs), which are normally cooled and controlled by regular water during the electricity generation (Thomas, 2019). In PWRs, two water circuits are applied by pumping the primary water coolant to the reactor core in order to remove the heat from fission products (Cummins & Matzie, 2018). The heat later can be transferred to secondary coolant system for steam generation, which could power the turbine for electricity generation. The common design of PWRs consists of a reactor vessel, pressurizers, steam generators, containment engineering safety systems, and radioactive material confinement systems.

The first PWR was commercially launched in 1957 and after that three generations of PWR designs have been successfully developed and commercialized. Shippingport power station is a typical example of Generation I PWRs, which are perceived as early prototype of PWRs. Generation II PWRs refer to those nuclear reactors that were built by the end of 1990s and most of them are current operation nuclear reactors. The typical design operational life is 40 years and most of operational Generation II PWRs in the West were manufactured by one of three companies: Westinghouse, AREVA, and General Electric (Goldberg & Rosner, 2011). In addition, for example, all of the UK' s operational PWRs as well as Chinese CPR-1000, French N4, Russian VVER-1000, and Korean OPR-1000 all belong to Generation II type design. Different from Generation II reactor type design, the lifetime of Generation III PWRs is aimed at 60 years. Generation III PWRs are improved based on the Generation II PWRs and typical design type includes AP600 and APR1400. Beyond these three generation PWRs, there are also Generation III+ PWRs such as AP1000 and EPR. Compared with Generation III PWRs, it has significant improvements in safety systems. The table below shows the typical examples of different generation reactors throughout the world.

Generation type	Reactor design examples
Generation II	CPR1000, VVER400, VVER1000, N4, OPR1000
Generation III	AP600, APR1400
Generation III+	AP1000, EPR

Table 1. PWR design examples among different generations

However, these design prototypes were modified based on different locations, which made each individual risk analysis less valuable for the innovation in the future PWR design in order to further improve the safety functions of reactors. Therefore, if a generic PSA level 2 can be generated by cross-comparing different PWR design, especially the containment safety management features, it will not only improve the safety design for future PWR development but also reduce the complexity of risk assessment for radioactivity release scenarios.

3.2 Plant Familiarization for Level 2 PSA

As discussed in the previous section briefly, the pressurized water reactor (PWR) is the dominant reactor type in the current nuclear industry. The primary system of PWR is consisted of the reactor vessel, the pressurizer, the reactor coolant pump, the steam generator, and the connecting piping. The reactor vessel is generally perceived as the key part of PWR because all nuclear fission reactions took place within it and it is the main house for the core barrel, the reactor core, and the upper internals package. The steam generator is applied between primary and secondary coolant loops for picking up the heat generated by the reactor vessel. Within the steam generator, hot reactor coolant flows through multiple tubes and exchanged heat with the outside feedwater. After the absorption of the heat, the secondary coolant system starts to form the steam. The pressurizer is used to control the system pressure within the primary system, which contains pressurizer sprays, relief valves, safety valves, and electrical heaters. These components are targeted to bring the pressure back to the desired value once the pressure drops or increases. The pressure deviation is normally induced by temperature changes in the reactor coolant system. For example, if the temperature starts to increase in the coolant system, the water will expand to the pressurizer through the surge line, which will make the steam within the pressurizer compressed. Therefore, the pressure in the pressurizer will get increased. In order to bring the pressure back to the normal stage, the pressurizer spray will function to condense the steam and reduce the pressure. If it does not stop the pressure increase, the pressurizer relief valve and safety valve will function to continue reducing the pressure.

Besides the primary system, there are several other systems within PWR containment to ensure the safety of nuclear fission reaction, which includes emergency core cooling system (ECCS), containment safety systems and so on. Generally, all PWRs are equipped with emergency water feed-up system in case that normal feed-up is lost or a major release in the reactor coolant loop. Those systems are named as emergency core cooling system, including the high-pressure injection system and the low-pressure injection system. One purpose of ECCS is to provide the make-up water to cool the core. In order to alleviate the core damage at the event of loss of coolant, the large amount of borated water will be injected to the coolant system. It is normally performed in a short period of time after the initiation of LOCA to maintain post-accident core cooling. The other purpose of ECCS is to ensure the reactor not to produce the power after cooldown by injecting the cool water into the coolant system. The high-pressure injection system automatically functioned when the reactor coolant system pressure is relatively high at the event of small LOCA. In comparison, the low-pressure injection system is used when there are large breaks in the reactor coolant system. Also, when the coolant loss capacity exceeds the range that the high-pressure injection system can control, the low-pressure injection system can function to mitigate the accident. Because the coolant system depressurized in a very fast speed, it allows the flow from the low-pressure injection system to limit the core temperature to rise. In addition to the short-term core cooling mode, the long-term core cooling is another feature of low-pressure injection system. It can take the water from containment sump to pick up the residual heat especially when the coolant water storage tank went empty.

If previous systems all failed to ensure the safety of PWR, containment and its safety systems are perceived as last barriers to mitigate the accident progression and confine the radioactive material release within the containment. In normal operation, the containment safety system is designed as the standby mode. The containment is designed to withstand the pressure, temperature and mechanical loading induced by the ejection or release of high energy fluid.

The containment structure is normally made from steel and concrete and the size of the containment is heavily determined by its design pressure and temperature. According to the Design of Reactor Containment Systems for Nuclear Power Plants published by IAEA (2004), it is recommended that the primary containment and its support systems should be available as needed. In addition, these systems must stay effective after the initiating events for a long period of time until they are not needed. The graph below depicts a simplified common design layout of the PWR containment building.

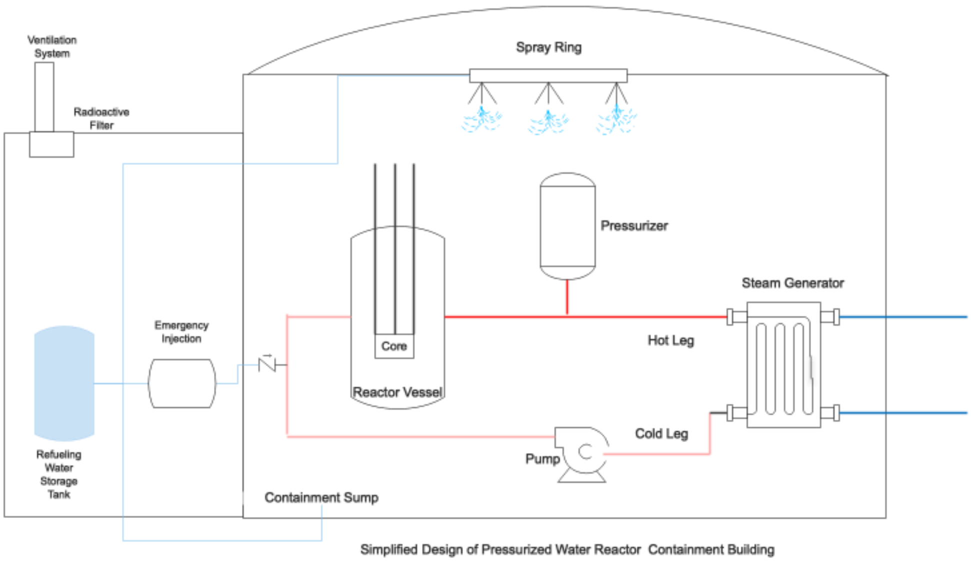

Figure 5. The simplified layout of common type PWR containment building

As level 2 PSA is designed to investigate the containment response regarding the accident progression, therefore, it is imperative to have a detailed look at the containment safety systems and their functions. Containment safety systems are used to prevent and mitigate the potential accidents that may lead to huge environmental impact. Based on different functions, it can be classified into pressure and temperature control systems, hydrogen control systems and radioactive release control systems. The purpose of these systems can be summarized as below:

1. To control the containment temperature and pressure at the desired level.
2. To remove and reduce the hydrogen concentration in the containment in order to prevent potential hydrogen detonation or deflagration.
3. To minimize the radioactive release to the environment after the internal accident (e.g. LOCA) by reducing the temperature and pressure within containment.
4. To reduce and confine the radioactive substance within the containment.

To maintain the temperature and pressure in the containment, several systems have been designed for different types of PWRs, including the containment spray and its sump water recirculation system, the air cooler system, the ice condenser system, and the passive cooling system. During the normal operation, these systems are effective on demand. However, not all PWRs have equipped these systems. Only several American PWRs have the ice condenser system and also the passive cooling system is only available for Generation III/III+ design type of reactors. In addition to those above systems, the containment structure and its volume are designed to withstand certain pressure and temperature in order to well confine the radioactive release into the containment in case the release accident happened. As the inherent safety design feature, the volume of the containment envelope determines the maximum pressure it can hold. Different from those standby systems, ventilation systems also keep operating during the normal operation. They are used to maintain the containment pressure and humidity at the desired level. The table below shows a generic classification of different temperature and pressure control system implementations among various generations of PWRs.

Function	System	Type of design (example)	Generation
Pressure suppression and temperature control	Containment spray system	AP600, AC600, KKB (most PWRs equipped spray systems)	II, III, III+
	Air cooler system (the air cooler fan)	VVER500/600, AP600, AC600, OPR1000	II, III, III+
	Passive containment cooling system (PCCS)	AP1000, HPR1000, VVER1200, CAP1400,	III+

Table 2. Generic simplification of temperature and pressure control systems in PWRs (NEA/CSNI/R(2014)8, 2014)

Detailed descriptions regarding these temperature and pressure control systems of PWRs can be viewed as follows:

3.2.1 Containment spray systems

Containment spray systems are primarily designed to remove the heat from the containment once the design-based accident happened so that they could control the temperature and pressure of the containment atmosphere without increasing too fast. Therefore, it also can be perceived as the mitigation system, which can mitigate the accident progression through heat and pressure control. According to the IAEA containment design guideline (2014), containment spray systems should be designed so that it can make the water effectively interact with the steam in the whole containment. Containment spray systems generally consist of spray headers, nozzles, and recirculation systems. Nozzles and headers should be able to evenly distribute the spray within the containment so that it could efficiently balance the temperature of containment atmosphere whenever the temperature goes up. Also, it is necessary to ensure the nozzles could function in case of any clogging issues caused by the intaken debris. In recirculation systems, there is always a large storage tank for the water supply. In case that the spray system will work in a recirculation loop, there should be a containment sump for the spray system to collect the water.

When the accident happened, the containment spray system is designed to automatically start and take the water from the refueling water storage tank. By pumping the water into spray rings, the water droplets could efficiently remove the heat from the steam and get them condensed. This will help to reduce the containment pressure as well as cooling down the containment atmosphere. Once it is in the recirculation mode, the containment spray system can take the water from the containment sump when the storage tank is empty.

Air cooler systems

Air cooler systems are designed to cool down the containment atmosphere through the air circulation. The heat removal capacity is an important aspect to look at during the design of fan coolers. Although it is another type of containment temperature control system, not a lot of plants solely rely on its dry cooling mechanism.

Passive Containment Cooling System (PCCS)

As the main passive safety system in the containment, the passive containment cooling systems are widely applied in advanced pressurized water reactors. The mechanism of these systems is to use the naturally induced airflow and gravity water flow to cool down the containment. When the heat is generated through accidents such as LOCA or main steam line break (MSLB), the water and airflow work together to provide the evaporative cooling for the containment. However, these systems are not well implemented for current PWRs around the world.

In addition to the overheated and over-pressurized issues that may damage the containment integrity, hydrogen detonation and deflagration are another big concern for containment safety management. In normal operation, hydrogen and oxygen are produced due to the water radiolysis. However, in severe accidents, hydrogen is mainly generated by the zirconium oxidation with steam. The zirconium is in the cladding and fuel element structures. When the core degraded, the heated zirconium will react with the steam producing a high local concentration of hydrogen in a short period of time.

In severe accidents, hydrogen can be produced even in the speed of 5 kg/s because of zirconium oxidation by steam (Bal, 2012). Under this scenario, the hydrogen combustion can create rapid pressure increase or detonation forces that could potentially result in early containment failure because the induced power may exceed the design limit. Moreover, the hydrogen can also be generated during the long-term pressure build-up process, in which the hydrogen is mostly generated through the interaction of the molten corium with containment basemat concrete. In order to prevent the potential hydrogen accumulation and burn accident, several mitigation measures are introduced in PWR containment design, which includes mixing, deliberate ignition, catalytic recombiner usage, and inertion. According to IAEA (2011), there are no strict regulatory requirements for hydrogen mitigating system, which means not all current PWRs have implemented these above mitigation systems. For example, USA does not set any specific requirements on the installation of hydrogen recombiner or igniters for current operating PWRs, but most European countries have required their operating plants to back-fit hydrogen mitigation systems (NEA, 2014). Therefore, the implementation situation for hydrogen mitigation systems varies from country to country and even among different plant generations within a country. In some plants, different approaches that have been developed are able to control and mitigate the potential hydrogen accidents. The table below shows a generic classification of different hydrogen control system implementations among various generations of PWRs.

Function	System	Type of design (example)	Generation
Hydrogen mitigation and control	Prevent flammable mixtures by hydrogen	Most European PWRs back-fitted	II, III, III+
	Prevent potential combustion accidents	AP600, AC600, AP1000, HPR1000,	II, III, III+

Table 3. Generic simplification of hydrogen control systems in PWRs (NEA/CSNI/R(2014)8, 2014)

3.2.2 Hydrogen Mitigation Systems

From the table above, it can be concluded that current hydrogen mitigation approaches for the PWRs can be classified as below:

1. Mixing or reducing the hydrogen with other gases to prevent localized concentration. (hydrogen mixing system and hydrogen recombiner)
2. igniting the hydrogen timely to prevent potential explosion. (hydrogen ignition systems.)

Hydrogen Mixing Systems

Hydrogen mixing systems are designed to prevent the hydrogen concentration accumulated in a certain location within the containment. The system normally consisted of hydrogen mixing fans, which could suck the hydrogen from the top of the containment and create the turbulence to delude the local concentration under flammability point. In addition to the hydrogen mixing system, the hydrogen vent system also helps to delude the hydrogen concentration through the purge system with certain filters. Filtered air from hydrogen vent system is used to maintain the containment pressure under desired limits.

Hydrogen catalytic recombiners are widely applied in different PWRs to prevent hydrogen concentration reaching the flammability point. By using catalysts to control the hydrogen oxidization process, current catalytic recombiners for containment hydrogen control can be classified into two categories: conventional catalytic recombiner and passive autocatalytic recombiner (PAR). Based on the guideline in the report NEA/CSNI/R(2014)8 published by NEA (2014), conventional catalytic recombiners are operated similarly like electrically powered thermal recombiners to limit the hydrogen concentration. Most of them are situated outside of the containment. In order to maintain reliability and basic functions, they required multiple testing and complex support systems. However, compared with conventional catalytic recombiner, passive catalytic recombiners have higher reliability and require no operator action because they use the oxidation reaction heat to produce natural convection flow through the unit (IAEA, 2001). Passive catalytic recombiners consist of catalyst surfaces with an open-ended enclosure. When the hydrogen and oxygen mixed gas reached the surfaces, the oxidization reaction automatically took place. The reaction heat will create the convection flow to exhaust the hydrogen-depleted air to the containment and suck the more combustible gas from below. Moreover, the capabilities of these systems are highly dependant on mass transfer limitations and the preferred working condition for them is to start under the cool conditions.

Hydrogen Ignition Systems

As another approach to reduce the hydrogen concentration, hydrogen igniters are designed to ignite the combustible gas when the flammable mixtures increase and remove the hydrogen through the gentle deflagration. They are mostly used when the release rate of hydrogen is beyond the processing capacity of the mixing system or recombiners. The reason why ignition could become one of hydrogen mitigation systems is that if eventually the flammable mixture will get ignited by random source, it would be better to apply slow deflagration to reduce the consequence as low as possible. However, the potential risk of this approach can also not be neglected because if the deflagration is initiated in one location, there will be possibilities to propagate to other regions. Especially when it propagated to the hydrogen release point region, it will become very dangerous because the deflagration speed will be out of control. It may result in unexpected accidents happened within the containment. Therefore, a detailed and careful placement analysis for ignition systems is required and from current application status, the potential combustion risk can be well controlled when they are coupled with spray systems (NRC, 1983).

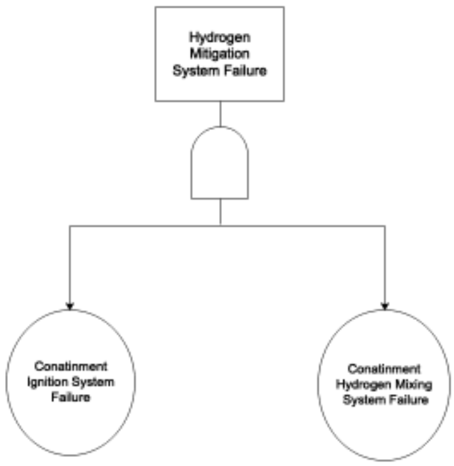

Figure 6. The simplified fault tree for hydrogen mitigation system in PWRs[1]

In addition to recombiner system and ignition system, there is another approach to mitigate the hydrogen risk, but most of them are applied for small PWRs and BWRs. This third approach is called inertion, which can be classified into pre-accident inertion and inertion after the accident. Inerting systems are designed according to certain regulations in case that there should be hydrogen-burning risk-free situations for nuclear plants. Therefore, it is imperative to create an atmosphere with oxygen-depleted in the containment before normal operation. Because nitrogen is very stable so that it was chosen as common inert gas. By injecting the inert gas into the containment to replace the air, the oxygen concentration could be reduced below the required level for potential combustion. Thus, a hydrogen burning risk free environment could be created for the containment. Inertion is also regarded as one of the efficient ways to mitigate hydrogen risk.

3.2.3 Radioactive Release Control System

containment system is designed to confine and envelope the radioactive substance within the containment. Therefore, it is important to ensure all the boundary structures and components well isolated because once radioactive substances got leaked, they may probably lead to the unacceptable radioactive release to the environment. Activated corrosion products are generally perceived as the primary radiation sources for most plants. But if there are a lot of

[1] Hereby the logic gate is "and gate", which denotes only two systems both failed, the hydrogen accident can happen. As long as one system is in operation, it can mitigate the hydrogen risk.

failed fuel cladding, the amount of fission products will also be lot (IAEA, 2005). Above radioactive sources are all originated from the core and transported to the containment. Therefore, containment structures and its supporting systems should be as reliable as possible no matter when the accident happened. Containment isolation systems served as the last barrier for the containment radioactive release control, which consists of actuators, isolation valves, and connecting pipes. When the accident happened, if the radioactive release control is required, they are functioned to be reliably closed in order to reduce the radionuclide escape paths to the environment. They are also required to work in the independent mode for the safety concern. During the normal operation, these systems are all in standby modes. However, the containment will be opened for the purpose of the maintenance work during the shutdown period. Typical radioactive release control systems are spray systems and filtered ventilation system. In general, two basic mechanisms are applied to manage the aerosol nuclides: 1. agglomeration and gravitational setting. 2. spray removal. IAEA (1999) has recommended the revised model for the gravitational setting of nuclides by introducing the removal rate as the function of time, which can be seen as below:

$$\frac{dC}{dt} = -\lambda\, C$$

λ is the removal rate coefficient, C is the mass concentration. In general, the removal rate coefficient is determined by the aerosol source strength and its corresponding concentration level. Two scenarios are proposed by IAEA (1999), which are large continuing source state and weak or zero source state. Therefore, it is important to understand which state should be used in different time steps, the criterion is given as below:

$$If \quad 3S/\lambda_{op} \begin{cases} \leq C, & use \text{ "}continuing\ sourece\text{"} \\ > C, & use \text{ "}weak\ or\ zero\ source\text{"} \end{cases}$$

$$\lambda_{op} = \left(\frac{\alpha K_0 g \rho}{\gamma x^2 \mu h^2 \varepsilon_0}\right)^{1/2} \Lambda_{op}$$

Hereby, S = source mass rate
λ_{op} = the optimal removal rate coefficient.
α = morphology correction factor
K_0= Brownian agglomeration rate coefficient
g = gravity acceleration rate
ρ = particle material density

γ = collision morphology correction factor
μ = gas kinetic viscosity
h = settling height (the ratio of the containment free volume to the horizontal surface area)
ε_0= collision efficiency scale factor
Λ_{op} = fraction of concentration

Based on different source strength relative to the concentration levels, the sedimentation rate for radionuclide removal can be obtained. In addition to agglomeration, the basic equations for spray removal algorithm can also be viewed as below:

$$\frac{dm_s}{dt} = -\lambda\, m_s$$

m_s is the mass fraction of remaining gas as a function of time t and the removal rate coefficient is related to spray volume, flow rate, droplet size, and its fall height. As the radionuclides released to the containment, the particles can be dissolved by the water spray and flowed along the containment wall to the bottom. If there are still airborne radioactive substances left in the containment, the filtered venting systems are available for purifying the gaseous mixture before they escaped to the environment. Filtered ventilation systems are now widely applied in new PWR designs and after stress test, most European plants have been back-fitted these systems (Bal, Jose & Meikap, 2019). Ventilation system are mostly used for filtering the discharged air in order to reduce the environmental impact due to the accident release. Filters must be designed to limit any potential radioactive release concentration below the allowed value. Moreover, the supporting system that can prevent the filter inlet air temperature dropping below dew point also needs to be equipped in order to ensure the reliability of the radioactive control. The common design types of filter systems are sand bed scrubber system, multi-venturi scrubber system, and charcoal filters (NEA, 2014). Furthermore, venting is also another useful method to mitigate overpressurisation when the core melt accident happened. During the normal operation, the ventilation system needs to be effective in order to balance the containment pressure with the outside environment. It can be used to adjust the temperature and keep the humidity of the containment. When the accident happened, it can also help to mitigate the combustion risk. For example, hydrogen buildup depends on the effluent rate and pressure. Venting can reduce the pressure and filter the radioactive nuclides so that it can effectively reduce the accident consequences. Although isolation is an effective way to control the release rate below the specific limits, spray and venting are also useful regarding radioactive inventory reduction. The table below describes the common type of radioactive control systems applied in PWRs among different generations.

Function	system	Type of Design (example)	Generation
Radioactive Release Control	Containment spray system recirculation	AP600, AC600, KKB (most PWRs equipped spray systems)	II, III, III+
	Filtered containment ventilation system (FCVS)	Most European PWRs back-fitted	II, III, III+
	Containment isolation system	EPR, AP600, CP4	II, III, III+

Table 4. Generic simplification of radioactive release control systems in PWRs (NEA/CSNI/R(2014)7, 2014)

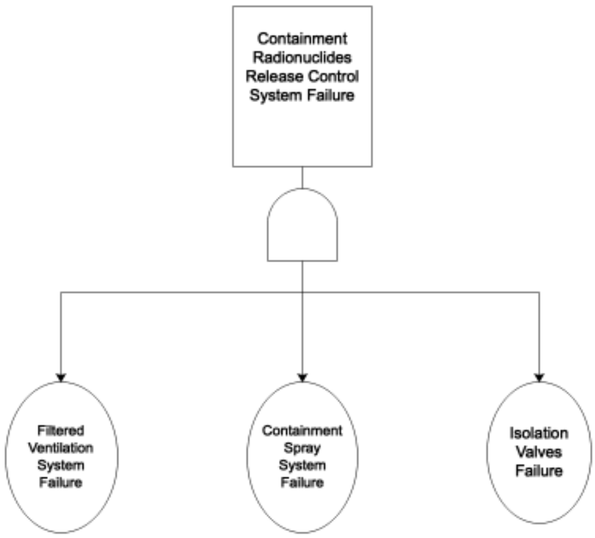

Figure 7. The simplified fault tree for the radioactive release control system in PWRs[2]

3.2.4 Summary of Containment Safety Systems

After a detailed discussion and analysis of containment safety systems, the typical common safety systems can be identified, which can be further used to analyze the accident progression in the containment event tree analysis. By taking these common types of safety system design into consideration, the developed containment event tree is more applicable for a large range

[2] Hereby the logic gate is "and gate", which denotes only three systems all failed, the radioactive release can escape to the environment. As long as one system is in operation, it can prevent the radioactive release escaping to the public.

of PWRs. It was expected to provide preliminary insight and general overview of accident mitigation methods for plant risk analysis. By applying the proposed containment event tree, it can improve the analysis efficiency without going through every specific design and accident types to find the optimal solutions. Furthermore, in order to better facilitate containment accident progression analysis, containment safety systems of different generation PWRs among 10 countries, 3 continents have been cross-compared. The common containment accident mitigation systems are summarized as below:

Containment safety system	Ringhals 2,3,4 (Sweden)	Biblis-B (Germany)	Sizewell-B (UK)	Beznau (KKB) (Switzerland)	Sanmen(AP1000) (China)
Containment spray system	√	√	√	√	√
Hydrogen control	Recombiners	Igniters/Recom-biners	Recombiners	Igniter/Recomb-iners	Igniter/Recombiners
Filtered containment venting system	√	√	√	√	√

Table 5. Cross comparison of containment safety systems in PWRs among different countries

Containment safety system	CP4, N4, EPR (France)	OPR1000 (South Korea)	Onagawa,Takahama (Japan)	Borssele (The Netherlands)	Most PWRs in operation (USA)
Containment spray system	√	√	√	√	√
Hydrogen control system	Recombiners	Igniters/Recombine-rs	Recombiner	Igniter/Recombine-rs	Thermal recombiners[3]
Filtered containment venting system	√	√	√	√	[4]

Table 6. Cross-comparison of containment safety systems in PWRs among different countries

[3] The current PWRs (besides AP1000) with large dry containment in the US do not use PARs nor ignition systems to prevent the hydrogen risk because the containment features are designed to withstand the hydrogen explosion. Hydrogen igniters are mainly used in those PWRs with ice condensers. (NEA/CSNI/R(2014)8, 2014). pp.55.

[4] Preparation of guidance documents on Hardened CVS for BWR Mark I & II implemented, the PWR implementation is still under development. (NEA/CSNI/R(2014)7, 2014) pp.44.

3.3 Plant Damage States Identification

Plant Damage States (PDSs) are the interfaces which linked Level 1 PSA to Level 2 PSA. Aiming at better investigating how accident progression would impact containment integrity and radioactive release, several ranges of failure events identified in level 1 PSA are grouped into different bins. These bins are known as plant damage states, which will be further used to define the initial and boundary conditions for severe accident analysis. By reducing a large number of fault sequences into several plant damage states, it makes the severe accident analysis more manageable. As the starting point of the containment event trees, key attributes to the accident progression and the source term release are used to define the PDSs. It includes the primary system functional status, operation system variables, and accident initiators, etc. Therefore, the general PDSs attributes for pressurized water reactor accident progression can be classified into these categories:

1. The initiating events for accidents. (e.g. Loss of Coolant Accident or Transient)
2. The pressure status of the primary system
3. The safety injection system or emergency cooling system status
4. The status of containment engineering safety system. (e.g. containment heat removal system, hydrogen mitigation system, radioactive release control system.)
5. The status of containment integrity.

By taking these PDSs attributes into consideration, PDSs logic diagram was developed in the research, from which corresponding plant damage states can be obtained. In this research, it was assumed the plant is under full power operation when the initiating events occurred. The graph below illustrates how failure sequences form into different plant damage states.

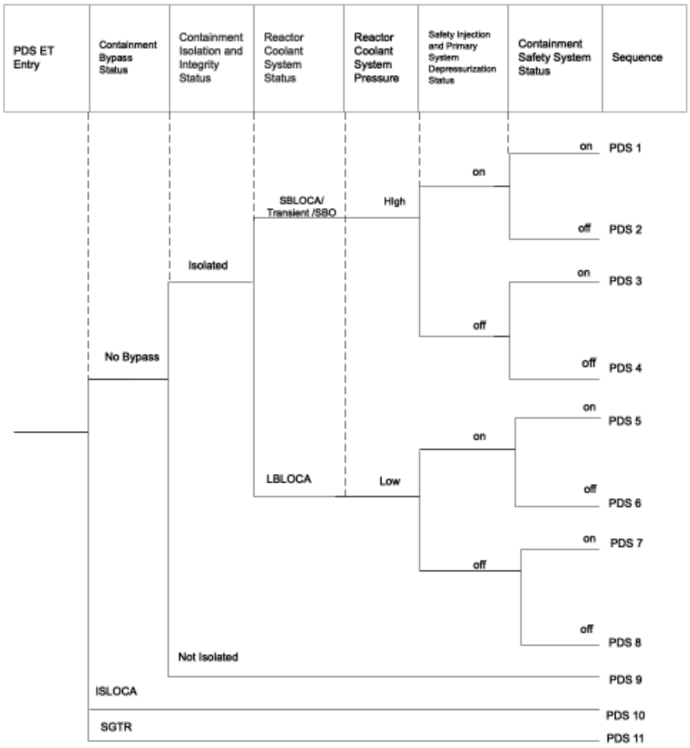

Figure 8. Plant damage states logic diagram

As depicted in the graph above, 6 plant damage state attributes are identified, which are containment bypass status, containment integrity and isolation status, reactor coolant system status, reactor coolant system pressure status, safety injection and primary system depressurization status and containment safety system status. These six key attributes basically covered all parameters recommended by IAEA for plant damage states identification (IAEA, 2010). The containment bypass status and isolation status are related to containment integrity. The containment could be bypassed because of the LOCA occurred in an interfacing system or steam generator tube rupture event, which may lead to significant release to the environment. The containment isolation system can fail before the core melt, once release accidents happened, it can result in large or small leakage. The isolation system also can fail to mitigate the accident progression after the core melt. When severe accidents happened, the isolation valves functioned as the last barriers to confine the radioactive release. When the pressure or temperature exceeded their design features, the isolation failure occurred.

For reactor coolant system status, Loss of coolant and transient events are considered as initiating events of potential containment failure accidents. Loss of coolant accidents can range from small rupture in the boundary of the reactor coolant system to the large leak in the primary coolant recirculation system. Due to the different size of the rupture, the system response varies a lot. For the large break in the primary system, the pressure rapidly drops down and the coolant water quickly discharges to the containment if no further safety injection applied. Even the reactor is at shut-down state, the decay heat still can lead to the temperature rise in the cladding. Once the fuel clad was overheated and ruptured, the fission products would get released to the containment. Also, depending on other containment safety system status, the release amount of fission products to the environment varies a lot. But the impact of large LOCAs to challenge the containment integrity should be well considered.

Different from large break in the primary system, small LOCAs include not only pipe breaks of limited aperture but also, for example, the inadvertent opening of relief valves and their possible failure to reseat (Myerscough, 1992). The small LOCAs normally will lead to slow depressurization but when it happened, the primary system still remains high pressure. If all safety injection systems are in operation, the safety status of the system will not be challenged very much. Only the coolant water directly leaks into the containment. In the plant damage analysis, different amounts of loss of coolant are taken into consideration because they may lead to different scenarios for containment accident progression. In particular, the pressure status of the reactor coolant system is very important for defining success criteria for in-vessel injection in the plant damage logic diagram. Also, the functional status of the in-vessel injection system implies whether the electric power system is on or not, the station blackout and normal operation scenarios could be captured. In addition, the containment safety system is closely related to mitigation and prevention of accident progression within containment. Therefore, its functional status reflects how severe the accident may lead to. By following these above PDSs attributes, 11 plant damage states are identified in the logic diagram and they will be further grouped into small bins as the entry points for containment event tree analysis.

3.3.1 Plant Damage State Grouping

In general, there are two methods can be applied for categorizing multiple plant damage states. One is to classify plant damage states based on their similarities and then select the typical event sequences which could characterize plant damage states for further level 2 analysis (NEA, 2007). The other one is using cut-off criteria based on occurrence frequency for plant damage state screening. In this research, the first approach is applied and 5 plant damage state bins are identified. Roughly speaking, two big clusters can be used to describe the plant damage states:

those where the radionuclide release occurred due to the accident progression and those where the containment safety systems were bypassed or the containment lost integrity on demand. In the first cluster, 3 bins are proposed which include in-vessel core cooled, core melt with high pressure in the primary system and core melt with low pressure in the primary system. The in-vessel phenomena not only consider about how core melt progressed during the in-vessel phase but also take the potential radioactive substance release scenario into account. Due to different pressure status within the vessel, it may result in different release scenarios for radionuclide ex-vessel transport. When the core melt accident progressed into the containment, it always led to severe accidents and radionuclide release paths should be well considered (NEA, 2007). The in-vessel core cooled group are used to define those in which the initiating events occurred, but the emergency coolant injection and depressurization systems worked to mitigate the accident progression within the primary system. The core melt with high pressure in the reactor group stands for those in which initiating events occurred but emergency injection and depressurization systems failed to bring down the pressure in the reactor, meanwhile the core melt progression was not stopped. The typical scenario can be the injection system and recirculation system both failed when the transient or small break LOCA occurred. For the group of core melt with low pressure in the primary system, it describes those in which initiating events happened while the primary system was already depressurized by the initiating events but safety injection systems failed to mitigate the core melt progression. These two plant damage states all can lead to the severe accidents due to the reactor vessel rupture. Due to the pressure status difference in the reactor, the high pressure may lead to high pressure ejection failure of reactor vessel, the low pressure can cause the corium melt through the reactor vessel. Therefore, these two plant damage state groups may lead to different containment response.

In the second cluster, it consists of containment isolation failure and bypass failure. The graph below shows 5 different groups of plant damage states, which will be further taken into consideration in containment event tree analysis.

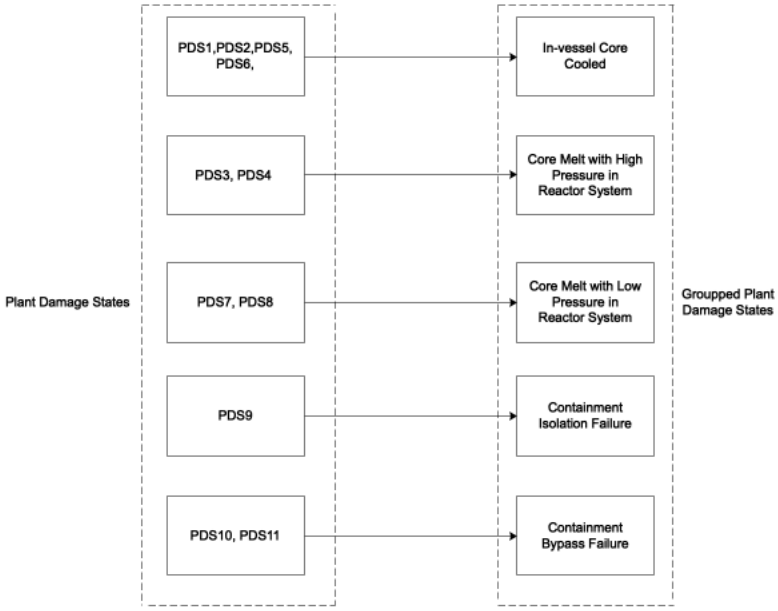

Figure 9. Plant damage sate grouping

3.4 Accident Progression Analysis

Accident progression analysis is applied to analyze how core damage progression could impact plant integrity. As the key part of level 2 PSA, it investigated the challenges posed by the core damage events on the containment and its supporting systems. Additionally, it depicted potential radioactive release sequences for identified PDSs, which could be used as the reference for the future safety design improvements of nuclear plants. In order to perform this analysis, containment event tree (CET) are developed for having a detailed look at how accidents could occur after severe initial events and how these accidents lead to the radioactive release.

As one of the key parts for accident progression analysis, CET is built through different event nodes and these nodes follow a chronological sequence based on accident progression. It started from the core damage through reactor vessel failure to the containment failure in both short time interval and large time interval. These different time stages are important for analyzing accident progression because fission product behaviors vary a lot in different progression stages. In general, the initiating events for core melt are the entry for the CET, events immediately following the reactor vessel failure and after the reactor vessel failure were analyzed for containment response.

Therefore, it covers progression stages from the core melt until the core melt into the containment. It is recommended that an adequate number of time frames and nodes need to be defined to allow all the significant phenomena that are relevant during each time frame to be addressed (NEA, 2007). In principle, there are two approaches to build the CET. One is to create a generic event tree so that it could cover most scenarios that may impact the containment integrity. The other is to build multiple event trees for different PDSs and analyze these scenarios one by one. As discussed in previous sections, the current research trend is to build multiple detailed event trees for different reactor designs, which can not provide a general overview of how containment failure could occur in the whole system.

Therefore, hereby a generic model was proposed in the research in order to capture most of the scenarios that may impact the accident progression. As the CET describes different containment conditions and their failure modes during the severe accident progressions. Any significant phenomena that may impact on source term release need to be treated as top events in the containment event tree. 8 top events are selected in this CET in order to build a generic CET model, which includes containment bypass status, containment isolation status, core melt progression with primary reactor rupture, corium release status, containment early status, containment spray system conditions, containment hydrogen mitigation system conditions and containment filtered venting system conditions. These 8 categories covered the progression stage starting from core melt through the reactor rupture up to the late stage of containment radioactive control system response. They also present how different containment failure modes can occur with the accident progression. Different paths taken by initial events can lead to various containment failure scenarios and due to the time effect, the containment condition also varied during the accident progression. Therefore, by following the structure and paths built in the generic CET, there are 22 CET events identified in this research, which can be viewed as the graph below.

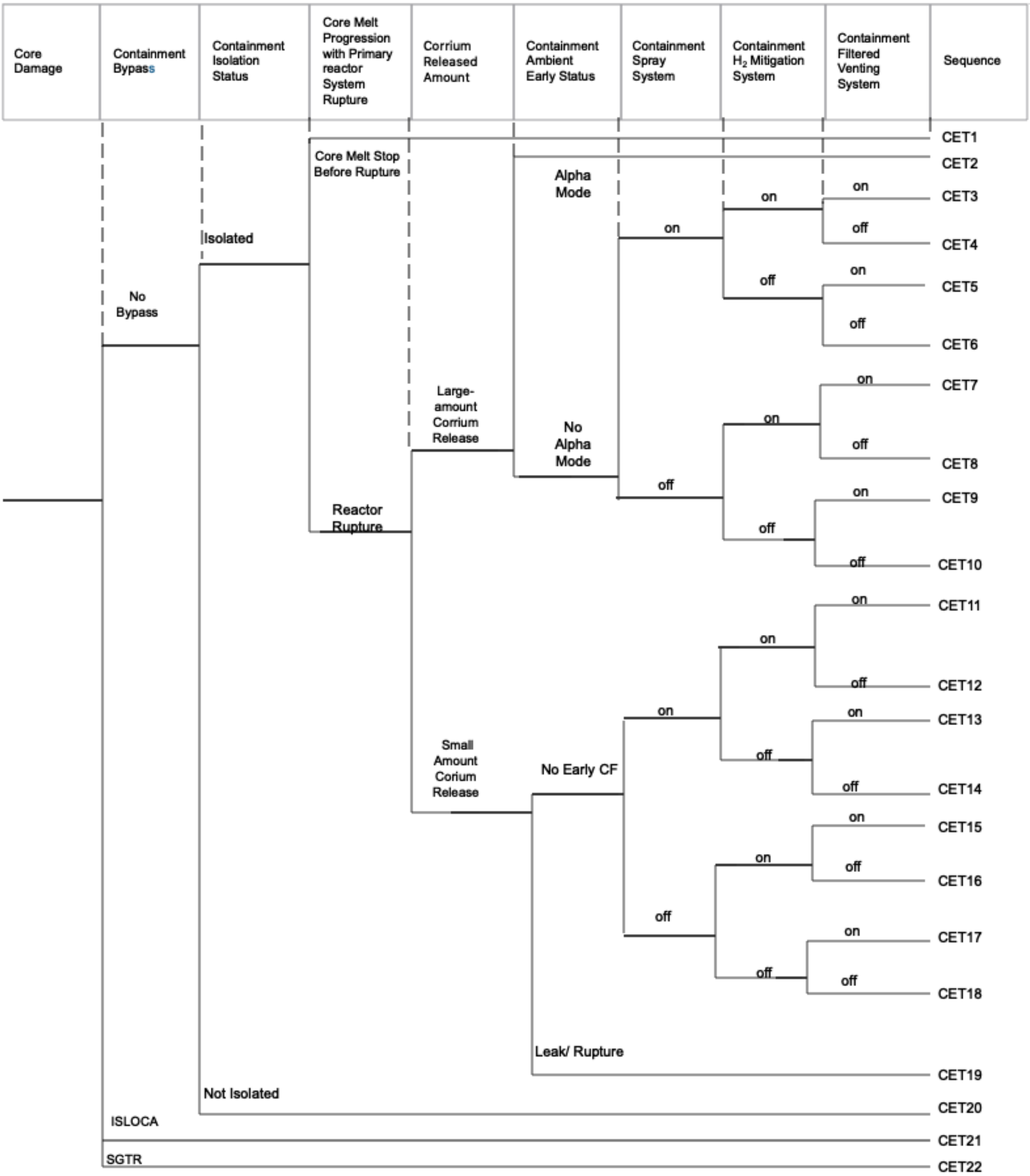

Figure 10. Containment event tree

Containment bypass events are those events that would lead to fission products released to the environment even though the containment structure is intact. The typical containment bypass accidents include steam generator tube rupture (SGTR) and interfacing system loss of coolant accident (ISLOCA). In general, there are two kinds of SGTR, spontaneous SGTR and consequential SGTR and the former one is normally caused by maloperation before the core melt (Song, et al., 2019). Consequential SGTR is considered because steam generator tube failures caused by hot gases from a damaged reactor core can result in a containment bypass event and will lead to a large amount of radioactive release to the environment (Song, et al., 2019). Several studies have been performed on the SGTR from tube thermal and structural responses perspective (USNRC, 2016). For ISLOCA, it is perceived as the pipeline rupture outside the containment building where the pipeline is linked to the reactor coolant system after the core damage. It will lead to the direct radioactive discharge to the environment without going through any filtering systems within containment.

Containment isolation status is another big concern regarding the release consequence it may cause to the environment and personnel. In general, there are two scenarios for isolation failure, one is the containment not isolated at the beginning. Another is the isolation failure due to the accident progression. If containment isolation failures happened in the early stage, the radioactive release will lead to big damage to the surrounding environment. The valves and other connecting systems are redundantly designed to isolate the penetrations within the containment. They are required to function reliably and independently when it is necessary to close in case of design-based accidents (IAEA, 2004). In normal operation, the venting system works to maintain the containment pressure and in shutdown states, they are also intentionally open to give access for operators to perform maintenance work. However, due to the very low probability of accidents happened during the operation, hereby the isolation events are more referred to those valves and connecting system failures.

If the containment is intact but the reactor got ruptured at the beginning, it will have the high chance to cause the severe accidents. However, if the core melt is confined and controlled within the reactor, the damage can be well reduced. The core melt accidents normally occurred because the heat generated from fission reaction exceeded the heat removal capacity of the plant coolant system. The typical events that will induce the core melt accident are LOCA and transient, which can be viewed in the plant damage state diagram in the previous section. Considering different scenarios of containment heat removal situations, different accident progression paths can be generated in the containment event tree. When there is a large release of corium meanwhile the containment safety system does not function well, it is likely to cause early containment failures. Those early containment failures are generally perceived as major severe accidents because of their huge impact on containment integrity. If the containment safety system functioned to mitigate the accident progression, there are also two scenarios that probably occur. One is the accident stops before the penetration through the containment and the other is the late containment failure. When there is a small amount of corium release, the release impact is less severe compared with the large amount release events or missile events. If the incidents are controlled in time, the damage and radioactive substance can be confined within containment, which will cause the least environmental damage among other core melt scenarios. But if the containment safety systems stopped working, even these small release events can still lead to the late containment failure because of the corium accumulation along with the time.

From the above analysis, it can be found containment safety systems play a vital role to mitigate the accident progression once the reactor ruptured. Therefore, in this research, containment safety systems are categorized into three different bins based on their primary function, which include spray systems, hydrogen mitigation systems, and filtered venting systems. By investigating the accident progression under different safety systems, it can provide more detailed insights on different radioactive release scenarios. Specific functions of safety systems have been widely discussed in the previous chapter and corresponding accident types can also be checked. By following through different event paths in the containment event tree, 22 end events can well represent different release scenarios. These release scenarios could be further used to investigate their environmental impact regarding different release amount and probability.

3.5 Release Category Identification

Release category identification served as the interface between the accident progression analysis and the source term analysis. In the previous section, there are 22 containment events obtained from containment event tree analysis. Based on accident progression characteristics, those events could be further classified into five different release categories for investigation, which are containment intact, early containment failure, late containment failure, bypass failure, and isolation failure. It is also imperative to point out that not all severe accidents after the core melt would lead to containment failure. NUREG-1150 report has shown that most PWRs have relatively high probability to remain intact after the severe accident (USNRC, 1990). The grouping scheme of CET events and detailed description regarding release categories can be viewed as below.

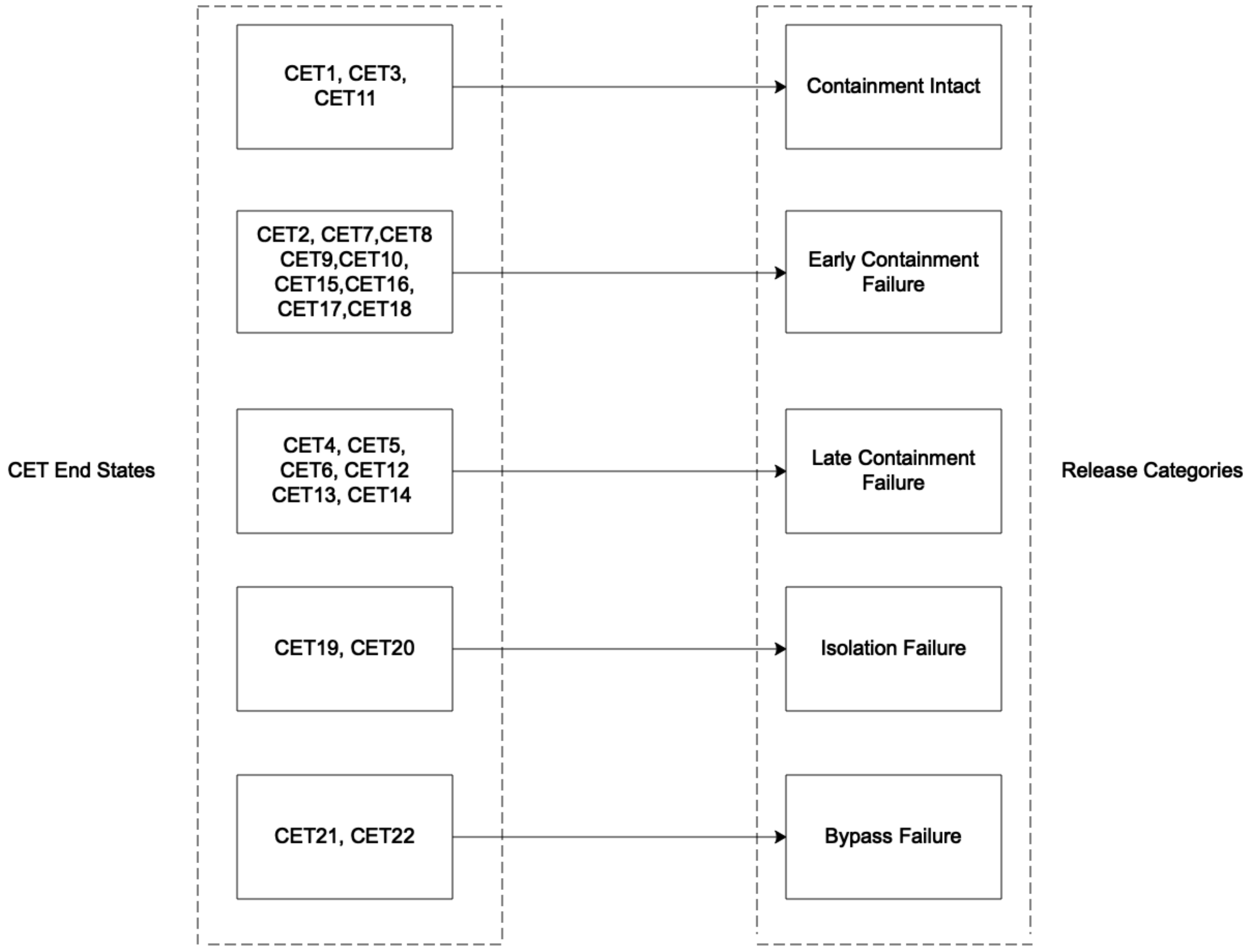

Figure 11. Containment source term release category grouping

3.5.1 Early Containment Failure

Early containment failure is one of the important containment failures leading to the large release of radioactive substances. It is the failure that occurred right after the core melt through the reactor vessel. In general, the early containment failure is perceived very dangerous because once it happened, it allows very short time to initiate the emergency measures, which also made it difficult to confine the radioactive release within the containment. The early containment failure can be induced by direct containment heating (DCH), steam explosions and hydrogen burn.

Direct Containment Heating

Direct containment heating is one of the primary causes that result in containment failure because it will produce the radioactive release directly discharged into the environment and it also gives very short time interval for emergency action. The heat exchange between metal particles and containment atmosphere and the hydrogen release are accounted as major contribution to this phenomenon (Bal, 2012). Additionally, in-vessel pressure is a key parameter that influences the consequence of direct containment heating when the reactor vessel failed. If the in-vessel pressure is much higher than the containment pressure, for example, when small LOCA or station blackout accidents happened, it is likely to induce the high-pressure melt ejection into the reactor cavity. The melt could also be quickly transported out of the cavity into the containment. Due to the oxidation of melt debris, the heat produced can immediately pressurize the containment as the containment is a constant volume system. These complex processes above are known as the DCH and it is perceived as one of big threats for containment integrity. Once DCH happened, it will lead to the containment pressurization.

Steam Explosion

Steam explosion, also known as the alpha mode failure, is one of early containment failure incidents. Because of the rapid fragmentation of molten fuel, the released energy quickly transferred to the coolant, which could lead to the steam generation and shock waves. If there is a large amount of molten fuel getting fragmented in a very fast way within the confined reactor pressure vessel, missiles will get generated. The energy released from those explosions could damage the containment and result in a direct early release of radioactive substances. However, based on experts' judgments, the probability of in-vessel explosion is quite low so that it is not considered as a credible threat for containment failure. Another scenario is the ex-vessel steam explosion, which is aroused due to the incident that the molten fuel dropped into the water outside from the vessel. The generated shock wave can progress through the water and damage the containment mechanical structure. But the damage scenarios varied a lot among different plant types and it seems these potential explosion scenarios have less impact on PWR large dry containment.

Hydrogen Combustion

Hydrogen combustion is another big threat to containment integrity because it can result in the rapid temperature and pressure increase in the containment. Once the increase exceeds the containment design boundary, the radioactive substances will get released to the environment. In the severe accident scenarios, hydrogen is produced through the oxidation process of metallic core debris. During the degraded core accident, the major hydrogen resource comes from the steam and zirconium oxidation. Especially when the reactor core is uncovered, the zirconium gets heated to very high temperature, the hydrogen concentration can quickly climb up. For the detailed description regarding hydrogen generation, it also can be viewed in chapter 3 hydrogen mitigation sections.

In addition to the hydrogen generation, hydrogen mixing and transport mechanisms are also crucial for the combustion requisite because different transport types can result in different combustion scenarios. For example, the quick and rapid mixing will lead to the even distribution of the gaseous mixtures and the burns are widespread around the containment. However, if the mixing process is relatively slow, it is likely to have localized gaseous mixture and localized burning. Therefore, the release rate of the hydrogen impacts a lot on the hydrogen combustion type. Once the gaseous mixture got ignited, the hydrogen combustion would pose a great threat to the containment wall because of the overheating and overpressurization effect.

Deflagration and detonation are two common types of hydrogen combustion. Deflagration is the phenomenon that unburned gases get quickly heated up due to the conduction and the generated combustion waves travelled subsonically. It generally created static loads on the surrounding structures. But it still requires the ignition source to initiate it. In PWR containments, the common random ignition sources are sparks from the electrical system or weak static charges. Detonation is another phenomenon that the unburned gaseous mixture gets quickly compressed and heated up. The generated combustion waves traveled supersonically and it can bring both static and dynamic loads on structures. Direct initiation and flame acceleration are two mechanisms that produce detonation. But direct initiation normally needs high energy to start, for the containment structure, it is very hard to induce this amount of energy. Therefore, in severe accidents, only flame acceleration is likely to result in detonation. As flame acceleration is dependant on the system geometry, the detonation likelihood also varied a lot due to different containment designs. But still, hydrogen burning is one of the big concerns for containment integrity during the severe accidents.

3.5.2 Late Containment Failure

If several mitigation systems functioned to prevent the early containment failure happening, late containment failures can still challenge the containment integrity. Late containment failure referred to those long-term failures after core molten debris released from the reactor vessel. Gradual/ slow overpressurization, late combustible gases burning, and basemat melt through are three typical types of late containment failures if the containment heat removal systems stopped working or venting systems failed to balance the pressure.

Slow pressurization can result in containment failure because additional steam and non-condensible gases emitted. The major source for non-condensible gases comes from the molten core and concrete interaction (MCCI). In the reactor cavity, the core debris interacted with concrete in the containment and then released steam and carbon dioxide. MCCI is complex progress, which depends on multiple factors ranging from the water outside the vessel to the debris coolability. If the cavity is dry initially, the core debris could remain hot to continue interacting with the concrete and non-condensible gases get emitted. If the cavity is flooded before the vessel rupture, the scenario will be different because the fallen molten debris could be cooled and fragmented quickly and coolable crusts get formatted, which could reduce the gas emission. Because boiling water absorbs all the decay heat, the core concrete interaction can be prevented. However, if the water flow can not be continuously provided, later on, the core debris would still have the possibility to interact with concrete. Compared with early containment failure, it is clear that the consequence made by the late containment failure is relatively small except for the underground water contamination due to the basemat melt-through (Bal, 2012). The large amount of fission products can be dissolved in the containment water when spray systems functioned. When the containment basemat melt through occurred, it is likely to release the fission products into the underground water.

3.5.3 Bypass Accident

Containment bypass accidents refer to those scenarios that even through the containment building is intact, fission products can still escape to the environment. These events happened when the primary coolant or fission products released without being processed by containment safety systems including combustible gas control and radionuclides management. Two accidents are typical bypass events. One is steam generator tube rupture (SGTR) and interfacing system loss of coolant accident (ISLOCA) is the other one.

SGTR

Steam Generator Tube Rupture (SGTR) is the accident that the heat transfer tubes rupture resulting in the radioactive release bypassed the containment system. It is characterized as the high frequency of occurrence and big consequence of radioactive release (Zhang & Chen, 2017). The shear fracture will lead to the rupture of multiple tubes and a big amount of flow will pass through the rupture because of the pressure difference between primary and secondary system. As the secondary system has a relatively slow flow rate and lower pressure, the water in the secondary system can easily get contaminated. Therefore, due to the SGTR events, the radioactive substances could release into the environment. As discussed in previous chapters, there are two types of SGTR accidents that could result in core damage. In order to estimate the radioactive release from these two types of events, two factors need to be taken into consideration. One is the core fraction released to the secondary system of the steam generator. The other is the fraction released to the environment through the relief valves or atmospheric steam dump valves (Song, et al., 2019).

ISLOCA

ISLOCA is the event that the pipeline linked to the reactor coolant system rupture outside of the containment or low pressure piping isolation valves fail. Because the interfacing between reactor coolant system and low pressure supporting system failed, the supporting system could be soon overpressurized resulting in rupture outside the containment. This accident could cause the direct release to the environment or auxiliary building. The typical characteristics of ISLOCA are the safety injection is impossible during the recirculation phase and there are potential risks that fission products can be released to the environment if no further RCS isolation applied. Therefore, it is imperative to increase the reliability of interfacing provisions between reactor coolant systems and the low pressure supporting system in order to prevent bypass accidents initiation.

3.5.4 Isolation Failure

When the accident happened, the containment isolation systems are designed to confine the potential radioactive release and prevent its release to the environment. The containment

isolation systems consist of valves, actuators, filters, and piping. The typical isolation failures are the containment breach and malfunction of final release prevention barriers.
The failure can be induced by either mechanical errors (e.g. isolation valves failed to open and close) or human errors (valves forgot to be closed). Once the radioactive substances leakage rate exceeds the expected value, it would cause big damage to the public and the environment.

3.6 Source Term Analysis

Source term analysis is one of the important components in Level 2 PSA regarding the potential release scenarios of radioactive substances. As the interface between level 2 PSA and Level 3 PSA consequence analysis, it depicted various release categories and their release consequences. As one of the key parameters for release consequence, the release amount of fission products from nuclear plants needs to be well considered, which includes the isotopes and fractions of these inventories. When cladding failure happened, the fission gases and other volatile fission products get released to the primary system along with the accident progression. With the core degradation, the volatile fission products from the fuel rods and pellets progressively escaped from the core soon to the system. When the core starts to melt, nearly all of these volatile products get released to the containment. The escaped volatile fission gases include krypton [Kr] and xenon [Xe]. The major volatile fission products are iodine [I], caesium [Cs], bromine [Br], rubidium [Rb], tellurium [Te], Strontium [Sr] and noble metals [Ru].

But regarding the environmental impact of these fission products release, it mainly depends on the physical and chemical conditions of fission products transfer through the reactor. The physical form of fission products and their chemical form are two dominant factors. During the fuel degradation, the mass of fission products released to the containment is probably high. Taking 900 MWe PWR as an example, it can go up to 1500 Kg (IRSN, 2015). But after agglomeration and sediment, those aerosols started to decrease quickly. Among those volatile fission products, Iodine needs to be given special attention because of its radiological consequence. After the core melt accident, the main physical form of iodine is the gaseous iodine (I_2), gaseous organic iodine ([CH_3I]), and caesium iodide ([CsI]). When the particle and gas iodine is released through the primary system to the containment, the gaseous molecular iodine can be quickly adsorbed by the containment wall and after the short reaction, the gaseous organic iodine gets emitted, which is hardest for the existing filtration system to trap (IRSN, 2015). If organic iodine converted to iodine oxides and mixed with the containment sump water, it may have the chance to get released outside the containment because it is not deposited. Based on the investigation performed by NRC (1995), the major radioactive chemical forms are summarized as below.

Radionuclide cluster	Elements	Representative element
Noble gases (most quickly, 1hr)	Xe, Kr	Xe
Halogens	I	I
Caesium	Cs	Cs
Chalcogens	Te, Sb, Se	Te
Alkaline earths	Ba, Sr	Sr
Noble metals	Ru, Rh, Pd, Mo,	Ru

Table 7. Summarized radionuclide group for source term analysis (NUREG-1465, USNRC, 1995)

3.6.1 TMI Unit1 Level 2 PSA Release Category Grouping

After the identification of major source term for radioactive release, it is imperative to analyze their release fractions under different release scenarios in order to better estimate the release consequences. Hereby, the data from level 2 PSA on TMI unit 1 are taken as the fraction data source to generate the simplified approach of radionuclide release consequence. The TIM Unit 1 level 2 PSA was performed in 2007 in order to improve the risk analysis regarding pressurized water reactor containment response during the accident progression. In general, TMI unit 1 is placed in the category of PWR large dry containment (USNRC, 2007). In the report, it includes the failure probability of systems, plausible accident scenarios with the estimated frequency. Moreover, it also investigated the core degradation physical process and fission products release category so that the accident consequence can be obtained. In the release category section, fission product release amount and accident progression were analyzed. All accident sequences which greatly contributed to the core damage were taken into release categories through grouping. In total, it created nine release categories for PWR plants. For each release category, they have multiple different event sequences within the category. The detailed description of sub-event sequences for each release category can be viewed in

Appendix A. The summarized 9 PWR release categories based on TMI Unit 1 level 2 PSA can be viewed in the table below:

Release Categories	Summarized Description
PWR1	Containment bypass with auxiliary building bypass
PWR2	Interfacing-systems LOCA (ISLOCA)
PWR3	Large isolation failures
PWR4	Small isolation failures
PWR5	Early containment failure
PWR6	Late containment failure (large)
PWR7	Late containment failure (small)
PWR8	Basemat melt-through
PWR9	Containment is intact, no containment failure

Table 8. Summarized release category description of TMI Unit 1 level 2 PSA (TMI-PRA_015.2, USNRC, 2007)

In order to better calibrate the fraction data under nine release categories, each release category proposed by TMI Unit 1 level 2 PSA has been compared with the release categories which have been proposed by this research. In this research, five release categories are proposed, which includes: the containment intact, early containment failure, late containment failure, isolation failure and bypass failure. Based on the description of nine release categories, they can be further calibrated and grouped into these five release categories. For early containment failure, PWR5 is RC1. For late containment failure, PWR6, PWR7, and PWR8 are grouped as RC2. PWR3 and PWR4 are characterized by isolation failure, which is grouped as RC3. For RC4, it stands for bypass failure which is characterized by PWR1 and PWR2. RC5 is the containment intact scenario, which is depicted by PWR9.

Proposed Release Category	Description	TMI Summarized Release Category
RC1	Early containment failure	PWR5
RC2	Late containment failure	PWR6, PWR7, PWR8
RC3	Isolation Failure	PWR3, PWR4
RC4	Bypass Failure	PWR1, PWR2
RC5	Containment Intact	PWR9

Table 9. Release category mapping

After categorizing the TMI Unit 1 level 2 PSA release categories into the research proposed release categories, corresponding radionuclide release fractions also need to be calibrated and fit in. Hereby, the method to calculate each release category inventory fractions has been proposed. By taking original TMI Unit 1 level 2 PSA release scenario occurrence frequency as the weighting scheme, the expected value of fraction for each new release category can be obtained. The below equations depicted the proposed method:

$$E[X] = \sum_{i=1}^{n} x_i p_i \;=\; x_1 p_1 + x_2 p_2 + \cdots + x_n p_n$$

$$p_i = \frac{p_j}{\sum_{j=1}^{n} p_j}$$

E[X] = the fraction of new release category

x_i = the original fractions of each radionuclide group

p_j = the occurrence frequency of original TMI Unit 1 level 2 PSA release category

n = the number of TMI Unit 1 level 2 PSA release category within the new RC.

Before the implementation of the calibration method, the first step is to get the frequency data and inventory fraction data for summarized 9 release categories. However, based on the data source provided by TMI Unit 1 level 2 PSA, only inventory fractions of noble gas for summarized release categories were provided. The rest inventory fractions and occurrence frequency were only for sub release events, which are sub-components of the release category. Thus, in order to get the inventory fractions and occurrence probability for each release category, a method based on the frequency contribution of each sub release event has been come up with. The calculated results for each release category can be viewed as below and the detailed calculation can be viewed in Appendix B.

Release Category	Frequency	Release Group (Fraction)					
		Noble Gases (Xe)	Iodine (I)	Caesium (Cs)	Tellurium (Te)	Strontium (Sr)	Ruthenium (Ru)
PWR1	2.05E-06	1.00E+00	1.09E-02	1.09E-02	1.64E-03	7.58E-06	3.96E-05
PWR 2	1.937E-07	9.20E-1	8.50E-01	8.50E-01	1.71E-01	8.58E-02	6.30E-01
PWR 3	6.59E-10	1.00E+00	1.30E-01	1.30E-01	5.95E-02	3.96E-03	5.77E-02
PWR 4	3.78E-07	8.30E-01	1.24E-02	1.40E-02	1.66E-02	2.73E-04	5.44E-03
PWR 5	9.05E-07	1.00E+00	1.57E-02	1.55E-02	8.70E-03	6.33E-05	1.08E-03
PWR 6	1.16557E-07	1.00E+00	3.33E-02	3.33E-02	6.00E-03	1.67E-06	6.07E-05
PWR 7	1.26E-06	7.00E-01	3.86E-04	7.71E-04	3.54E-04	7.71E-07	7.71E-06
PWR 8	3.19E-06	3.00E-01	3.00E-02	2.00E-05	1.40E-06	2.50E-07	7.00E-06
PWR 9	1.44E-05	1.00E-03	1.30E-06	1.30E-06	4.84E-07	1.23E-08	2.43E-07

Table 10. Calculated frequency and release fractions for TMI Unit 1 level 2 PSA

Based on the proposed method, the calibrated fraction of radioactive release group for new release categories can be viewed in the table below. For the detailed calculation of each radioactive group of new release categories can be viewed in Appendix C.

Release category	Release group (Fraction)					
	Noble Gases (Xe)	Iodine (I)	Caesium (Cs)	Tellurium (Te)	Strontium (Sr)	Ruthenium (Ru)
RC1 (early containment failure)	1.00E+00	1.57E-02	1.55E-02	8.70E-03	6.33E-05	1.08E-03
RC2 (late containment failure)	5.22E-01	1.60E-02	1.85E-03	4.35E-04	5.62E-07	9.71E-06
RC3 (containment isolation failure)	8.30E-01	1.26E-02	1.42E-02	1.66E-02	2.80E-04	5.53E-03
RC4 (containment bypass failure)	9.93E-01	8.34E-02	8.34E-02	1.63E-02	7.41E-03	5.44E-02
RC5 (containment intact)	1.00E-03	1.30E-06	1.30E-06	4.84E-07	1.23E-08	2.43E-07

Table 11. Calibrated release fractions for generic level 2 PSA model

From the release fraction table, it can be found the noble gas is released very quickly as long as the radionuclide release accident occurred. By comparing late containment failure and early containment failure scenarios, the release of fission products Caesium and Tellurium decreased a lot along with the time progression. Among five release categories, it can be found containment bypass failure will lead to the most amount of radionuclide release regarding their release magnitudes. Containment intact scenario has the least amount of radionuclide release. However, considering the health and environmental impact, Iodine should be paid more attention. In these release categories, the fractions of iodine did not show big variance.

3.6.2 Release Frequency Quantification

In addition to the identification of source term fractions under different release category scenarios, it is imperative to analyze the occurrence frequency of each release category so that the consequences for different release paths can be analyzed. The calculated consequences can be further used for level 3 PSA to further evaluate the environmental and health impact in case of severe accidents. However, in order to calculate the occurrence frequency for each release category, the basic event occurrence frequencies of containment event tree need to be calculated. Hereby, component failure rate and event frequency are calibrated based on the data collected from the literature review, the detailed frequency calibration method can be viewed in Appendix D. The below table describes the occurrence frequency of containment end states in the containment event tree.

Containment events	Frequency(/RY)	Description
CET1	3.70E-05	No containment failure
CET2	1.46E-7	Early containment failure-Alpha mode
CET3	1.30E-06	No containment failure, with radioactive release, with filtered venting
CET4	1.83E-09	Late containment failure, with large radioactive release, without filtered venting
CET5	1.32E-08	Late containment failure, with large radioactive release, without hydrogen igniter/recombiner, but with filtered venting
CET6	1.85E-11	Late containment failure, with large radioactive release, without hydrogen igniter/recombiner and filtered venting
CET7	1.45E-07	Early containment failure, with large radioactive release, without spray, but with hydrogen igniter/recombiner and filtered venting
CET8	2.03E-10	Early containment failure, with large radioactive release, without spray and filtered venting, but with hydrogen igniter/recombiner
CET9	1.46E-09	Early containment failure, with large radioactive release, without spray and hydrogen igniter/recombiner, but with filtered venting

CET10	2.05E-12	Early containment failure, with large radioactive release, without any containment safety system function
CET11	1.17E-05	No containment failure, with small radioactive release, with fitered venting
CET12	1.64E-08	Late containment failure, with small radioactive release, without filtered venting
CET13	1.19E-07	Late containment failure, with small radioactive release, without hydrogen igniter/recombiner, but with filtered venting
CET14	1.66E-10	Late containment failure, with small radioactive release, without hydrogen igniter/recombiner and filtered venting
CET15	1.30E-06	Early containment failure, with small radioactive release, without spray, but with hydrogen igniter/recombiner and filtered venting
CET16	1.83E-09	Early containment failure, with small radioactive release, without spray and filtered venting, but with hydrogen igniter/recombiner
CET17	1.32E-08	Early containment failure, with small radioactive release, without spray and hydrogen igniter/recombiner, but with filtered venting
CET18	1.85E-11	Early containment failure, with small radioactive release, without any containment safety system function
CET19	1.68E-07	Containment rupture due to the accident progression, with radioactive release
CET20	3.78E-07	Containment not isolated at the begining
CET21	1.83E-06	ISLOCA
CET22	2.47E-06	SGTR

Table 12. Calibrated CET end states frequency and their brief descriptions

3.6.3 Results and Discussion

After calculating the occurrence frequency for each containment end state, the frequency of release category can be obtained based on the release category grouping scheme proposed in previous chapters. By adding up corresponding containment end sates frequency within the same category, 5 release categories frequency can be viewed in the table below and detailed calculation can be found in Appendix D.

Release category	Frequency (yr)	Release group (Fraction)					
		Noble Gases (Xe)	Iodine (I)	Caesium (Cs)	Tellurium (Te)	Strontium (Sr)	Ruthenium (Ru)
RC1 (early containment failure)	1.61E-06	1.00E+00	1.57E-02	1.55E-02	8.70E-03	6.33E-05	1.08E-03
RC2 (late containment failure)	1.17E-05	5.22E-01	1.60E-02	1.85E-03	4.35E-04	5.62E-07	9.71E-06
RC3 (containment isolation failure)	5.46E-07	8.30E-01	1.26E-02	1.42E-02	1.66E-02	2.80E-04	5.53E-03
RC4 (containment bypass failure)	4.30E-06	9.93E-01	8.34E-02	8.34E-02	1.63E-02	7.41E-03	5.44E-02
RC5 (containment intact)	5.00E-05	1.00E-03	1.30E-06	1.30E-06	4.84E-07	1.23E-08	2.43E-07

Table 13. The calibrated release frequency and inventory fractions for generic level 2 PSA model

Based on the calculated results of different release scenarios, it can be found the most likely scenario is the event that containment can still remain intact after the severe accident. Also, late containment failure is much more likely to happen compared with early containment failure. In order to have better insight regarding the frequency of release category, the release category frequencies from studies such as WASH-1400, French 1300 MWe level 2 PSA, and TMI Unit 1 level 2 PSA were taken to compare with those obtained from a generic level 2 PSA model in this research. WASH-1400, known as The Reactor Safety Study, is a systemic risk analysis report for the light water reactor safety assessment. In the report, it has 9 release categories including early containment failure, late containment failure, etc. Under each release category, the occurrence frequency and release fractions were calculated. Table 13 summarized the 9 release categories and their descriptions.

Release categories	Summarized description
WH1	Very early containment failure, alpha mode explosion
WH 2	Early containment failure, hydrogen burn, steam explosion
WH 3	Similar to PWR1 and PWR2, but partial success of radioactivity removal systems
WH 4	Isolation failure core melt, radioactivity removal system off
WH 5	Isolation failure core melt, radioactivity removal system on
WH 6	Late containment failure, core melt through basemat, radioactivity removal system on
WH 7	Late containment failure, core melt through basemat, radioactivity removal system on
WH 8	Early containment leakage without containment failure at the beginning.
WH 9	Only some of release from gaps, containment is intact

Table 14. Summarized release categories in WASH-1400 (USNRC, 1975)

Table 14 was taken from WASH-1400 to present the release frequency and radionuclide release fractions of each release category. In WASH-1400, large dry PWR containment was investigated to provide a general overview of release scenarios and their consequences.

Release Category	Probability per reactor year	Energy Release (10^6 Btu/Hr)	Release Group (Fraction)					
			Noble Gases (Xe)	Iodine (I)	Caesium (Cs)	Tellurium (Te)	Strontium (Sr)	Ruthenium (Ru)
WH 1	9×10^{-7}	520	0.9	0.706	0.4	0.4	0.05	0.4
WH 2	8×10^{-6}	170	0.9	0.707	0.5	0.3	0.06	0.02
WH 3	4×10^{-6}	6	0.8	0.206	0.2	0.3	0.02	0.03
WH 4	5×10^{-7}	1	0.6	0.092	0.04	0.03	5×10^{-3}	3×10^{-3}
WH 5	7×10^{-7}	0.3	0.3	0.032	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}
WH 6	6×10^{-6}	N/A	0.3	2.8×10^{-3}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}
WH 7	4×10^{-5}	N/A	6×10^{-3}	4×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}
WH 8	4×10^{-5}	N/A	2×10^{-3}	1.05×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0
WH 9	4×10^{-4}	N/A	3×10^{-6}	1.07×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0

Table 15. Release frequency and inventory fraction of WASH-1400 (USNRC, 1975)

From the data provided, it can be seen that the calibrated results in this research generally comply well with the results from WASH-1400. For example, in WASH-1400, release categories WH1, WH2, WH3 belong to early containment failure scenario and the summed frequency to have early containment failure is around 10^{-6} /yr, which shares the similar order of magnitude with the proposed RC1 early containment failure. Besides, WH4 and WH5 are isolation failure scenarios in WASH-1400 and they also share the same order of magnitude with RC3, the isolation failure in the generic level 2 PSA model. Additionally, both studies showed the containment is likely to maintain its integrity after the severe accident. From the comparison results, it can be concluded that the results obtained by using a generic level 2 PSA model could represent the results obtained by using large event trees.

In addition to WASH-1400, the results from level 2 PSA of French 1300 MWe PWR were also taken to cross-compare with research results (Cénérino, et al., 2016). It is found that the frequency of early containment failure and late containment failure in the research are one order of magnitude larger than the data obtained in French 1300 MWe report. For containment bypass and isolation failure scenarios, results from two researches complied with each other well. From the comparison, it can be found the results calculated from the generic model are more conservative than the level 2 PSA results performed by IRSN. But they still can reflect the associated risk and their consequences for the general application of PWR risk analysis. The table below is the summarized release category from French 1300 MWe level 2 PSA report by classifying certain containment failure modes.

Containment failure mode	Classified release category	Frequency/yr
I-SGTR (consequential steam generator tube rupture)	Containment bypass failure	6.67E-07
Reactor containment isolation failure	Isolation failure	3.70E-07
Reactor containment failure after hydrogen combustion during in-vessel phase	Early containment failure	3.32E-07
Reactor containment bypasses (heterogeneous dilutions, initial SGTR, interfacing LOCA)	Containment bypass failure	2.30E-07
Ex-vessel steam explosion	Early containment failure	2.00E-07
Reactor containment failure after direct containment heating	Early containment failure	6.30E-08
Basemat penetration by the corium	Late containment failure	3.00E-06
Long term containment overpressurization	Late containment failure	2.41E-07
Hydrogen combustion followed by secondary containment failure	Early containment failure	4.70E-07

Table 16. Summarized release frequency of French 1300 MWe level 2 PSA (Cénérino, et al., 2016)

Additionally, in order to provide a better comparison insight for the frequency results of generic level 2 PSA, in table 17, the release frequency of all these mentioned studies have been presented together. It can be found the results obtained from the generic event tree model are able to represent the results obtained through large event tree or small event tree models in level 2 PSA studies.

Release category	Release frequency (Generic level 2 PSA)	Release frequency (WASH-1400)	Release frequency (TMI Unit1 level 2 PSA)	Release frequency (French 1300 MWe level 2 PSA)
Early containment failure	1.61E-06	1.29E-05	9.05E-07	5.92E-07
Late containment failure	1.17E-05	4.60E-05	4.57E-06	3.24E-06
Isolation failure	5.46E-07	1.20E-06	3.78E-07	3.70E-07
Containment bypass	4.30E-06	-	2.24E-06	8.97E-07
Containment Intact	5.00E-05	4.00E-04	1.44E-05	-

Table 17. The cross-comparison of release frequency calculation results in different PSA level 2 studies.

However, the main objective to develop this generic model is not about accurate prediction of the release consequence. It is more designed to work as the reference for precursor analysis. By applying generic level 2 PSA model for PWR assessment during design or repairment phase, the potential release paths can be well identified and the consequences can be estimated. These information together at least could provide a preliminary view on risk significant events of the target plants.

Chapter 4 Conclusion & Reflections

This chapter is designed to reflect the whole research structure as well as conclude the final findings during the development of generic level 2 PSA models for pressurized water reactors. As mentioned at the beginning, there is one research question along with four sub-questions designed as guiding stars for the research development. Therefore, after the generic level 2 PSA model development, it is imperative to revisit them in order to see whether the developed model can handle these questions and what answers can be provided based on the research results. After the reflection on research questions, there is another section designed in this chapter aiming at providing the summarized creative points in this research. It was also pointed out in what way these results could contribute to the future development of the study. Last but not least, the limitation of the work and potential development based on the research findings were addressed at the end of this chapter.

4.1 Revisiting the Research Questions

Sub-question1: *What type of nuclear plants should be selected as the reference target?*
This question is designed to narrow down the research scope because level 2 PSA can be tailored and applied for all nuclear plants. However, due to the design and operation difference, different types of reactors may result in different results. Therefore, a reference reactor type is needed as the research basis. As discussed in the section "reference reactor type selection", pressurized water reactors were selected as the investigation targets for generic level 2 PSA model building. The main reasons for selecting PWRs among other types of nuclear plants are as follows:

1. The worldwide dominant nuclear plant type is pressurized water reactor and most current operating nuclear plants are generation II pressurized water reactors. Therefore, if a generic level 2 PSA model can be developed, it will help most of the plants to perform precursor analysis.
2. The basic layout and operation mechanism among PWRs did not vary a lot, which make it possible to explore the common design features to build generic level 2 PSA model.

Sub-question 2: *How to perform level 2 Probabilistic Safety Analysis for reference reactor?*

This question is designed to come up with the research outline and explore the potential research method. In order to build the generic level 2 PSA model, there are several things need to be well considered before analysis. The first thing is to understand what is level 2 PSA and what procedures need to be followed in order to perform level 2 PSA. The second thing is to spot the research gap through the literature review process, which means to summarize what other people did and think about how the generic approach can be applied for level 2 PSA. In chapter 3, the generic level 2 PSA was performed by following the general outline required by IAEA for level 2 PSA analysis. It consists of plant familiarization, plant damage state identification, accident progression analysis, radioactive release category identification, and source term analysis.

Sub-question 3: *Why is it creative to develop a generic model of level 2 PSA?*
Considering that there are a lot of level 2 PSA existed in current studies, this question is designed to challenge the research work creativity. By reviewing and researching level 2 PSA development, it can be found at the beginning, containment event trees are too large to graphically present and among hundreds of end states, it is not easy to find necessary information. Later on, the improvement has been made to reduce the size of containment event trees, but they gradually became very detailed and plant-specific, which made it difficult for other plants to take as reference to implement. Therefore, a generic containment event tree for level 2 PSA is needed so that it could provide the general impression for potential risks of the plant. The major events and release paths can still be well-identified without checking specific details.

Sub-question 4: *How does a generic level 2 PSA model contribute to the risk analysis of nuclear reactors?*
It is designed to question the value of the research in order to compare it with the general level 2 PSA. In chapter 4, the validity of the results obtained from generic level 2 PSA has been cross-compared with other level 2 PSA studies. It was found that the obtained release frequency and inventory fractions shared a similar magnitude with other analyses. That is to say, the results obtained through the generic model are reasonable. Therefore, the generic level 2 PSA could be used as the reference base for nuclear plant risk analysis if no further detailed information provided. By applying the generic level 2 PSA, potential risks and risk significant events can be estimated before detailed investigation. Additionally, it could help those plants

who share similar safety systems to build up their own level 2 PSA through modifying the common prototype of generic level 2 PSA.

Based on the detailed answers provided for above four sub-questions, it is clear that the generic level 2 PSA proposed in this research is the analysis using one generic containment event tree to capture the whole accident progression as well as taking all key plant damage states into consideration. By identifying the potential release paths and estimating corresponding consequences, it could be largely applied for the pressurized water reactor precursor analysis around the world.

4.2 Creative Points in the Research

The ultimate goal of this research is to develop a generic level 2 PSA model for pressurized water reactor. Through the literature review, it was found current level 2 PSA studies are all based on the specific type of plants. The results obtained from these specific plants maybe not very valuable for other plants to take as the reference. Therefore, the first creative point in this research is the development of a generic level 2 PSA. The results obtained from this generic model can well reflect the results got from those detailed analyses.

The second creative point in this research is the summarization and classification of different containment safety systems among different PWR generation designs. In order to make the study comprehensive, various PWRs in different countries have been studied. The classification tables can be good references for future containment safety system studies. It provides an overview of the major functions of containment safety systems as well as the implementation status quo across the world.

The third creative point is the establishment of generic containment event tree. As mentioned in previous chapters, the containment event tree is one of the key elements in level 2 PSA. Thus, the quality of level 2 PSA highly depends on the quality of the containment event tree. In other studies, each plant damage state has its own corresponding containment event trees. Therefore, if there were 40 plant damage states, there would be 40 containment event trees, which could lead to a huge number of event tree end sates. It also makes it very difficult for non-experts to quickly understand release paths in the plant by going through 40 event trees. But in the research, a generic containment event tree was built up by taking various initiating events into consideration. Different plant damage states have been considered in order to come up with top event attributes. Therefore, by just using one generic containment event tree, different release scenarios can be represented, which greatly reduced the complexity and redundancy among large number of end state events.

The fourth creative point is the development of simplified release categories, which made the source term release paths more structuralized under different accident scenarios. It includes containment bypass failure, containment isolation failure, early containment failure, late containment failure and containment intact. In order to group corresponding generic event tree end states into the proposed release category, the grouping scheme based on the characteristics of each containment event paths was created. It helped to classify different release scenarios into clusters so that a systemic view of potential release accidents can be provided.

The last but not least creative point is the selection of pressurized water reactor as the reference reactor. Pressurized water reactors are major operating reactors all over the world. Therefore, by selecting pressurized water reactor as the research target, it could benefit most of the plants for further developing level 2 PSA. Moreover, in the research, different generations of PWR safety systems have been widely compared, which could be helpful for future studies on improvements of safety systems.

4.3 Limitations and Future Work

Although the generic level 2 PSA for pressurized water reactors has been successfully developed and the quantification results could provide the estimated value for different release categories, there are still several limitations in this research. In the plant damage state analysis and containment event tree development, all accidents are assumed happened during the full power operation. For the containment safety system inspection, because all safety systems are considered at the system level and they are all assumed independent from each other, the data accuracy issues for the reliability of these systems need to be well considered in the future. Also, during the establishment of the plant damage state logic diagram, only key attributes for the accident progression have been taken into consideration in order to prepare the generic containment event tree. Besides the structure design issues, there are also some limitations in data analysis. For example, in the inventory fraction calculation, the radioactive elements are simplified by just using the key elements to represent. For the frequency calculation, the plant damage state frequency, top events, and system failure rate are also simplified by just taking the mean value obtained from the literature. Although the results obtained from the current generic level 2 PSA roughly complied with other level 2 PSA studies, it still has a lot of uncertainties that need to be well considered. Therefore, if better data source can be provided, it can better improve the release consequence estimation.

As mentioned above, there are several limitations in the research, therefore, future work can be developed to tackle these issues. The first aspect that future work can be developed is to expand the generic level 2 PSA for other type reactors, for example, boiling water reactors and heavy water reactors. By investigating the design and system differences between pressurized water reactors and other type reactors, the generic level 2 PSA model can be modified based on the requirements. If these types of reactors can be studied, it will cover over 90% of nuclear power plants throughout the world and they could use the generic level 2 PSA as the pre-assessment for the plant risks. The second aspect of future work development is to push forward the current generic models for pressurized water reactors. For instance, if the reactor shut-down scenario also can be taken into consideration, the generic model will be more comprehensive. Moreover, if the interdependency among containment safety systems can be studied, it will better improve the frequency quantification. Also, if the data uncertainty issues regarding the plant damage state frequency, top event and component failure rate can be better managed, the accuracy of the consequence can be largely improved. Last but not least, if the future studies aim to add the granularity of the event tree, it will be another option by adding more necessary key attributes for both plant damage logic diagram and containment event tree.

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Appendix A: Descriptions of Release sub-events in TMI Unit 1 level 2 PSA

Release sub-event sequences	Release category	Brief description
1-01	PWR1	Containment bypass, outside the auxiliary building, without ex-vessel radioactive release, with fission product scrubbing
1-02	PWR1	Containment bypass, outside the auxiliary building, without ex-vessel radioactive release, without fission product scrubbing
2-01	PWR2	Containment bypass, to the auxiliary building, without ex-vessel radioactive release, with fission product scrubbing
2-02	PWR2	Containment bypass, to the auxiliary building, without ex-vessel radioactive release, without filtered venting
2-03	PWR2	Containment bypass, to the auxiliary building, with ex-vessel radioactive release, with filtered venting
2-04	PWR2	Containment bypass, to the auxiliary building, with ex-vessel radioactive release, without filtered venting
3-01	PWR3	Large isolation failure, to the auxiliary building, without ex-vessel radioactive release, with filtered venting
3-02	PWR3	Large isolation failure, to the auxiliary building, without ex-vessel radioactive release, without filtered venting
3-03	PWR3	Large isolation failure, to the auxiliary building, with ex-vessel radioactive release, with filtered venting
3-04	PWR3	Large isolation failure, to the auxiliary building, with ex-vessel radioactive release, without filtered venting
3-05	PWR3	Large isolation failure, outside the auxiliary building, without ex-vessel radioactive release
3-06	PWR3	Large isolation failure, outside the auxiliary building, with ex-vessel radioactive release
4-01	PWR4	Small isolation failure, to the auxiliary building, without ex-vessel radioactive release, with filtered venting
4-02	PWR4	Small isolation failure, to the auxiliary building, without ex-vessel radioactive release, without filtered venting
4-03	PWR4	Small isolation failure, to the auxiliary building, with ex-vessel radioactive release, with filtered venting
4-04	PWR4	Small isolation failure, to the auxiliary building, with ex-vessel radioactive release, without filtered venting
4-05	PWR4	Small isolation failure, to the environment, without ex-vessel radioactive release, with fission product scrubbing
4-06	PWR4	Small isolation failure, to the environment, without ex-vessel radioactive release, without filtered venting
4-07	PWR4	Small isolation failure, to the environment, with ex-vessel radioactive release, without filtered venting
4-08	PWR4	Small isolation failure, to the environment, with ex-vessel radioactive release, without filtered venting
5-01	PWR5	Early containment failure, without ex-vessel radioactive release
5-02	PWR5	Early containment failure, with ex-vessel radioactive release

Table 18. The overview of release sub-event description from TMI Unit 1 level 2 PSA (TMI-PRA_015.2, USNRC, 2007)

Sub-release event sequences	Release category	Brief description
6-01	PWR6	Late overpressurization, with large containment failure, without ex-vessel radioactive release, without re-vaporization, with filtered venting
6-02	PWR6	Late overpressurization, with large containment failure, without ex-vessel radioactive release, without re-vaporization, without filtered venting
6-03	PWR6	Late overpressurization, with large containment failure, without ex-vessel radioactive release, with re-vaporization, with fission product scrubbing
6-04	PWR6	Late overpressurization, with large containment failure, without ex-vessel radioactive release, with re-vaporization, without filtered venting
6-05	PWR6	Late overpressurization, with large containment failure, with ex-vessel radioactive release, without re-vaporization, with filtered venting
6-06	PWR6	Late overpressurization, with large containment failure, with ex-vessel radioactive release, without re-vaporization, without filtered venting
6-07	PWR6	Late overpressurization, with large containment failure, with ex-vessel radioactive release, with re-vaporization, with filtered venting
6-08	PWR6	Late overpressurization, with large containment failure, with ex-vessel radioactive release, with re-vaporization, without filtered venting
7-01	PWR7	Late overpressurization, with small containment failure, without ex-vessel radioactive release, with filtered venting
7-02	PWR7	Late overpressurization, with small containment failure, without ex-vessel radioactive release, without filtered venting
7-03	PWR7	Late overpressurization, with small containment failure, with ex-vessel radioactive release, with filtered venting
7-04	PWR7	Late overpressurization, with small containment failure, with ex-vessel radioactive release, without filtered venting
8-01	PWR8	Containment failure from basemat melt-through, with ex-vessel radioactive release
9-01	PWR9	No containment failure, without ex-vessel radioactive release, with filtered venting
9-02	PWR9	No containment failure, without ex-vessel radioactive release, without filtered venting
9-03	PWR9	No containment failure, with ex-vessel radioactive release, with filtered venting
9-04	PWR9	No containment failure, with ex-vessel radioactive release, without filtered venting

Table 19. The overview of release sub-event description from TMI Unit 1 level 2 PSA (TMI-PRA_015.2, USNRC, 2007)

Appendix B: Data Processing for TMI Unit 1 Level 2 PSA

B.1 Release sub-event frequency and inventory fraction in TMI Unit 1 level 2 PSA

The frequency contribution and release fraction tables for release sub-event taken from TMI Unit 1 level 2 PSA report (USNRC, 2007)

Sub-release event	Release category	Frequency (1/YR)
1-01	PWR1	4.57E-07
1-02	PWR1	1.59E-06
2-01	PWR2	0
2-02	PWR2	1.81E-07
2-03	PWR2	0
2-04	PWR2	1.27E-08
3-01	PWR3	9.07E-11
3-02	PWR3	9.07E-11
3-03	PWR3	1.90E-10
3-04	PWR3	2.88E-10
3-05	PWR3	0
3-06	PWR3	0
4-01	PWR4	3.9E-08
4-02	PWR4	1.46E-08
4-03	PWR4	8.54E-09
4-04	PWR4	3.16E-07
4-05	PWR4	0
4-06	PWR4	0
4-07	PWR4	0
4-08	PWR4	0
5-01	PWR5	7.39E-07
5-02	PWR5	1.66E-07
6-01	PWR6	0
6-02	PWR6	0
6-03	PWR6	2.2E-08
6-04	PWR6	2.36E-10
6-05	PWR6	2.08E-11
6-06	PWR6	0
6-07	PWR6	8.0E-08
6-08	PWR6	1.43E-08
7-01	PWR7	2.25E-07
7-02	PWR7	2.75E-09
7-03	PWR7	7.45E-07
7-04	PWR7	2.89E-07
8-01	PWR8	3.19E-06
9-01	PWR9	1.20E-05
9-02	PWR9	1.69E-08
9-03	PWR9	2.36E-06
9-04	PWR9	1.91E-08

Table 20. The overview of release subevent contributions to total frequency taken from TMI Unit 1 level 2 PSA report (TMI-PRA_015.2, USNRC, 2007)

Sub-release event sequences	Release category	Release fractions				
		Iodine (I)	Caesium (Cs)	Tellurium (Te)	Strontium (Sr)	Ruthenium (Ru)
1-01	PWR1	3.5E-03	3.5E-03	1.8E-03	4.3E-06	1.2E-04
1-02	PWR1	1.3E-02	1.3E-02	1.6E-03	8.5E-06	1.7E-05
2-01	PWR2	1.7E-01	1.7E-01	1.8E-01	1.7E-02	1.4E-01
2-02	PWR2	8.5E-01	8.5E-01	9.0E-02	8.5E-02	7.0E-01
2-03	PWR2	1.7E-01	1.7E-01	1.6E-01	1.8E-02	1.4E-01
2-04	PWR2	8.5E-01	8.5E-01	8.2E-01	9.2E-02	7.2E-02
3-01	PWR3	2.6E-02	2.6E-02	2.0E-04	9.0E-04	1.6E-02
3-02	PWR3	1.3E-01	1.3E-01	1.0E-03	4.5E-03	8.0E-02
3-03	PWR3	4.4E-02	4.4E-02	2.4E-02	1.3E-03	1.8E-02
3-04	PWR3	2.2E-01	2.2E-01	1.2E-01	6.5E-03	9.0E-02
3-05	PWR3	1.3E-01	1.3E-01	1.0E-03	4.5E-03	8.0E-02
3-06	PWR3	2.2E-01	2.2E-01	1.2E-01	6.5E-03	9.0E-02
4-01	PWR4	2.0E-03	2.0E-03	2.0E-04	2.4E-05	2.0E-04

Sub-release event sequences	Release category	Release fractions				
		Iodine (I)	Caesium (Cs)	Tellurium (Te)	Strontium (Sr)	Ruthenium (Ru)
4-02	PWR4	1.0E-02	1.0E-02	1.0E-03	1.2E-04	1.0E-03
4-03	PWR4	2.8E-03	3.2E-03	4.0E-03	6.4E-05	1.3E-03
4-04	PWR4	1.4E-02	1.6E-02	2.0E-02	3.2E-04	6.5E-03
4-05	PWR4	2.6E-03	2.8E-03	2.0E-04	5.8E-05	2.0E-04
4-06	PWR4	1.3E-02	1.4E-02	1.0E-03	2.9E-04	1.0E-03
4-07	PWR4	5.0E-03	6.2E-03	7.0E-03	1.0E-06	2.4E-03
4-08	PWR4	2.5E-02	3.1E-02	3.5E-02	6.9E-04	1.2E-02
5-01	PWR5	1.6E-02	1.6E-02	8.0E-03	3.0E-05	9.3E-04
5-02	PWR5	1.4E-02	1.3E-02	1.2E-02	2.2E-04	1.8E-03
6-01	PWR6	8.0E-04	1.8E-03	4.0E-03	1.0E-06	4.0E-06
6-02	PWR6	4.0E-03	9.0E-03	1.0E-04	5.0E-06	2.0E-05
6-03	PWR6	2.0E-02	2.0E-02	2.0E-05	1.0E-06	4.0E-06
6-04	PWR6	1.0E-01	1.0E-01	1.0E-04	5.0E-06	2.0E-05
6-05	PWR6	8.0E-04	1.8E-03	4.0E-03	1.0E-06	4.0E-05
6-06	PWR6	4.0E-03	9.0E-03	2.0E-02	5.0E-06	2.0E-04

Sub-release event sequences	Release category	Release fractions				
		Iodine (I)	Caesium (Cs)	Tellurium (Te)	Strontium (Sr)	Ruthenium (Ru)
6-07	PWR6	2.0E-02	2.0E-02	4.0E-03	1.0E-06	4.0E-05
6-08	PWR6	1.0E-01	1.0E-01	2.0E-02	5.0E-06	2.0E-04
7-01	PWR7	2.0E-04	4.0E-04	2.0E-05	4.0E-07	4.0E-06
7-02	PWR7	1.0E-03	2.0E-03	1.0E-04	2.0E-06	2.0E-05
7-03	PWR7	2.0E-04	4.0E-04	2.0E-04	4.0E-07	4.0E-06
7-04	PWR7	1.0E-03	2.0E-03	1.0E-03	2.0E-06	2.0E-05
8-01	PWR8	3.0E-02	2.0E-05	1.4E-06	2.5E-07	7.0E-06
9-01	PWR9	7.0E-07	7.0E-07	2.0E-09	4.0E-09	2.0E-09
9-02	PWR9	2.0E-05	2.0E-05	1.0E-06	2.5E-07	1.0E-06
9-03	PWR9	4.0E-06	4.0E-06	2.8E-06	5.0E-08	1.4E-06
9-04	PWR9	2.0E-05	2.0E-05	1.4E-05	2.5E-07	7.0E-06

Table 21. The overview of sub-release event release fraction from TMI Unit 1 level 2 PSA (TMI-PRA_015.2, USNRC, 2007)

B.2 Release Frequency and Inventory Fraction Calculation for Summarized Release Category

Summarized release category frequency calculation and release fraction calibration are based on the release sub-event data source, which is taken from TMI Unit 1 level 2 PSA report (USNRC, 2007)

Frequency calculation for summarized release category:

The proposed solution is based on the frequency of release sub-events. If release sub-events belong to the same release category, then add their frequency together to get the total frequency for each release category.

PWR1

4.57E-07+1.59E-06 = 2.05E-06

PWR2

1.81E-07+0+1.27E-08 = 1.937E-07

PWR3

9.07E-11+ 9.07E-11+1.90E-10+2.88E-10= 6.59E-10

PWR4

3.90E-08+1.46E-08+8.54E-09+3.16E-07= 3.78E-07

PWR5

7.39E-07+1.66E-07 = 9.05E-07

PWR6

2.20E-08+2.36E-10+2.08E-11+8.00E-08+1.43E-08 = 1.16557E-07

PWR7

2.25E-07+2.75E-09+7.45E-07+2.89E-07 = 1.26E-06

PWR8

3.19E-06

PWR9

1.20E-05+1.69E-08+2.3E-06 +1.91E-08= 1.44E-05

Release fraction calculation for summarized release category:

The proposed solution is based on the frequency weight of release sub-events. By taking individual frequency contribution to the corresponding release categories into consideration, weighted release fractions for each release sub-events can be obtained. If release sub-events belong to the same release category, then add their weighted release fractions together to get the total release fraction for each release category.

Note: Noble gas fractions were directly taken from TMI Unit 1 level 2 PSA report (USNRC, 2007).

PWR1:

Iodine (I)

$$\frac{4.57\times10^{-7}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 3.5\times10^{-3}+\frac{1.59\times10^{-6}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 1.3\times10^{-2}=1.09\text{E-}02$$

Caesium (Cs)

$$\frac{4.57\times10^{-7}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 2.0\times10^{-3}+\frac{1.59\times10^{-6}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 1.0\times10^{-2}=1.09\text{E-}02$$

Tellurium (Te)

$$\frac{4.57\times10^{-7}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 3.5\times10^{-3}+\frac{1.59\times10^{-6}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 1.3\times10^{-2}=1.64\text{E-}03$$

Strontium (Sr)

$$\frac{4.57\times10^{-7}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 3.5\times10^{-3}+\frac{1.59\times10^{-6}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 1.3\times10^{-2}=7.58\text{E-}06$$

Ruthenium (Ru)

$$\frac{4.57\times10^{-7}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 3.5\times10^{-3}+\frac{1.59\times10^{-6}}{4.57\times10^{-7}+1.59\times10^{-6}}\times 1.3\times10^{-2}=3.96\text{E-}05$$

PWR2:

Iodine (I)

$$\frac{1.81\times10^{-7}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 8.5\times10^{-1}+\frac{1.27\times10^{-8}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 8.5\times10^{-1}=8.50\text{E-}01$$

Caesium (Cs)

$$\frac{1.81\times10^{-7}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 8.5\times10^{-1}+\frac{1.27\times10^{-8}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 8.5\times10^{-1}=8.50\text{E-}01$$

Tellurium (Te)

$$\frac{1.81\times10^{-7}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 9.0\times10^{-2}+\frac{1.27\times10^{-8}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 8.2\times10^{-1}=1.71\text{E-}01$$

Strontium (Sr)

$$\frac{1.81\times10^{-7}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 8.5\times10^{-2}+\frac{1.27\times10^{-8}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 9.2\times10^{-2}=8.58\text{E-}02$$

Ruthenium (Ru)

$$\frac{1.81\times10^{-7}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 7.0\times10^{-1}+\frac{1.27\times10^{-8}}{0+1.81\times10^{-7}+0+1.27\times10^{-8}}\times 7.2\times10^{-2}=6.30\text{E-}01$$

PWR3:

Iodine (I)

$$\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 2.6\times10^{-2}+\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 1.3\times10^{-1}+\frac{1.9\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 4.4\times10^{-2}+\frac{2.88\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 2.2\times10^{-1} = 1.30\text{E-}01$$

Caesium (Cs)

$$\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 2.6\times10^{-2}+\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 1.3\times10^{-1}+\frac{1.9\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 4.4\times10^{-2}+\frac{2.88\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 2.2\times10^{-1} = 1.30\text{E-}01$$

Tellurium (Te)

$$\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 2.0\times10^{-4}+\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 1.0\times10^{-3}+\frac{1.9\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 2.4\times10^{-2}+\frac{2.88\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 1.2\times10^{-1} = 5.95\text{E-}02$$

Strontium (Sr)

$$\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 9.0\times10^{-4}+\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 4.5\times10^{-3}+\frac{1.9\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 1.3\times10^{-3}+\frac{2.88\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 6.5\times10^{-3} = 3.96\text{E-}03$$

Ruthenium (Ru)

$$\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 1.6\times10^{-2}+\frac{9.07\times10^{-11}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 8.0\times10^{-2}+\frac{1.9\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 1.8\times10^{-2}+\frac{2.88\times10^{-10}}{9.07\times10^{-11}+9.07\times10^{-11}+1.9\times10^{-10}+2.88\times10^{-10}}\times 9.0\times10^{-2} = 5.77\text{E-}02$$

PWR4:

Iodine (I)

$$\frac{3.90\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 2.0\times10^{-3}+\frac{1.46\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 1.0\times10^{-2}+\frac{8.54\times10^{-9}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 2.8\times10^{-3}+\frac{3.16\times10^{-7}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 1.4\times10^{-2} = 1.24\text{E-}02$$

Caesium (Cs)

$$\frac{3.90\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 2.0\times10^{-3}+\frac{1.46\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 1.0\times10^{-2}+\frac{8.54\times10^{-9}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 3.2\times10^{-3}+\frac{3.16\times10^{-7}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 1.6\times10^{-2} = 1.40\text{E-}02$$

Tellurium (Te)

$$\frac{3.90\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 2.0\times10^{-4}+\frac{1.46\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 1.0\times 10^{-3}+\frac{8.54\times10^{-9}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 4.0\times10^{-3}+\frac{3.16\times10^{-7}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 2.0\times10^{-2} = 1.66\text{E-}02$$

Strontium (Sr)

$$\frac{3.90\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 2.4\times10^{-5}+\frac{1.46\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 1.2\times 10^{-4}+\frac{8.54\times10^{-9}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 6.4\times10^{-5}+\frac{3.16\times10^{-7}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 3.2\times10^{-4} = 2.73\text{E-}04$$

Ruthenium (Ru)

$$\frac{3.90\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 2.0\times10^{-4}+\frac{1.46\times10^{-8}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 1.0\times 10^{-3}+\frac{8.54\times10^{-9}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 1.3\times10^{-3}+\frac{3.16\times10^{-7}}{3.90\times10^{-8}+1.46\times10^{-8}+8.54\times10^{-9}+3.16\times10^{-7}}\times 6.5\times10^{-3} = 5.44\text{E-}03$$

PWR5:

Iodine (I)

$$\frac{7.39\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 1.6\times10^{-2}+\frac{1.66\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 1.4\times10^{-2} = 1.57\text{E-}02$$

Caesium (Cs)

$$\frac{7.39\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 1.6\times10^{-2}+\frac{1.66\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 1.3\times10^{-2} = 1.55\text{E-}02$$

Tellurium (Te)

$$\frac{7.39\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 8.0\times10^{-3}+\frac{1.66\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 1.2\times10^{-2} = 8.70\text{E-}03$$

Strontium (Sr)

$$\frac{7.39\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 3.0\times10^{-5}+\frac{1.66\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 2.2\times10^{-4} = 6.33\text{E-}05$$

Ruthenium (Ru)

$$\frac{7.39\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 9.3\times10^{-4}+\frac{1.66\times10^{-7}}{7.39\times10^{-7}+1.66\times10^{-7}}\times 1.8\times10^{-3} = 1.08\text{E-}03$$

PWR6:

Iodine (I)

$$\frac{2.20\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 2.0\times10^{-2}+\frac{8.00\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 2.0\times10^{-2}+\frac{1.43\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 1.0\times10^{-1} = 3.33\text{E-}02$$

Caesium (Cs)

$$\frac{2.20\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 2.0\times10^{-2}+\frac{8.00\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 2.0\times10^{-2}+\frac{1.43\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 1.0\times10^{-1} = 3.33\text{E-}02$$

Tellurium (Te)

$$\frac{2.20\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 2.0\times10^{-5}+\frac{8.00\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 4.0\times10^{-3}+\frac{1.43\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 2.0\times10^{-2} = 6.00\text{E-}03$$

Strontium (Sr)

$$\frac{2.20\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 1.0\times10^{-6}+\frac{8.00\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 1.0\times10^{-6}+\frac{1.43\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 5.0\times10^{-6} = 1.67\text{E-}06$$

Ruthenium (Ru)

$$\frac{2.20\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 4.0\times10^{-6}+\frac{8.00\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 4.0\times10^{-5}+\frac{1.43\times10^{-8}}{2.20\times10^{-8}+8.0\times10^{-8}+1.43\times10^{-8}}\times 2.0\times10^{-4} = 6.07\text{E-}05$$

PWR7:

Iodine (I)

$$\frac{2.25\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times10^{-4}+\frac{2.75\times10^{-9}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 1.0\times10^{-3}+\frac{7.45\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times10^{-4}+\frac{2.89\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 1.0\times10^{-3} = 3.86\text{E-}04$$

Caesium (Cs)

$$\frac{2.25\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 4.0\times10^{-4}+\frac{2.75\times10^{-9}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times10^{-3}+\frac{7.45\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 4.0\times10^{-4}+\frac{2.89\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times10^{-3} = 7.71\text{E-}04$$

Tellurium (Te)

$$\frac{2.25\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times10^{-5}+\frac{2.75\times10^{-9}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 1.0\times 10^{-4}+\frac{7.45\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times10^{-4}+\frac{2.89\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 1.0\times10^{-3}=3.54\text{E-}04$$

Strontium (Sr)

$$\frac{2.25\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 4.0\times10^{-4}+\frac{2.75\times10^{-9}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times 10^{-3}+\frac{7.45\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 4.0\times10^{-4}+\frac{2.89\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times10^{-3}=7.71\text{E-}07$$

Ruthenium (Ru)

$$\frac{2.25\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 4.0\times10^{-6}+\frac{2.75\times10^{-9}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times 10^{-5}+\frac{7.45\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 4.0\times10^{-6}+\frac{2.89\times10^{-7}}{2.25\times10^{-7}+2.75\times10^{-9}+7.45\times10^{-7}+2.89\times10^{-7}}\times 2.0\times10^{-5}=7.71\text{E-}06$$

PWR8:

Iodine(I)

3.0E-02

Caesium (Cs)

2.0E-05

Tellurium (Te)

1.4E-06

Strontium (Sr)

2.5E-07

Ruthenium (Ru)

7.0E-06

PWR9:

Iodine (I)

$$\frac{1.2\times10^{-5}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 7.0\times10^{-7}+\frac{1.69\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.0\times 10^{-5}+\frac{2.36\times10^{-6}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 4.0\times10^{-6}+\frac{1.91\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.0\times10^{-5}=1.30\text{E-}06$$

Caesium (Cs)

$$\frac{1.2\times10^{-5}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 7.0\times10^{-7}+\frac{1.69\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.0\times10^{-5}+\frac{2.36\times10^{-6}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 4.0\times10^{-6}+\frac{1.91\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.0\times10^{-5}=1.30\mathrm{E}\text{-}06$$

Tellurium (Te)

$$\frac{1.2\times10^{-5}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.0\times10^{-9}+\frac{1.69\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 1.0\times10^{-6}+\frac{2.36\times10^{-6}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.8\times10^{-6}+\frac{1.91\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 1.4\times10^{-5}=4.84\mathrm{E}\text{-}07$$

Strontium (Sr)

$$\frac{1.2\times10^{-5}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 4.0\times10^{-9}+\frac{1.69\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.5\times10^{-7}+\frac{2.36\times10^{-6}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 5.0\times10^{-8}+\frac{1.91\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.5\times10^{-7}=1.23\mathrm{E}\text{-}08$$

Ruthenium (Ru)

$$\frac{1.2\times10^{-5}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.0\times10^{-9}+\frac{1.69\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 1.0\times10^{-6}+\frac{2.36\times10^{-6}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 1.4\times10^{-6}+\frac{1.91\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 7.0\times10^{-6}=2.43\mathrm{E}\text{-}07$$

Appendix C: Release Fraction Calculation for Generic Level 2 PSA Model

Release fraction data calibration for new proposed release categories based on the data obtained from the previous 9 TMI release category results. According to the grouping scheme, by using each TMI release category frequency difference, weighted inventory fractions can be obtained for each new release category.

RC1:

Noble Gas (Xe)

1.00E+00

Iodine (I)

1.57E-02

Caesium (Cs)

1.55E-02

Tellurium (Te)

8.70E-03

Strontium (Sr)

6.33E-05

Ruthenium (Ru)

1.08E-03

RC2:

Noble Gas (Xe)

$$\frac{1.17\times10^{-7}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{1.26\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{3.19\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}=$$

5.22E-01

Iodine (I)

$$\frac{1.17\times10^{-7}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{1.26\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{3.19\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}=$$

1.60E-02

Caesium (Cs)

$$\frac{1.17\times10^{-7}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{1.26\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{3.19\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}=$$

1.85E-03

Tellurium (Te)

$$\frac{1.17\times10^{-7}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{1.26\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{3.19\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}=$$

4.35E-04

Strontium (Sr)

$$\frac{1.17\times10^{-7}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{1.26\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{3.19\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}=$$

5.62E-07

Ruthenium (Ru)

$$\frac{1.17\times10^{-7}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{1.26\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}+\frac{3.19\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times10^{-1}=$$

9.71E-06

RC3:

Noble Gas (Xe)

$$\frac{6.95\times10^{-10}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 1+\frac{3.78\times10^{-7}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 8.3\times10^{-1} = \text{8.30E-01}$$

Iodine (I)

$$\frac{6.95\times10^{-10}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 1.3\times10^{-1}+\frac{3.78\times10^{-7}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 1.24\times10^{-2} = \text{1.26E-02}$$

Caesium (Cs)

$$\frac{6.95\times10^{-10}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 1.3\times10^{-1}+\frac{3.78\times10^{-7}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 1.24\times10^{-2} = \text{1.42E-02}$$

40

Tellurium (Te)

$$\frac{6.95\times10^{-10}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 5.95\times10^{-2}+\frac{3.78\times10^{-7}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 1.66\times10^{-2} = \text{1.66E-02}$$

Strontium (Sr)

$$\frac{6.95\times10^{-10}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 3.96\times10^{-3}+\frac{3.78\times10^{-7}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 2.73\times10^{-4} = \text{2.80E-04}$$

Ruthenium (Ru)

$$\frac{6.95\times10^{-10}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 5.77\times10^{-2}+\frac{3.78\times10^{-7}}{6.95\times10^{-10}+3.78\times10^{-7}}\times 5.44\times10^{-3} = \text{5.53E-03}$$

RC4:

Noble Gas (Xe)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 9.2\times10^{-1} = \text{9.93E-01}$$

Iodine (I)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1.09\times10^{-2}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 8.5010^{-1}= \text{8.34E-02}$$

Caesium (Cs)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1.09\times10^{-2}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 8.5010^{-1}= \text{8.34E-02}$$

Tellurium (Te)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1.64\times10^{-3}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1.71\times10^{-1} = 1.63\text{E-02}$$

Strontium (Sr)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 7.58\times10^{-6}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 8.58\times10^{-2}= 7.41\text{E-03}$$

Ruthenium (Ru)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 3.96\times10^{-5}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 6.3\times10^{-1}= 5.44\text{E-02}$$

RC5:

Noble Gas (Xe)	1.00E-03
Iodine (I)	1.30E-06
Caesium (Cs)	1.30E-06
Tellurium (Te)	4.84E-07
Strontium (Sr)	1.23E-08
Ruthenium (Ru)	2.43E-07

Appendix D: Frequency Quantification for Generic Level 2 PSA Model

D.1 Top Events and Component Failure Rate Determination

The entry for containment event tree is the frequency of plant damage state group. As the level 2 PSA mainly focused on severe accidents, therefore, those plant damage states with core melt scenarios are only considered here. In order to get the frequency data for each plant damage state group, the Surry plant level 2 PSA was taken as reference (NUREG-1150).

The below table summarized the frequency of core damage due to PDS group in Surry plant:

PDS group name	Mean frequency (/yr)
Station blackout	2.74E-05
Transient	3.6E-06
LOCA	6.0E-06
ISLOCA	1.6E-06
SGTR	1.8E-06

Based on the table above, the proposed plant damage state group frequency can be obtained by further classification:

For core melt with high pressure in reactor system, it is station blackout+ transient +small break LOCA, the corresponding core melt frequency with high pressure in reactor system can be obtained as below:

2.74E-05+3.6E-06+5.84E-06= 3.684E-05/yr

For core melt with low pressure in reactor system, it is Large break LOCA

1.56E-07/yr

(Here because we only have core damage frequency due to LOCA, therefore, we need to find small LOCA and Large LOCA contribution difference for core damage.

According to the thermal-hydraulic test performed for pressurized water reactor (Cho et al., 2017), Large LOCA contribution to CDF is 0.3/29.4, small LOCA contribution to CDF is 11.4/ 29.4, the total LOCA contribution is 11.7/29.4. so small LOCA takes up 0.974, large LOCA takes up 0.026 for the CDF due to LOCA.

Therefore, It can be assumed in our case the corresponding core damage frequency due to large LOCA is 1.56-E07/yr, the core damage frequency due to small LOCA is 5.84-E06/yr.)

For the reactor pressure vessel (RPV) rupture scenario, there are three main ways that can lead to the vessel rupture, which are pressurized ejection, gravity pour, or gross bottom head failure. According to the expert judgement from NUREG-1150, the pressurized ejection takes 60% of the case that the vessel is at high or intermediate pressure. The probability for pressurized ejection is 0.397 based on NUREG-1150. (pp.278) From the information above, we could also infer the 40 % of vessel breach case is melt-through and the probability would be 0.265.

Also, based on the small rupture and large rupture occurrence frequency comparison for RPVs, it was found the big leak frequency is 10 percent of the small leak frequency. Therefore, when the rupture happened, the probability to have a large hole is 0.1 and the probability to have a small hole is 0.9.
(NUREG-1150, pp.388)

The below is the safety system failure probability:

Containment spray system failure frequency: 0.1
Taken from TMI Unit1 level 2 PSA (TMI Unit 1 level 2 PSA report SPRAYEFF

Containment hydrogen mitigation system failure frequency: 0.01
Taken from TMI Unit 1 level 2 PSA report Hydrogen Spark

Containment radioactive control system (venting system) failure frequency 1.042E-3/yr
(Silvonen, T. (2011). Reliability analysis for passive systems – A case study on a passive containment cooling system.)

D.2 CET End State Probability Calculation

Based on the information provided above, each CET end state frequency can be calculated as below:

For the plant damage state group -core melt with high reactor pressure (unit: /yr):

CET1-1

3.68E-05 (No rupture, just core melt)

CET2-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.1 \times \ a*^{5} = 1.46\text{E-07}$$

CET3-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.1 \times 0.9 \times 0.99 \times 0.9986 = 1.30\text{E-06}$$

CET4-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.1 \times 0.9 \times 0.99 \times 1.4 \times 10^{-3} = 1.82\text{E-09}$$

CET5-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.1 \times 0.9 \times 0.01 \times 0.9986 = 1.31\text{E-08}$$

CET6-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.1 \times 0.9 \times 0.01 \times 1.4 \times 10^{-3} = 1.84\text{E-11}$$

CET7-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.1 \times 0.1 \times 0.99 \times 0.9986 = 1.44\text{E-07}$$

CET8-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.1 \times 0.1 \times 0.99 \times 1.4 \times 10^{-3} = 2.02\text{E-10}$$

CET9-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.1 \times 0.1 \times 0.01 \times 0.9986 = 1.46\text{E-09}$$

CET10-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.1 \times 0.1 \times 0.01 \times 1.4 \times 10^{-3} = 2.05\text{E-12}$$

CET 11-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.9 \times 0.9 \times 0.99 \times 0.9986 = 1.17\text{E-05}$$

CET12-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.9 \times 0.9 \times 0.99 \times 1.4 \times 10^{-3} = 1.64\text{E-08}$$

CET13-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.9 \times 0.9 \times 0.01 \times 0.9986 = 1.18\text{E-07}$$

CET14-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.9 \times 0.9 \times 0.01 \times 1.4 \times 10^{-3} = 1.66\text{E-10}$$

CET15-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.9 \times 0.1 \times 0.99 \times 0.9986 = 1.30\text{E-06}$$

[5] According to TMI Unit 1 level 2 PSA Report, the probability that the reactor vessel becomes a rocket and impinges into the containment vessel is equally as unlikely (USNRC, 2007). Therefore, taking TMI Unit 1 level 2 PSA and French 1300 MWe level 2 PSA both as reference, hereby a is given 0.1 in this research.

CET16-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.9 \times 0.1 \times 0.99 \times 1.4 \times 10^{-3} = 1.82\text{E-09}$$

CET17-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.9 \times 0.1 \times 0.01 \times 0.9986 = 1.31\text{E-08}$$

CET18-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.9 \times 0.1 \times 0.01 \times 1.4 \times 10^{-3} = 1.84\text{E-11}$$

CET19-1

$$3.68 \times 10^{-5} \times 0.397 \times 0.9 \times b*$$ [6]

CET20-1

For the isolation failure occurred before the core melt, data from TMI Unit 1 level 2 PSA was used to estimate the magnitude.

CET21-1

2.30E-07 (French 1300 MWe level 2 PSA)

CET22-1

6.67E-07 (French 1300 MWe level 2 PSA)

According to Surry Plant PSA results, except for LOCA, transient and SBO both contributed to induced bypass accidents. Hereby, the induced SGTR value is taken from the literature for induced bypass accident.

For the plant damage state group- core melt with low reactor pressure (unit: /yr):

CET1-2

1.56E-07 (No rupture, just core melt)

CET2-2

0

(Because the reactor system is at low pressure, no alpha mode can occur)

CET3-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.1 \times 0.9 \times 0.99 \times 0.9986 = 3.68\text{E-09}$$

CET4-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.1 \times 0.9 \times 0.99 \times 1.4 \times 10^{-3} = 5.16\text{E-12}$$

CET5-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.1 \times 0.9 \times 0.01 \times 0.9986 = 3.72\text{E-11}$$

[6] Hereby, the factor a depends on the leakage size, based on the mean value for this kind of event in German Risk Study, it was given 0.01 in this research. (EPRI, (1981). A study of the risk due to accidents in nuclear power plants. Palo Alto,)

CET6-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.1 \times 0.9 \times 0.01 \times 1.4 \times 10^{-3}$$ = 5.21E-14

CET7-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.1 \times 0.1 \times 0.99 \times 0.9986$$ = 4.09E-10

CET8-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.1 \times 0.1 \times 0.99 \times 1.4 \times 10^{-3}$$= 5.73E-13

CET9-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.1 \times 0.1 \times 0.01 \times 0.9986$$ = 4.13E-12

CET10-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.1 \times 0.1 \times 0.01 \times 1.4 \times 10^{-3}$$= 5.79E-15

CET 11-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.9 \times 0.9 \times 0.99 \times 0.9986$$ = 3.31E-08

CET12-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.9 \times 0.9 \times 0.99 \times 1.4 \times 10^{-3}$$ = 4.64E-11

CET13-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.9 \times 0.9 \times 0.01 \times 0.9986$$ = 3.34E-10

CET14-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.9 \times 0.9 \times 0.01 \times 1.4 \times 10^{-3}$$ = 4.69E-13

CET15-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.9 \times 0.1 \times 0.99 \times 0.9986$$ = 3.68E-09

CET16-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.9 \times 0.1 \times 0.99 \times 1.4 \times 10^{-3}$$ = 5.16E-12

CET17-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.9 \times 0.1 \times 0.01 \times 0.9986$$ = 3.72E-11

CET18-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.9 \times 0.1 \times 0.01 \times 1.4 \times 10^{-3}$$ = 1.95E-12

CET19-2

$$1.56 \times 10^{-7} \times 0.265 \times 0.9 \times b*$$[7]

CET20-2

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[7] Hereby, the factor b depends on the leakage size, based on the mean value for this kind of event in German Risk Study, it was given 1 in this research. (EPRI, (1981). A study of the risk due to accidents in nuclear power plants. Palo Alto,)

For the isolation failure occurred before the core melt, data from TMI Unit 1 level 2 PSA was used to estimate the magnitude.

CET21-2

0

CET22-2

0

(According to Surry plant PSA results, LOCA has no contribution to bypass accident, thereby, for the frequency of bypass accident mainly induced by Large break LOCA, it was set as 0.)

Besides the core damage induced bypass accidents, initiating bypass accidents which lead to core damage, also need to be taken into consideration. Based on Surry plant result, the probability for core damage frequency due to ISLOCA and SGTR is 1.6E-06 and 1.8E-06

After the above detailed calculation, the table below shows the release frequency for each CET end states.

Containment events	Frequency(/yr)
CET1	3.70E-05
CET2	1.46E-07
CET3	1.30E-06
CET4	1.82E-09
CET5	1.32E-08
CET6	1.85E-11
CET7	1.45E-07
CET8	2.03E-10
CET9	1.46E-09
CET10	2.05E-12
CET11	1.17E-05
CET12	1.64E-08
CET13	1.19E-07
CET14	1.66E-10
CET15	1.30E-06
CET16	1.83E-09
CET17	1.32E-08
CET18	1.85E-11
CET19	1.68E-07
CET20[8]	3.78E-07
CET21	1.83E-06
CET22	2.47E-06

D.3 Release Category Frequency Calculation

[8] In TMI Unit 1 level 2 PSA, the frequency for isolation failure occurred before the core melt was calculated as below 3.78E-07 +6.59E-10 = 3.78E-07.

RC1 (Early Containment Failure)	1.61E-06/yr
RC2 (Late Containment Failure)	1.17E-05/yr
RC3(Containment Isolation Failure)	5.46E-07/yr
RC4 (Containment Bypass Failure)	4.30E-06/yr
RC5 (Containment Intact)	5.00E-05/yr