Generic Level 2 Probabilistic Safety Analysis on Pressurized Water Reactors

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Executive Summary

Research Background

The first nuclear power station was put into operation in 1954. Since then, nuclear power plants have been widely applied for electricity generation all over the world. The electricity generation 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. Because there are no carbon dioxide emissions during energy production, nuclear energy is perceived as one of the most environmentally friendly energy forms in the world. Moreover, because nuclear fuel source, uranium, is largely stored in the earth, it also helps to guarantee the energy supply security for the long-term run. However, although nuclear energy could bring a lot of benefits to society, there are still a lot of safety concerns regarding its development.

In general, the risks associated with nuclear power plant operation can be classified into technical risks and non-technical risks. Technical risks are normally induced by the malfunction of the component or human errors, which can be quantified by using the occurrence probability of the undesired events and their consequence magnitude. Different from technical risks, non-technical risks can not be easily quantified because they refer to the societal and ethical issues that brought by undesired events. But these two aspects of the nuclear risk both should be well considered during the accident mitigation management so that the nuclear risk can be well mitigated. In order to analyze the technical risks associated with nuclear power plant operation, probabilistic safety analysis (PSA) has been developed and it is required by most power plants for their safety assessment. Among three different levels of PSA, the level 2 PSA is one of the key components in the risk analysis. It is designed to investigate the containment response and identify the potential release paths for severe accidents. However, currently the level 2 PSA studies are very detailed and plant-specific, which makes it very hard to use the analysis result for the precursor analysis. Therefore, a generic level 2 PSA is needed so that a general overview of potential accidents and their consequences for a certain design type of reactors can be provided. In addition, the analysis result can be further used as a reference for the optimization of the risk mitigation management. But one of big limitations of the PSA is that PSA overlooks the societal and ethical aspect of the risk. Therefore, the societal and ethical discussion of nuclear risk management also needs to be performed so that the nuclear risk can be well addressed. Social acceptance and ethical acceptability are two important factors to analyze if the development plan or management activities are socially and morally acceptable by the public. Therefore, if nuclear risk management can be evaluated from these two aspects, the societal and ethical issues associated with nuclear risk can be well managed.

Methodology

Literature review and quantitative analysis are two major methodologies that have been used in this research. In order to synthesize the related studies and identify the research gap, a broad literature review has been performed. Databases such as Science Direct, Scopus and Google Scholar have been used for literature study. Academic publications ranging from organization reports to journal papers have been well selected for detailed study in order to obtain a state-of-art knowledge of level 2 PSA development and societal and ethical discussion regarding nuclear development. After the literature review study, the quantitative risk analysis is applied in order to develop the generic level 2 PSA.

During the development process of the generic model, the general set-up steps of level 2 PSA have been followed, which are recommended by the International Atomic Energy Agency (IAEA). It includes plant familiarization, plant damage state identification, accident progression analysis, release category identification, and source term analysis. However, different from other level 2 PSA studies, in this research, a generic containment event tree has been built so that various release scenarios can be identified. The data for release consequence calculation are taken from the existing studies. In order to better test the validity of the model results, different level 2 PSA studies such as WASH-1400, French 1300Mwe level 2 PSA, and TMI Unit 1 level 2 PSA have been selected for further comparison.

After the generic model development and result discussion, the major accident mitigation plan can be developed based on the analysis results. By discussing the risk perception, public emotion, and trust issues, the societal problems regarding the risk mitigation management are properly addressed. By analyzing the distributional justice, procedure justice, and recognition justice issues, the ethical acceptability of the mitigation plan has been evaluated. Therefore, through the literature review and quantitative study, both technical and non-technical nuclear risk can be analyzed, which will greatly contribute to the optimization of risk mitigation management.

Final Deliverables

All in all, three main deliverables are provided by this research, which are summarized as follows:

A conceptual review of 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 functionalities. The classified containment safety systems are the temperature and pressure control system, the hydrogen mitigation system as well as the radioactive release control system. Based on the above classification, the implementation situation of these systems has also been discussed through a wide comparison among different pressurized water reactors around the world.

A generic level 2 PSA for pressurized water reactors

A generic level 2 PSA has been performed in the research. By following pressurized water reactor operation and safety system analysis, the plant damage state logic diagram was developed, which can be used to select the severe accidents for containment response analysis. In the accident progression analysis, a generic containment event tree has been built based on those severe accident states that are 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 can be studied. In total, 22 containment end-state events are spotted, which have been further grouped into 5 release categories for release consequence analysis. Furthermore, the estimated release frequencies have been cross-compared with other level 2 PSA studies so that the approximation of the analysis can be evaluated.

A broad societal and ethical discussion regarding nuclear risk management

In the research, societal and ethical issues regarding nuclear risk management have been properly addressed. By analyzing the social acceptance issues and justice issues, it helps to present a full picture of the risk information of the associated nuclear risk. Different from studies that solely consider the technical side of the risk, the societal and ethical discussion makes the work more comprehensive. In addition, a nuclear risk governance framework has been proposed which can integrate both technical and non-technical risk aspect into the decision-making process so that the risk mitigation management can be continuously optimized.

Study Relevance with the MSc. Program-Engineering & Policy Analysis

According to the program introduction, the EPA master is designed to challenge the students' engineering skills and engineering DNA to analyze and solve the grand challenges. Therefore, one of the main focus of the EPA program is to help students gain a deep insight into the grand challenges and find optimal solutions to deal with these challenges. Among several major grand challenges, energy safety issues are always perceived as the top priority during normal operation because they have a great impact on people's everyday life. Therefore, during my master project, I selected nuclear safety issues as my research topic aiming at providing a better solution to deal with the nuclear risk. In the project, a generic level 2 PSA model has been developed and the societal and ethical issues of nuclear risk management have been well discussed. In this way, both technical and non-technical risks of nuclear energy development can be analyzed so that the risk mitigation management can be optimized. Therefore, this project is closely related to the master program and the work results can be used as the reference to deal with nuclear safety issues in the future.

Nomenclature

ACP
Activated Corrosion Product

APET
Accident Progression Event Tree

CDF
Core Damage Frequency

CET
Containment Event Tree

DCH
Direct Containment Heating

DET
Decomposition Event Tree

ECCS
Emergency Core Cooling System

ETA
Event Tree Analysis

FCVS
Filtered Containment Venting System

IAEA
International Atomic Energy Agency (IAEA)

ISLOCA
Interfacing Loss of Coolant Accident

LBLOCA
Large Break Loss of Coolant Accident

LOCA
Loss of Coolant Accident

NEA
Nuclear Energy Agency (OECD)

NIMBY
Not In My Back Yard

NPP
Nuclear Power Plant

MCCI
Molten Core and Concrete Interaction

PAR
Passive Autocatalytic Recombiner

PCCS
Passive Containment Cooling System

PDS
Plant Damage State

PSA
Probabilistic Safety Analysis

PWR
Pressurized Water Reactor

RCS
Reactor Coolant System

RPV
Reactor Pressure Vessel

SBO
Station Black Out

SGTR
Steam Generator Tube Rupture

TEPCO
Tokyo Electric Power Company

TMI
Three Mile Island

(US) NRC
Nuclear Regulatory Commission of the United States

WHO
World Health Organization

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Chapter 1 Introduction

In order to better abate 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 sources, nuclear energy has been applied for electricity production all over the world because it can not only help to reduce carbon dioxide emission but also improve energy supply security. At the end of 2018, there are 451 units of nuclear power plants that have been connected to the grid and the total amount of nuclear electricity generation is 2563 TWh (IAEA, 2019). In addition, by using nuclear energy instead of coal to generate this equivalent amount of electricity, it shows nuclear energy produce 1 million ton less carbon dioxide emissions than the coal. For the OECD countries, according to the latest data published by the Nuclear Energy Agency (NEA) in 2018, the nuclear electricity generation takes up 17.9 % of their total electricity generation (NEA, 2018). It is also estimated by the IAEA that nuclear electricity production will continuously increase in East Europe and Asia Pacific regions; this could have both high and low projections (IAEA, 2017). In addition, it is found more and more countries are interested in nuclear energy development in recent years. According to Taebi & Mayer (2017), there might be another 18 countries that will become newcomers to explore nuclear energy within the next 10 years. Moreover, at present, 55 reactors are under the construction and 110 reactors have been planned for future development (IAEA, 2019). Therefore, it can be seen nuclear energy not only plays an important role in current energy development but also will have a big influence on future electricity generation.

The general electricity production process for most nuclear power plants is through a well-controlled heavy element (i.e. uranium or plutonium) fission reaction. Nuclear fission reaction is the process that heavy element atom nuclei absorbed neutrons and releases the kinetic energy and neutrons. By capturing the heat resulting from the fission energy of the nuclei splitting, 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 methods in the future. However, one of the biggest downsides of this technology is nuclear safety issues. Nuclear safety issues are always perceived as a top priority during the operation. The fission products in nuclear power plants are very unstable, which include Iodine, Caesium, Chromium, Krypton, etc. Once these radioactive substances get released, the damage to the environment and human beings is unbearable and its impacts always last for decades. In general, severe accidents will lead to a large amount of radioactive release to the environment. 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, the Chernobyl accident in 1986 and the 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 to escape 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). Fukushima Daichi accident occurred on 11 March of 2011 and the direct cause of the accident is the tsunami that resulted in a long term SBO (Station Black Out) and the loss of an ultimate heat sink (Yang, 2014). After the accident, around 164,000 people are evacuated due to health concerns (Do, 2019). However, different from the TMI and Fukushima Daiichi accident, Chernobyl is a well-known catastrophe in which a large amount of radioactive substances are released to the environment. Around 4760 Km_2 nearby areas have been evacuated and detrimental effects are recorded in wildlife (Beresford, Scott, & Copplestone, 2019). Therefore, when it comes to the commercial application of electricity generation through nuclear plants, the associated risk should be well considered.

Currently, there is no agreed notion of the risk, but the traditional notion of the risk is interpreted as the probability and the expected value of the event. According to Kaplan & Garrick (1981), the risk is defined as a triplet, which includes the occurrence possibility of the event, a possible impact of the event, and the vulnerability and mitigation solutions. Over the years, various notions regarding the risk have been developed, which broaden the risk concept to a wider view. For example, the knowledge and perception regarding the event also influence the risk estimation.

In general, when it speaks to the dimensions of the risk, the risk can be classified into two categories: technical risk and non-technical risk. From the technical aspect, the risk is described as the occurrence probability of an undesired event and its potential consequence magnitude. It is normally formulated through a mathematic approach. Within the technical notion, the occurrence probability is used to express the uncertainty of the event and the consequence magnitude is expressed through the real loss or damage to the property, human life or the environment. For the non-technical perspective, the risk is more associated with societal impact and psychological influence. However, no matter from which aspect to analyze the associated risk, they both can contribute to the decision-making process in a great way. For example, the results obtained through the quantitative calculation can provide direct insight into the probabilities of the risk and the estimated loss will be due to the undesirable event. The discussion from a societal and philosophical aspect would then focus on the acceptability of such risks and by that, they could make the decision-making process more comprehensive because it includes those aspects that are difficult to be estimated through a quantitative approach.

The risks associated with nuclear energy development are one of the good examples to illustrate how multi-dimensional risks influence the decision-making process. In general, the primary risks for nuclear development can be classified into two bins: the risk associated with normal operation and the risk associated with potential accidents. For risks regarding operation errors and system failures, probabilistic safety analysis (PSA) has been widely applied and continuously reviewed among different generation nuclear plants, aiming to provide a general overview of what kind of accidents may occur in nuclear plants. Also, the associated risk magnitude of these events can be estimated from the technical point of view.

1.1 Level 2 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, level 1 PSA investigates a sequence of events that may lead to core damage. It offers 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 to develop the corresponding accident mitigation measures. Therefore, a detailed study and novel development of level 2 PSA are needed in order to further optimize the nuclear plant mitigation management.

The level 2 PSA is designed to analyze different accident scenarios and potential radioactive release pathways within the containment. In order to analyze accident consequences and quantify release frequencies, several steps are suggested by the IAEA for level 2 PSA development, which include familiarization with plants, plant damage states definition, severe accident analysis, containment performance analysis, source term analysis, and quantification (IAEA, 2010). Following a core damage accident, level 2 PSA is applied to model the accident progress within the containment, especially those which may cause potential containment failure and get the fission products released to the environment. In the accident progress analysis, the design features of containment safety systems need to be well considered because they can determine the accident release frequency and the release magnitude. The widely applied method to analyze accident progress and containment response is to develop containment event trees (CET). Therefore, the CET development approach is very important for the whole level 2 PSA development. Since the level 2 PSA was first introduced for plant risk assessment, various CET approaches have been developed in order to better identify potential accidents.

In 1975, the level 2 PSA was first introduced by WASH 1400, in the study of plant risk assessment. Since then, the level 2 PSA methodology and plant-specific application analysis have been developed very fast. In 1981, a level 2 PSA has been performed to assess the severe accident scenarios of Zion and Indian Point nuclear power plants by using code MARCH. Compared with WASH-1400, it was greatly expanded and organized. Therefore, in the early 80s, most of the Level 2 PSA performed by the USA and Europe were mainly based on this 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. In the 1990s, NUREG-1150 has been proposed and it was perceived as the most comprehensive report for level 2 PSA study during that time. In NUREG-1150, the accident progression event tree (APET) approach was introduced to analyze the accident progress for some American nuclear plants, which include PWR Surry and Zion (NEA, 2007). However, these event trees all contain more than one hundred events and each event has multiple branches, which made it impossible to graphically represent. Moreover, because these event trees have generated numerous end event states, these diagrams are very difficult to understand if no computer-based reduction technique is applied. Although these large event trees can provide a relatively complete estimation of accident scenarios, it is still very difficult to figure out the occurrence logic for a specific event due to the too many details.

Later on, by taking NUREG-1150 as the reference, NUERG-5602 was proposed. In this study, the simplified containment event trees (SCETs) have been developed in order to analyze the containment behaviors of the Sequoyah nuclear plant. Even though it succeeded in simplifying the event trees and was able to reproduce the results of the APET approach, it still has a lot of end states which are difficult to understand. In order to tackle these issues, a small event tree approach has been gradually developed. Different from the large event tree method, the small event tree is 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 1997, NEA (1997) evaluated the development of level 2 PSA among 10 existing pressurized water reactors (PWRs) around the world. Among these studies, only 2 of them have applied large event tree methods and the rest of them all used small event trees for accident progression analysis.

Besides the CET approach development, other developments of level 2 PSA also have been made in recent years, especially for the studies related to the physical and chemical behaviors of nuclear plants. Through the detailed investigation of these behaviors, the analysis results can be further used to develop mitigation plans regarding certain accident scenarios (NEA, 2004). Kujal has performed containment behavior analysis for VVER-1000 under 5 different accident scenarios (NEA, 2007). IRSN also studied the hydraulic and thermal conditions of French 1300 MW PWR fleets by using ASTEC V1.3 code. In the research, 60 events at full power states and 20 events at shutdown states have been selected to develop event trees (Tregoures et al., 2010). Later on, Zvoncek, et al. (2017) has investigated different hazards which are associated with the KKL nuclear plant so that the mechanism of accident progress can be better understood. In addition, Hwang, et al. (2018) applied the DET approach for APR1400 level 2 PSA study so that the sequence events can be identified for accident management. Gunhyo & Moosung (2018) also developed multiple event trees in order to study the accident phenomena in the Westinghouse 3-loop system.

Although probabilistic safety analysis can inform decision-makers about the potential accident consequences for nuclear plants, it still cannot properly address the associated societal issues during the operation and accident mitigation management. For example, regarding nuclear accident mitigation management, the societal and ethical aspects of the nuclear risk are important factors that need to be considered because public distrust and opposition can be the main obstacles for the implementation process. If the authorities overlook the societal and ethical aspects of the risk and fail to properly communicate the risk with the public, it will lead to big social anxiety towards nuclear operation safety. This may largely impact the future development of nuclear energy. For instance, due to the poor risk management, the accident that happened to the Fukushima nuclear power plant has again brought a lot of social resistance towards nuclear technology (Wheatley, Sovacool, & Sornette, 2016). After the accident, over 7 years have passed but still around 60,000 evacuated residents live in temporary houses and they are not allowed to go back to their original hometown (Suzuki, 2019). Considering these affected communities are suffering the accident consequence, the justice issues will also need to be brought up because the purpose of nuclear energy development is to provide sustainable energy to all individuals. If the burdens and benefits that are brought by nuclear energy development can not be equally distributed to all individuals, it may result in a lot of societal problems and this may greatly damage the economic development and societal progress. Therefore, justice such as distribution justice, procedure justice, and recognition justice should also be well discussed so that the societal and ethical aspects of the nuclear risk can be properly addressed.

1.2 Social Acceptance and Ethical Acceptability in Nuclear Power Development

1.2.1 Social Acceptance

Social acceptance is perceived as a major concern for energy technology development, especially in nuclear power development (Yuan et al., 2017). If the government and the company stakeholders want to implement its safety management plan smoothly, a good public acceptance is needed because the government actions should be taken based on a good consensus (Roh, 2017). Therefore, there is no wonder that countries that are operating nuclear power plants endeavored to raise the public acceptance of nuclear technology. A better social acceptance towards the nuclear power plant will reduce a lot of resistance towards its continuous operation and accident mitigation management.

As suggested by Assefa and Frostell (2007), social acceptance is one of the key factors influencing the social sustainability of energy technology. Therefore, the social acceptance of nuclear technology has been widely studied in both academia and industry. Several indicators have been proposed aiming to better evaluate the public acceptance towards nuclear power development, which include knowledge of nuclear technology, public emotions, economic benefit, and risk perception. Among these indicators, the familiarity of nuclear technology and the risk perception towards nuclear accidents influence a lot regarding the acceptance level of the public. For example, in the UK, even though people are aware of the accident consequence, they still prefer nuclear power generation because it could abate climate change and contribute to the energy security (Teravainen et al., 2011). While in Germany, due to the public fear of the accident consequence and opposition regarding the development plan, nuclear power development was ceased. Moreover, when it speaks about the risk perception from the local community, it was found the acceptance level from the local community also plays an important role in the management plan development (Wüstenhagen et al., 2007). Take the China Haiyang nuclear power plant development as an example, according to the survey performed by Yuan et al. (2017), many respondents showed their worries about local environmental issues because potential accidents may cause great damage to the surroundings. At the national level, 77% of respondents support nuclear power development, but when speaking of nuclear power development in their local regions, the figure drops sharply. This big difference of social acceptance regarding nuclear power development between the national level and local level might reflect the "Not in My Backyard" (NIMBY) attitude of the local community (Van der Horst, 2007). However, if decision-makers simply use NIMBY to explain why the local acceptance level drops, it may be very biased because NIYMB overlooks the ethical objections that many people hold towards the nuclear power development (Ramana, 2011). As mentioned by Wolsink (2006), NIMBY has been frequently used to impede and obscure the genuine analysis for the public response. Some scholars even suggested to discard the NIMBY for the public response analysis (Freudenburg and Pastor 1992; Burningham 2000; Snary 2004). Therefore, it is unfair to just use NIMBY to explain the result difference between the national level and the local level. The main cause for this phenomenon is that local communities hold different perceptions regarding the nuclear risk and they also do not trust the ability of the authority for risk mitigation. According to Whitfield (2009), public opposition is correlated to public perceptions of the risk and the trust of the authorities. Thus, if the government aims to reduce the opposition level, a good and transparent communication mechanism must be guaranteed so that the affected parties can participate in the risk mitigation development. In this way, it may further help to increase the public trust level towards risk mitigation measures.

In addition to the knowledge and risk perception, speaking of public emotions, Taebi, Roser and van de Poel (2012) pointed out the public emotions towards nuclear power development are not always well treated by decision-makers because the public was assumed too emotional to engage in the rational debate of management plan development. According to Taebi et al. (2012), there are two typical ways used by decision-makers to respond to public emotions, which are either neglecting these emotions or using them as the reason to stop potential plan development. Therefore, if the government or major stakeholders want to further improve the level of social acceptance, public emotions should be well taken care of. Especially for accident mitigation management, the effectiveness of the emergency plan highly depends on public participation. When it speaks to the long-term effect, there is another big concern regarding nuclear power development because the economic benefit distribution among generations is not justified. It seems the current generation can gain economic benefits at the cost of the interest of future generations. Especially for nuclear waste management and contaminated area treatment, the economic benefit and cost need to be well considered.

From the above discussion, it can be concluded that social acceptance has a great impact on the development of nuclear management plans. However, beyond these social acceptance issues, there are still a lot of moral issues that may also impact nuclear plant management. These ethical issues are normally excluded from social acceptance studies. For example, regarding the public acceptance of nuclear plant development and its accident mitigation plan, in the China Haiyang nuclear power plant case, the nationwide acceptance level is much higher than the one in local communities. It will lead to the question about which acceptance level should be considered. If the government takes the national level consensus, the distributional justice issues will be brought up because the nuclear power development may burden the local environment and the local communities have to bear the risk of potential accidents, but the electricity generated by the plant will benefit a large amount of nations' population. If the government take the local acceptance as the primary concern, the national interest might be undervalued and different regions may hold different interest regarding nuclear plant management. Additionally, even if the local acceptance level can be increased through government-sponsored means of appeasement for the local regions, there are still arguments regarding whether receiving these subsidies is morally right or not. It is legitimate to receive a certain amount of compensation because of the extra burden and risk exposure caused by potential nuclear accidents. But some scholars worry that if there are no well-established ethical guidelines, these compensation methods could become the way to "bribe" the local regions so that the social acceptance level within local communities can be largely increased (Hannis & Rawles, 2013). In addition to these ethical issues regarding local social acceptance and national acceptance towards nuclear plant management, there are some other justice issues that also need to be well considered during the project development.

1.2.2 Ethical Acceptability

As mentioned in previous sections, there are a lot of ethical issues that are associated with nuclear plant management, but social acceptance can not properly stress these ethical issues. Therefore, it is important to bring up the concept of ethical acceptability in order to differentiate from social acceptance. According to Taebi (2016), social acceptance refers more to the notion that the technology or proposed management plan is merely accepted by the public. But the ethical acceptability strengthens more on whether the introduction of the technology is morally right and acceptable by the public.

Among ethical discussions about energy development, justice issues were perceived as one of the most important factors that need to be well considered. According to Campbell (2010), justice is to ensure and recognize the equal worth of all individuals with the commitment to distribute good and bad things. By introducing justice principles to energy development, it targeted to offer safe, affordable and sustainable energy to all individuals across various fields (McCauley et al. 2013). There are a lot of studies that have been performed by scholars regarding energy justice, especially in nuclear power development. According to studies conducted by McCauley et al. (2013) and Jenkins et al. (2016), energy justice can be analyzed from these three core aspects: distributional justice, procedural justice, and recognition justice. These three tenets interlink with each other and function as basic pillars in order to gain a better insight into energy justice.

1.2.2.1 Distributional Justice

Distributional justice is the first pillar of energy justice, which consists of spatial and temporal dimensions. It is associated with the benefits and costs distribution during the energy supply and consumption. For the spatial dimension of distributional justice, it is about the uneven allocation of the environmental benefits and costs brought by nuclear management plan development. It is also about the unequal responsibility share among different regions. The spatial issues are closely related to energy justice. Because of the geographical location difference, certain regions will get more benefits or bear more risk exposures than others from nuclear power development. Therefore, the equality of cost and benefit distribution among regions should be well guaranteed during the decision-making. For example, the nuclear power plant development could provide a stable electricity supply nationwide with less CO2 emissions compared with fossil fuels. But it will pose the local communities and environment under the risk of potential nuclear accidents. In addition, if major accidents happen in the future, the local communities may have to resettle to another place, the cost for this relocation should be well communicated with all affected regions.

For temporal dimensions, nuclear power development can bring some benefits for the current generation but it may burden future generations due to its contaminated area treatment and waste repository issues. This again will bring about the intergenerational justice into the discussion. Researchers including Kermisch and Taebi (2017) have identified that the aim of intergenerational justice is to keep the opportunities equally open for future generations regarding key interests and well beings. Taking the contaminated area treatment as an example, system failure or maloperation in nuclear power plants may cause a large amount of radioactive release. The impact of these accidents can last for several generations. Therefore, the treatment plan on contaminated areas becomes very critical. It seems only the current generation benefits from the plant operation, but the accident consequence burdens several generations. Additionally, future generations may have their own choice regarding whether or not to accept the treatment plan made by the current generation. Therefore, during the accident mitigation management, this aspect should be well considered and the current generation should continuously make an investment in the research and technology which can help reduce the radioactive release impact in case of major accidents. Beyond these technology improvements, global governance of nuclear risk management may be also helpful because different countries have different capabilities to deal with nuclear risks (Taebi & Mayer, 2017).

In addition, this also explains the reason why probabilistic safety analysis is important for decision-making because it can help to identify the risks of the plants in order to ensure that the nuclear plant operation is as safe as possible. Therefore, if continuous improvement can be made through the safety technology development and plant occupational training, the plant accident induced justice issues may get further reduced.

1.2.2.2 Procedure Justice

Procedure justice is the second tenet of energy justice. It aims to establish an equitable procedure that could get all affected parties engaged during the decision-making process. This procedure should not have any discriminations towards different stakeholders and all of their opinions should be taken into consideration. Another important feature of the procedure justice is to require the information of the project fully disclosed by the government. Take the Louisiana uranium enrichment facility case as an example, according to Wigly and Shrade-Frechtte (1996), the statement made by NRC has major justice flaws because they selected facility host community based on their own criteria without any reasonable justification. Also, the local community was not fully informed about the project risk. From this case, it can be seen that the procedure justice was neglected. Because the local communities did not have the chance to participate in the decision-making process, it would lead to further social resistance towards the mitigation plan development in case of any potential accidents. It would be difficult to effectively organize the local communities to participate in emergency preparation because they are even not aware of the accident consequence. Another example is the Fukushima Daiichi accident, the nuclear plant operators were not fully informed about emergency plans in case of accidents. The government agencies also failed to communicate effectively after the accident. These together made the accident progress into the worst-case scenario (Kyne & Bolin, 2016). Moreover, transparency is another issue that will affect procedure justice in nuclear accident development. If there is a lack of transparency during the nuclear development discussion among stakeholders, it may also cause big social resistance. For example, the public should be well informed of subsidies offered for affected areas. Besides the information availability and transparency issue, public emotions are also very important for procedure justice. These emotions should be seriously considered because either positive attitudes or negative attitudes towards the accident mitigation plans will impact its implementation effectiveness later on.

1.2.2.3 Recognition Justice

Recognition justice is the third pillar of energy justice. The concept argues that all individuals should be represented in a fair way without any physical threats. They should be given complete and equal rights regarding nuclear development. From the recognition justice perspective, nuclear energy development should recognize and respect different standpoints embedded in social, ethnic, racial and cultural differences. Sometimes the decision-makers tend to overlook these aspects during the project implementation. For example, in the Louisiana Uranium enrichment facility case, it was found that the facility site was located in an economic disadvantage community without justified arguments to support their selection criteria (Wigly and Shrade-Frechtte, 1996). It can be seen in this example that, there are justice issues about treating affected parties unfairly due to their social or economic status difference. Another example is the relocation issues after the Fukushima accident. At the time of the accident, there are about 76 000 people who live within an area of a 20 km radius from the plant (Hayano, et. al., 2013). After the accident, over 20% of them were forced to relocate over six times due to the lack of emergency plans (Hasegawa, et. al., 2016). Many of them suffered from mental and physical health problems because they cannot adapt to the shelter life and worry much about the separation from families and community. From this example, it can be seen if recognition justice is not considered during the accident mitigation management, the accident induced social problems are far beyond the control. Therefore, ethical justice should be properly addressed and considered during the plant accident mitigation so that the accident induced social issues can be better tackled.

1.3 Research Motivation

From the above discussions, it can be found both technical risk and societal risk are very important for nuclear risk mitigation management. In order to continue optimizing the risk mitigation approach, the associated technical risks and societal issues should be properly addressed and considered. For technical risks, level 2 PSA is one of the good risk analysis approaches to identify potential accident scenarios and the analysis results can be used to develop the corresponding mitigation plans. However, most of the current level 2 PSA studies are very detailed, which makes it difficult for non-experts to evaluate the containment response behaviors under hundreds of accident scenarios. If major accident scenarios cannot be easily identified, the analysis results will be less valuable for the development of risk mitigation strategies. In addition, many level 2 PSA studies have been performed based on specific features of the plant. Although a lot of analysis data can be obtained, it is very difficult to use these data as the reference for similar type nuclear plant risk analysis. Therefore, if a generic level 2 PSA model can be developed, it will benefit a large variety of nuclear plant risk assessments. In addition, the analysis results can be used as a reference for developing risk mitigation plans. However, there are still a lot of uncertainties and limitations embedded in the PSA, which make the decision-makers tend to overlook the societal and ethical issues of the risk. For example, some of the key stakeholders are not involved and participated in the risk assessment during PSA development. The nuclear risk is directly imposed on individuals and affected communities. But during the PSA development, the accident consequences are only evaluated based on the experts' judgment. The risk perceptions towards the accident impact and cultural context of affected parties are normally overlooked during the PSA evaluation. In addition, PSA is designed to quantify the accident consequences by using numbers to represent the accident impact. But a lot of societal and ethical issues can not be measured purely by numbers, such as culture loss and environmental deterioration. If decision-makers only take the PSA results as the risk information for policymaking, it will make the mitigation strategy fail to deal with the social and ethical problems, which may further worsen the situation. Furthermore, the data uncertainty issues of the PSA have been widely discussed regarding its credibility to represent the risk. For example, Taebi (2012) has pointed out there is a serious discrepancy between historical data and estimated accident frequencies. It echoes the fact that PSA can not be solely relied on for decision-making regarding accident mitigation management because there are still several uncertainties that need to be well considered. For example, it is very difficult to predict and quantify human errors in PSA and these human behaviors are normally related to social and ethical problems. Therefore, the PSA can only be used as the reference for policy-making by providing the basis for prioritization of problems, the embedded societal and ethical issues still need to be properly discussed so that the accident mitigation strategy can be continuously optimized. Therefore, it is imperative to include those societal and ethical discussions into the mitigation plan development. Based on the PSA results and the societal and ethical issue discussion, the accident mitigation strategy will be much more comprehensive and effective for risk mitigation.

1.4 Research Question

Based on the analysis and discussion in the above sections, the research question can be proposed as follows:

How can a generic level 2 PSA model optimize mitigation management for a NPP while properly taking into account associated societal and ethical issues?

In order to answer the research question in a systematic way, the main research question can be further subdivided into these following sub-questions:

1. *What steps should be followed in order to create a generic level 2 PSA model?*
2. *How can the generic level 2 PSA model contribute to optimizing risk mitigation within a NPP?*

3. *How should societal issues be addressed in order to complement the probabilistic safety analysis?*

These listed sub-questions help to provide a basic research outline regarding how to create a generic level 2 PSA model and how to use the generic model to improve the nuclear risk mitigation measures with full consideration of societal challenges.

The first sub-question outlines the steps that researchers should follow in order to build a generic level 2 PSA model. It also functions as the guideline for researchers to think about what kind of approaches should be used in different steps to develop the generic model. In general, there are 5 steps to follow during the development of a level 2 PSA model. A broad literature review can be used for reference reactor selection and plant familiarization. Event tree based quantitative risk analysis can be used to investigate the core melt accident progression and estimate the release consequence.

The second sub-question focuses on how to mitigate the associated risk in a nuclear power plant by using the generic level 2 PSA model. Level 2 PSA model is used to analyze the containment response under severe accidents. Therefore, the results obtained from the generic model can be used as the reference for developing the mitigation management plan in case of severe accidents. A qualitative study will be performed based on the release consequence obtained from the generic model of level 2 PSA. It can be further used to analyze what can be optimized in the severe accident mitigation plan from both technical and human intervention perspectives.

The last sub-question helps to understand how societal issues can be properly understood, addressed and included in the probabilistic safety analysis. Therefore, besides the technical risk, the embedded societal issues in the whole life cycle of nuclear power generation can also be considered. A qualitative analysis approach is applied to analyze nuclear societal issues from both social and ethical perspectives.

1.5 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 first two chapters are designed to define the research scope through a broad literature review and qualitative discussion. In Chapter 3, the primary methodology to build a generic level 2 PSA model has been presented. Model-based accident mitigation plans with the full consideration of societal risk have been discussed. In the conclusion chapter, the creative points of the research are summarized, as well as its limitations, and future works.

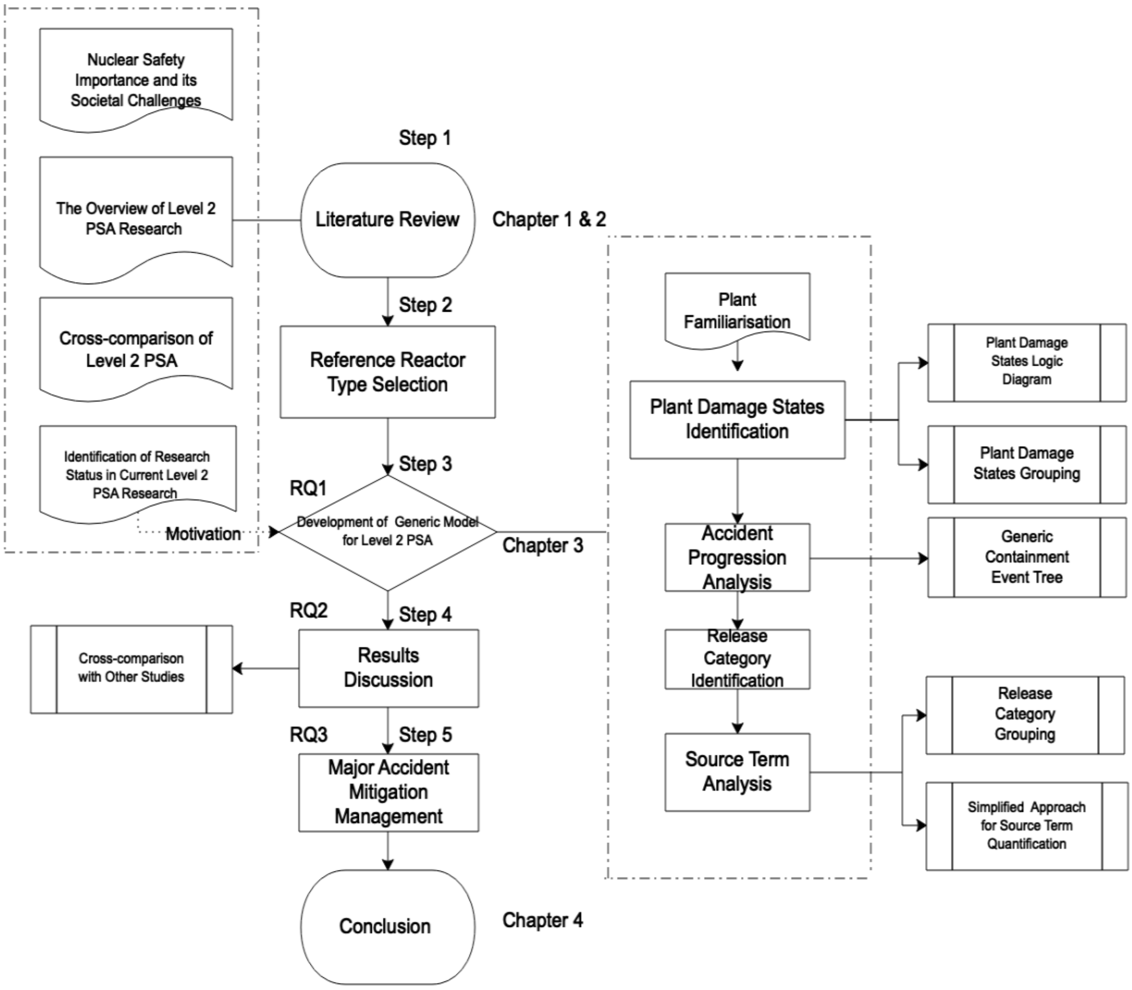

Figure 1. Level 2 PSA generic model development flow

Chapter 2 Research Methodology

In order to better investigate the research question and conduct the research, this chapter is designed to explore what kind of research methods should be used for research development. The corresponding research steps are also pointed out so that a clear overview of research methodologies that are applied in this work can be presented. Based on the discussion in the previous chapter, literature review and quantitative analysis are two major methodologies that have been used to investigate how generic level 2 PSA can contribute to optimizing mitigation management by properly taking societal and ethical issues into account.

2.1 Literature Review

If science is a cumulative endeavor, a literature review is one of the important tools to review and identify shinning points from this long-term endeavor. According to Paré & Kitsiou (2015), literature review is normally used to identify what kind of work has been done on a specific topic and determine the extent to which the corresponding research area can reveal any interpretable trends. It is also used to summarize empirical findings for a specific research question in order to provide the reference for evidence-based practice. Last but not least, based on the results obtained from the literature review, new research development areas can be spotted and corresponding novel frameworks and theories can be generated (Paré & Kitsiou, 2017).

In general, literature reviews can take different forms. One of the important forms is to synthesize the literature and identify the research gap for a specific research area. It provides a solid theoretical base for the proposed study and supports the arguments for the research problem. It also helps to justify that the proposed research can contribute something new to the current knowledge or validate the methods used in the study (Levy & Ellis, 2006). During the generic level 2 PSA development and societal and ethical discussion regarding nuclear plants, this form of the literature review is widely applied in the study. In order to conduct a broad literature review study, various databases and sources have been selected for the reference study. Science Direct, Scopus and Google Scholar are three major databases that have been used to search for scientific papers. Academic publications in the Journal of Nuclear Engineering and Technology, Nuclear Science and Technology, Nuclear Engineering and Design, Progress in Nuclear Energy, and Energy Policy have been selected for intensive study.

Furthermore, key documents published by IAEA, NEA, and USNRC regarding level 2 PSA development and social issues have been given a special focus for the research development.

Considering the research purpose is to use the generic level 2 PSA model to optimize mitigation management as well as taking societal issues into account, therefore, the first step is to investigate how to develop the generic level 2 PSA model and understand what are social and ethical problems during the plant operation. In order to develop the generic model, a wide range of papers which are related to level 2 PSA development have been studied. By searching keywords such as nuclear probabilistic safety analysis, nuclear containment safety analysis, etc., various level 2 PSA reports have been compared so that the research gap and model development strategy can be identified. During the model development, the general guidelines published by IAEA and NEA are selected as the main references to set up analysis procedures. In order to have a generic view of safety systems in pressurized water reactors (PWRs), a lot of reports regarding the design features of PWRs have been studied. By summarizing common design features among different types of PWRs, the summary for PWR containment safety systems have been made, which can be further used for containment response analysis. Additionally, in order to categorize accident progress paths based on their development mechanism, different types of containment failures have been detailed researched so that the release category can be developed. All these literature review works regarding level 2 PSA development have laid a solid foundation for model set-up, which provides a good reference for performing quantitative analysis.

However, most of level 2 PSA reports focus on technical risks that will affect the normal operation of nuclear plants. Therefore, for the societal and ethical issues associated with nuclear safety and mitigation management, papers related to energy justice issues and nuclear social acceptance issues are selected instead for further study. In order to have a state-of-art knowledge of the current discussion in academia, most papers with the publish date after 2000 have been selected for further study. Furthermore, keywords such as ethics of nuclear energy, nuclear social acceptance, nuclear justice, public acceptance of nuclear energy, etc., have been used to search for corresponding articles. Around 70 papers have been studied and 30 papers which hold an interesting view of point regarding the societal and ethical discussion of nuclear energy have been cited. By studying above academic discussions, it helps to analyze the reason why societal and ethical issues are very critical for nuclear development. Different from system failures, the societal and ethical issues are much more complex and if these issues cannot be well considered, the cascading impact is very huge because it may challenge the societal stability and economic development. In addition, after a broad literature review, it also helps to understand the reason why a lot of nuclear risk management plans are insufficient to mitigate the consequence. Therefore, by studying these state-of-art works, it provides a good insight regarding how to properly tackle societal and ethical issues during risk management, which would greatly contribute to the optimization of mitigation management.

2.2 Quantitative Risk Analysis

Besides the literature review approach, quantitative risk analysis is another method that has been applied in this research. Quantitative risk analysis is a widely applied research approach to estimate potential risk exposure. By assigning the occurrence probability for identified hazards and determining the accident impact, the corresponding risk exposure magnitude can be obtained (Broder & Tucker, 2012). Among various quantitative risk analysis tools, the event tree is widely applied in nuclear risk analysis because it can be used to identify all potential accident scenarios and sequences in a complex system. Event tree analysis (ETA) is a top-down, logic modeling technique that shows all possible outcomes resulting from an initiating event by taking into account whether installed safety barriers are functioning or not. In this method, an initiating event such as the malfunctioning of a system, process, or construction is considered as the starting point and the predictable accidental results, which are sequentially propagated from the initiating event, are graphically presented in order (Hong, et.al., 2009). In general, an event tree is made up of three main components, which include an initiating event, probable subsequent events and the results caused by different event sequences. Because all subsequent event attributes are independent of each other, the final results for specific event sequence can be calculated by multiplying all the occurrence frequency of subsequent events in a specific path.

ETA is normally applied to analyze the propagation of the initiating events in the complex system. Due to different availabilities of safety features in the system, the initiating events can result in various consequences. Therefore, it is very suitable to evaluate the core melt accident propagation in the containment for level 2 PSA study. The main purpose of level 2 PSA is to investigate how core melt accidents will progress under different scenarios and identify potential radioactive release paths, which can be well reflected by using the ETA approach. Because containment emergency cooling systems and safety systems can mitigate the development of core melt accidents, they can be treated as safety barriers in the event tree and based on their functionality characteristics, accident consequences under different scenarios can be obtained. In this research, the event tree model has been used to develop plant damage state logic diagram and the generic containment event tree. For the plant damage state analysis, six top event attributes are selected, which include containment bypass status, containment isolation status, reactor coolant system status, reactor coolant system pressure, safety injection and primary system depressurization status, and containment safety system status. These six top event attributes are recommended by IAEA because they basically cover all major sequences that will impact the core melt propagation. The first two attributes are used to differentiate the accident under the situation whether the containment is intact or not at the very early stage. The reactor coolant system status and the reactor coolant system pressure are used to represent different core cooling scenarios, which will impact the mechanism of core melt propagation in the containment. The rest three attributes are major accident mitigation measures in the containment and the mitigation effectiveness is decreasing in order. Due to the different safety features of these systems, it will result in different accident results. Based on their core melt propagation mechanisms, these accident results are further grouped into the plant damage states, which function as the entries for the containment event tree.

The containment event tree is designed to analyze the containment response and estimate the release consequence under different accident scenarios. Eight top event attributes are selected to represent the accident progression in the containment. Different core melt mechanisms will cause different forms of reactor ruptures and they will directly impact the radioactive release consequences. Therefore, it is necessary to consider the reactor status and corium release amount during the containment event tree development. when the corium is released outside of the reactor, the top events such as early containment status and different containment safety system status, are used to determine how these accidents will propagate in the containment and what type of containment failures they may result in. In this way, the final release consequence under different accident scenarios can be obtained. In addition, through the weekly discussion with experts, the applicable value of the event tree model is also examined. By developing the containment event tree, it helps to understand different accident progress phenomena and potential containment failures. It also helps to optimize risk mitigation management because it can provide a good reference for potential improvements in safety measures.

After the development of the containment event tree, the historical data are used for determining the probabilities associated with the branches and corresponding release fractions.

NUREG-1150 is the major source which is consulted to determine branch frequencies because it covers most of PWR event probabilities and it is also perceived as one of the important documents for level 2 PSA development. The TMI Unit1 level 2 PSA report published by USNRC in 2007 is another document that is consulted for release fraction calculation because it is an updated database for nuclear inventory release fraction calculation. By consulting these established reports and their data sources, the release frequency and release consequence for different event paths can be determined for the event tree. Based on different risk magnitudes associated with accidents and their development mechanism, nuclear risk mitigation management can be further optimized.

By performing the literature review and quantitative risk analysis in the research, this research provides a clear overview of the nuclear risks from both technical and societal aspects. By developing a generic level 2 PSA model and investigating societal and ethical issues regarding nuclear risk management, this study greatly contributes to the optimization of nuclear risk mitigation management.

Chapter 3 Generic Level 2 PSA Development

3.1 Reference Reactor Type Selection

At the end of 2017, around 82% of operational nuclear power plants are light water reactors (Sornette, Wolfgang & Spencer, 2017). Among these light water reactors, 80% of them are pressurized water reactors (PWRs), which indicates that the majority of current operational nuclear power plants are PWRs. Therefore, if a generic level 2 PSA model can be developed for pressurized water reactors, it will benefit a wide range of operational nuclear power plants and it may also help to speed up the precursor analysis process for the nuclear industry.

3.1.1 Pressurized Water Reactor

Pressurized water reactor (PWR) is a typical design of light water reactors (LWRs), which is normally cooled and controlled by borated water during the electricity generation (Thomas, 2019). In order to remove the heat generated from the reactor core, two water circuits are used to pump the primary coolant into the core (Cummins & Matzie, 2018). The removed heat is transferred to the secondary coolant system for steam generation. By using the steam to power the turbine, the electricity can be continuously generated.

The first PWR was commercially launched in 1957. After that, different types of PWRs have been designed and developed for more than three generations. Generation I PWRs are perceived as early prototypes of PWRs and the typical examples for Generation I PWR are nuclear power plants in Shippingport power station. Generation II PWRs refer to those nuclear reactors that are built by the end of the 1990s and most of them are currently in operation. In general, the operational life of Generation II PWRs is designed for 40 years and most operational Generation II PWRs in the West are normally manufactured by these three companies: Westinghouse, AREVA, and General Electric (Goldberg & Rosner, 2011). Typical examples of Generation II PWRs include all of the UK' s operational PWRs, Chinese CPR-1000, French N4, Russian VVER-1000, and Korean OPR-1000, etc. Later on, based on the design of Generation II PWRs, Generation III PWRs are developed with the implementation of many evolutionary improvements and the lifetime of Generation III PWRs is designed for 60 years. AP600 and APR1400 are two examples of Generation III nuclear power plants. Moreover, beyond these three generation PWRs, there are also Generation III+ PWRs such as AP1000 and EPR. Compared with Generation III PWRs, they have significant improvements in safety systems. After the above detailed discussion about different generation PWRs, the table below summarizes several examples of different generation PWRs.

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

In practice, most of these design prototypes are modified slightly based on different requirements, which makes each individual risk analysis difficult to be compared with each other to summarize the common failures for safety improvements. Therefore, if a generic level 2 PSA can be developed by cross-comparing different PWR designs, especially the containment safety systems, it will not only help to optimize risk mitigation management but also reduce the complexity of risk assessment for release path identification.

3.2 Plant Familiarization for Level 2 PSA

As discussed in the previous section, the pressurized water reactor (PWR) is the dominant reactor type in the current nuclear industry. In order to perform level 2 PSA for PWRs, it is important to have a good knowledge of the structure of PWRs. The major components of PWR are consisted by the reactor vessel, the pressurizer, the reactor coolant pump, the steam generator, and the connecting piping. The reactor vessel is perceived as the key component of PWR because all nuclear fission reactions take place within it and it is the main house for the core barrel, the reactor core, etc. The steam generator is used between primary and secondary coolant loops for picking up the heat generated from the reactor vessel. Within the steam generator, hot reactor coolant flows through multiple tubes and exchanges heat with the outside feedwater. After the absorption of the heat, the secondary coolant system starts to generate 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 used to bring the pressure back to normal values in case that 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 so that the steam within the pressurizer will get compressed. This will later on lead to the pressure increase in the pressurizer. In order to bring the pressure back to the acceptable value, 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 the emergency core cooling system (ECCS), the containment safety system and so on. Generally, all PWRs are equipped with an emergency water feed-up system in case that normal feed-up is lost or a major break in the reactor coolant loop. These systems are named as emergency core cooling systems. 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 into the coolant system. It is normally performed in a short period of time to maintain post-accident core cooling after the initiation of LOCA. The other purpose of ECCS is to ensure the reactor not produce the power after cooldown by injecting the corresponding coolant into the coolant system. In general, the ECCS includes two types of systems which are the high-pressure injection system and the low-pressure injection system. The high-pressure injection system automatically functions 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. When the coolant loss capacity exceeds the range that the high-pressure injection system can control, the low-pressure injection system can be used to mitigate the accident because the coolant system depressurized at 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 emergency core cooling mode, a low-pressure injection system also can function to remove the heat from the core for a longer period of time. It can take the water from containment sump to pick up the residual heat especially when the coolant water storage tank goes empty.

If the above systems all fail to ensure the safety of PWR, containment and its safety systems are perceived as the last barriers to mitigate the accident progression and confine the radioactive substances within the containment. 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. Therefore, the containment should be designed to withstand the pressure, temperature and the mechanical loading induced by the ejection of high-energy fluid. 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 when they are needed. That is the reason why the containment safety system is designed as the standby mode during the normal operation. However, if the accident occurs, these systems must stay effective 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.

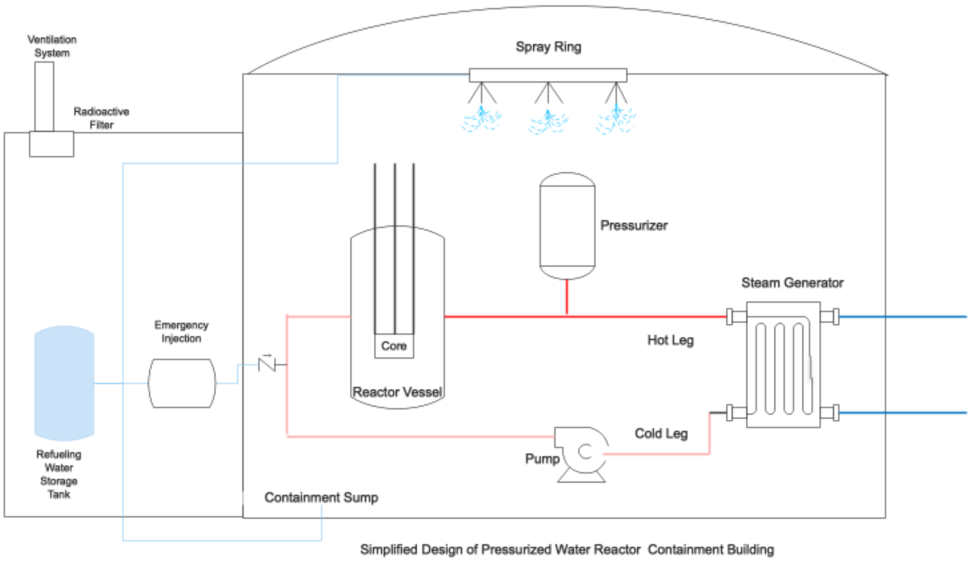

Figure 2. The simplified layout of common type PWR containment building

After a general overview of PWR structures, it is also imperative to have a detailed look at the containment safety systems and their functionalities considering that level 2 PSA is designed to investigate the containment response regarding the accident progression. Containment safety systems are normally used to prevent and mitigate the potential nuclear accidents that may lead to huge environmental impact. Based on their safety features, they can be classified into pressure and temperature control systems, hydrogen control systems and radioactive release control systems. The purpose of these systems are summarized as follows:

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 an accident (e.g. LOCA)

3.2.1 Containment Temperature & Pressure Control

To maintain the temperature and pressure of the containment, several systems have been designed for PWRs, including the containment spray and its sump water recirculation system, the air cooler system, the ice condenser system, and the passive cooling system. If accidents occur in the containment, these systems are effective on-demand. Different PWRs may implement different systems for temperature and pressure control. For example, only several American PWRs have the ice condenser system and the passive cooling system is only available for Generation III/III+ PWRs. In addition to those above systems, the containment structure and its volume are also designed to withstand certain pressure and temperature in case of any severe accidents. As an inherent safety design feature, the volume of the containment envelope determines the maximum pressure it can hold. In order to further investigate the temperature and pressure control features in the containment, the table below shows a generic classification of primary temperature and pressure control systems among different 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, EPR	III+

Table 2. The generic classification 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:

3.2.1.1 Containment Spray Systems

The main function of the containment spray systems is to remove the heat from the containment building in case of severe accidents. By using water spray to control the temperature and pressure of the containment, it helps to mitigate the accident progression and prevent the containment from fast overheating. According to the IAEA containment design guideline (2014), containment spray systems should be designed to make the water effectively interact with the steam in the whole containment. In general, containment spray systems 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 the containment atmosphere whenever the temperature goes up. Also, it is necessary to ensure the nozzles are able to function in case of any clogging issues. For recirculation systems, a large storage tank for the water supply is required. 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 happens, the containment spray system is designed to automatically start and take the water from the refueling water storage tank. Even if the storage tank is empty, the containment spray system should still be able to take the water from the containment sump during the recirculation mode. 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 cool down the containment atmosphere.

3.2.1.2 Air Cooler Systems

Air cooler systems are designed to cool down the containment atmosphere through the fan coolers. The heat removal capacity is the key parameter that needs to be considered during the design of fan coolers. Different from water spray, this kind of air circulation cooling mechanism is normally considered as the assisting approach for the containment temperature control in most of nuclear power plants.

3.2.1.3 Passive Containment Cooling System (PCCS)

As the main passive safety system in the containment, passive containment cooling systems are widely applied in the advanced designs of PWRs. The mechanism of these systems is to use the naturally induced airflow and water flows to cool down the containment. If the heat is released through the core-melt accident, the water flow and airflow can work together to provide the evaporative cooling for the containment so that the containment temperature can be brought back to the acceptable level. However, these passive cooling systems are not well implemented in current operating PWRs.

3.2.2 Containment Hydrogen Mitigation

In addition to the overheating and over-pressurized issues that may challenge the containment integrity, hydrogen detonation and deflagration are other big concerns for containment safety management. In the normal operation, hydrogen and oxygen are generated from the water radiolysis. However, in severe accidents, hydrogen is mainly generated from the oxidation process between zirconium and steam. The zirconium is the material used in the cladding and fuel element structures. When the core degraded, the heated zirconium can react with the steam producing a high local concentration of hydrogen in a very short period of time. In severe accidents, hydrogen can be produced even at the speed of 5 kg/s due to the fast zirconium oxidation (Bal, 2012). Under this scenario, the hydrogen combustion can create rapid pressure increase so that the detonation forces will exceed the containment design limits and result in the early containment failure. But the hydrogen can also be produced during the long-term pressure build-up process, in which the hydrogen is mostly generated from the interaction between the molten corium and containment basemat concrete. Different from the fast oxidation, this long-term pressure build-up may lead to the late containment failure.

In order to prevent the potential hydrogen accumulation and hydrogen combustion accident, several mitigation measures are introduced during the design of PWR containment, which includes mixing, deliberate ignition, catalytic recombining, and inertion. According to IAEA (2011), there are no strict regulatory requirements for the implementation of the hydrogen mitigation system, which means not all current PWRs have implemented these above mitigation systems. For example, the 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 it even varies among different plant generations within a country. The table below shows a generic classification of different hydrogen control system implementation situation among various generations of PWRs.

Function	System	Type of design (example)	Generation
Hydrogen mitigation and control	Hydrogen mixing system and hydrogen	Most European PWRs back-fitted	II, III, III+
	Hydrogen ignition system	AP600, AC600, AP1000, HPR1000,	II, III, III+

Table 3. The generic classification of hydrogen control systems in PWRs (NEA/CSNI/R(2014)8, 2014)

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 a potential explosion. (hydrogen ignition systems.)

3.2.2.1 Hydrogen Mixing Systems

Hydrogen mixing systems are designed to prevent the local hydrogen accumulation within the containment. Hydrogen mixing fans are the primary components of the mixing system. By activating the mixing fan, it can suck the hydrogen from the top of the containment and create the turbulence to delude the local concentration. In this way, the concentration level will quickly drop below the flammability point. Besides 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 the hydrogen venting 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. Different catalysts have been used in catalytic recombiners to control the hydrogen oxidization process. In general, catalytic recombiners can be classified into two categories: conventional catalytic recombiner and passive autocatalytic recombiner (PAR). Based on the design guideline published by NEA (2014), conventional catalytic recombiners share a similar working mechanism with electric-powered thermal recombiners regarding the hydrogen concentration reduction process. Most of conventional catalytic recombiners are situated outside of the containment. In order to maintain their reliability and basic functions, multiple testing and complex supporting systems are required.

Different from conventional catalytic recombiner, passive catalytic recombiners have higher reliability and require no human intervention because the heat that has been captured from the oxidation reaction is used to produce natural convection flow through the unit (IAEA, 2001). Passive catalytic recombiners are made up by catalyst surfaces with an open-ended enclosure. When the mixed hydrogen and oxygen reach the surfaces, the oxidization reaction automatically takes place. The reaction heat will create the convection flow to exhaust the hydrogen-depleted air to the containment and suck more combustible gases from below. However, the processing capacity of these systems is highly dependent on mass transfer limitations and the preferred working condition for PARs is to start under the cool conditions.

3.2.2.2 Hydrogen Ignition Systems

Hydrogen igniters are normally used to ignite combustible gases when the flammable mixtures start to accumulate in the containment. In case that the flammable mixtures get ignited by random sources, it will be much better to use igniters to deliberately ignite them so that the combustion process is under control. Through the gentle deflagration, the hydrogen combustion risk can be well mitigated. In general, when the hydrogen release rate exceeds the processing capacity of other hydrogen mitigation systems, such as hydrogen mixing systems or hydrogen recombiners, hydrogen igniters are activated to prevent hydrogen accumulations. However, there are also potential risks associated with this ignition process. Because even if the deflagration is initiated at one small point, there are still chances that the deflagration can propagate to other containment areas. Especially when it propagates to the hydrogen release point, it may lead to huge consequences because the deflagration speed is uncontrollable. Therefore, a detailed and careful placement analysis for ignition systems is required. From the current application status, it is found that the potential combustion risk could be well controlled if they are coupled with spray systems (NRC, 1983).

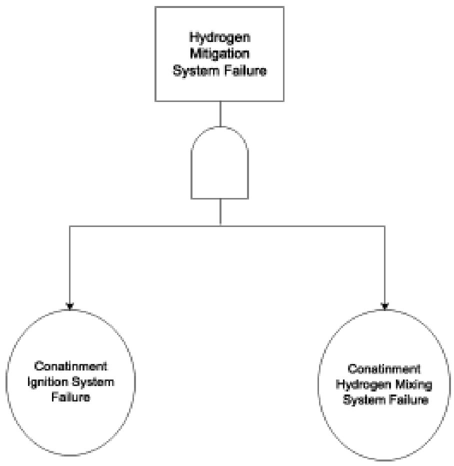

Figure 3. The simplified fault tree for hydrogen mitigation system in PWRs[1]

In addition to the catalytic recombining and ignition, inertion is the third approach to mitigate the hydrogen risk, which can be classified into pre-accident inertion and post-accident inertion. The criteria to choose which inertion methods mainly depends on the plant requirement. For example, if some nuclear plants require a hydrogen-burning risk-free environment, then it is imperative to create an atmosphere with oxygen-depleted before the normal operation. Considering the nitrogen is very stable in normal conditions, therefore, it is widely used as the common inert gas. By injecting the nitrogen into the containment to replace the air, the oxygen concentration can be reduced far below the required level to induce potential combustion. In this way, a hydrogen-burning risk free environment can be created. However, due to the technical issues and other factors, inertion is not perceived as the main hydrogen mitigation measures in large PWRs.

3.2.3 Containment Radioactive Release Control

Containment systems are 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 get leaked, it will probably lead to unacceptable consequences to the environment. Activated corrosion products (ACPs) are generally perceived as the primary radiation sources in most nuclear power plants. But if there are a lot of fuel cladding failures, it will also increase the amount of fission product release

[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.

(IAEA, 2005). These two major radioactive sources are all originated from the core and they are transported to the containment along with the accident progression. Therefore, in order to well confine these potential releases, containment structures and their supporting systems should be as reliable as possible no matter when the accident will happen.

Among various containment supporting systems, containment isolation systems serve as last barriers for the containment radioactive release control, which consist of actuators, isolation valves, and connecting pipes. During the normal operation, all these components are in standby modes. However, when the accident occurs, if the radioactive release control is required, they are activated to shut down all the release paths in order to reduce potential radionuclide escape. Due to the safety concern, these systems are required to work in an independent mode during the accident mitigation process. In addition to the isolation systems, there are also other supporting systems that help to mitigate the radioactive release, such as spray systems and filtered ventilation systems. In general, the radioactive release is in the form of aerosol. Therefore, two basic mechanisms are applied in the nuclear power plants to manage the aerosol nuclides: 1. agglomeration and gravitational setting. 2. spray removal. For the gravitational setting of nuclide management, IAEA (1999) has recommended the revised model by introducing the removal rate as the function of time and the revised model can be viewed as below:

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

C is the mass concentration and λ is the removal rate coefficient. In general, the aerosol source strength and its corresponding concentration level are two important factors that determine the removal rate coefficient. In order to understand how the agglomeration mechanism works, two different aerosol source states have been proposed by IAEA (1999), which are large continuing source state and weak or zero source state. The criterion to determine which state should be used for the model is given 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 determined by spray volume, flow rate, and droplet size. When the radionuclides get released to the containment, the particles can be dissolved by the water spray and the dissolved particles will flow along the containment wall to the containment bottom, which will greatly reduce the potential radioactive substances during the accident.

However, if these two mechanisms fail to reduce the radioactive release and there are still airborne radioactive substances left in the containment, the filtered venting systems are designed to purify the gaseous mixture before they escape to the environment. Currently, filtered venting systems are widely applied in new PWR designs and most European nuclear power plants have been back-fitted these systems after the stress test (Bal, Jose & Meikap, 2019). Filtered venting systems are mostly used for filtering the containment aerosols before they are discharged into the environment. In order to minimize the radioactive release, filters must be designed to guarantee the release concentration is always below the acceptable level. Moreover, their supporting systems should also be available to keep the temperature of filter inlet air above the dew point so that the filters can properly function to reduce the radionuclides. The common design types of filter systems include sand bed scrubber system, multi-venturi scrubber system, and charcoal filters (NEA, 2014). However, besides the filtering the aerosols, filtered venting systems can also be used to mitigate over-pressurization in case that the core melt accident occurs. For example, hydrogen buildup depends on the effluent rate and pressure. Venting can greatly reduce the pressure and filter the radioactive nuclides so that it can effectively reduce the accident consequences. Therefore, the ventilation system needs to be activated all the time in order to balance the containment pressure with the outside environment. From the above discussion, it can be seen isolation systems, spray systems, and filtered venting systems are three major containment safety systems that contribute a lot to the radioactive control in case of accidents. Therefore, the table below summarized these radioactive control systems among different PWR generations for future comparative study.

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

Table 4. The generic classification of radioactive release control systems in PWRs (NEA/CSNI/R(2014)7, 2014)

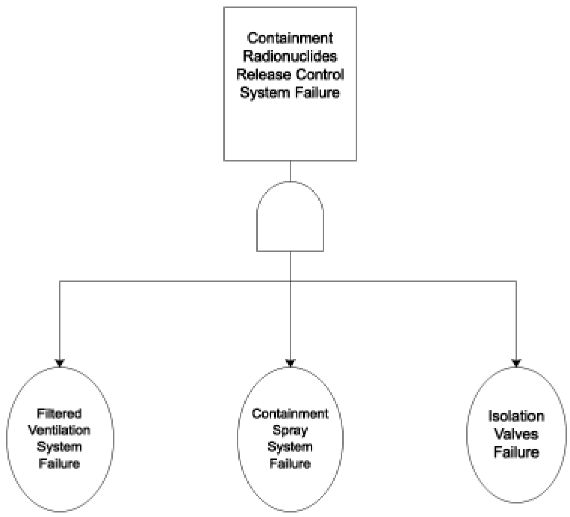

Figure 4. The simplified fault tree for the radioactive release control system in PWRs[2]

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.

3.2.4 Summary of Containment Safety Systems

After a very detailed discussion and analysis of the containment safety systems, the commonly applied PWR safety systems have been identified, which can be used to investigate the containment response during the accident progression analysis. By taking these commonly applied safety systems into consideration, the proposed containment event tree will be more applicable for a large variety of PWRs. In addition, by applying the proposed containment event tree, it also can help to provide a general overview of accident mitigation measures under different scenarios. Therefore, in order to better facilitate accident progression analysis, containment safety systems of different types of PWRs have been selected for cross-comparison. The commonly applied PWR containment safety 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 system	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 State Identification

Aiming at better investigating how accident progression will impact containment integrity and radioactive release, plant damage states (PDSs) are designed in order to classify different failure events that have been identified in the level 1 PSA. As the interface between Level 1 PSA and Level 2 PSA, PDSs contribute a lot to defining the initial and boundary conditions for severe accident analysis. One of the major reasons is that grouped PDSs are the entries for the containment event trees. By reducing a large number of fault sequences into several plant damage states, it makes the severe accident analysis more concise and manageable. However, in order to develop the PDSs, some key attributes to the accident progression should be well considered, such as the primary system status, operation system variables, and accident initiators, etc. Therefore, the general PDSs attributes for PWR 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 has been developed in the research, from which corresponding plant damage states can be obtained. In this research, it was assumed the nuclear power plant is under full power operation when the initiating events take place. The graph below illustrates how failure sequences form into different plant damage states.

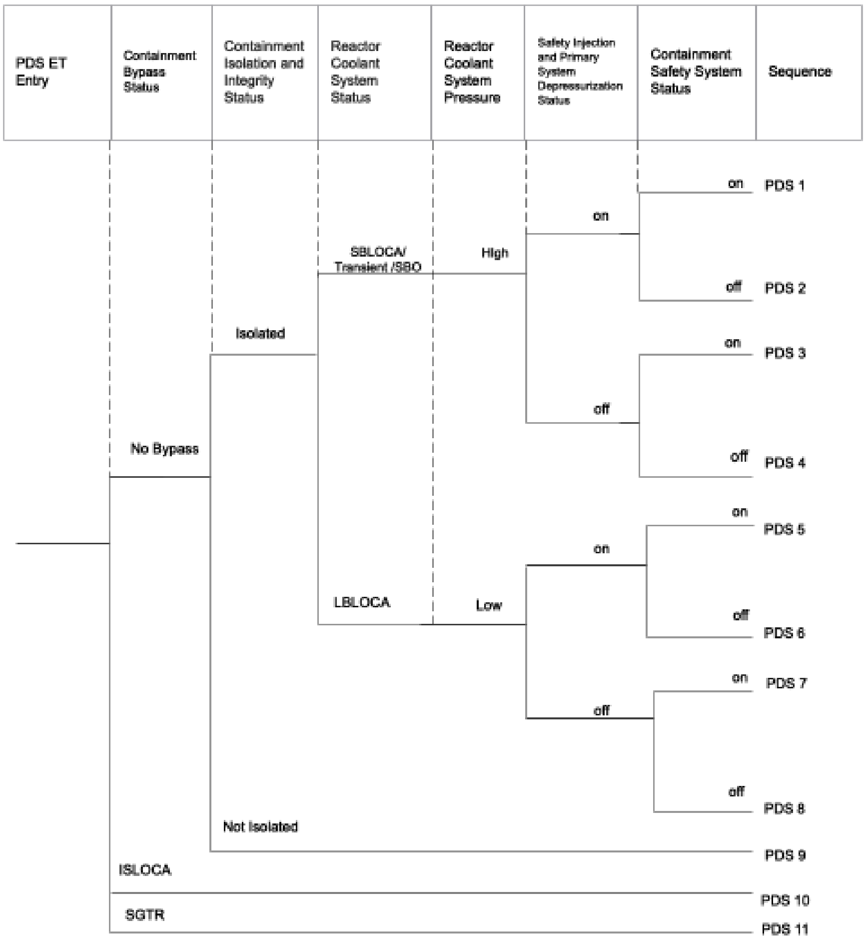

Figure 5. Plant damage states logic diagram

As shown in the graph above, six plant damage state attributes have been 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 cover all parameters recommended by IAEA for plant damage state identification (IAEA, 2010). The containment bypass status and isolation status are related to containment integrity. The containment could be bypassed because the rupture takes place in an interfacing system outside the containment or the steam generator tubes rupture inside the containment. These two kinds of bypass accidents both can lead to significant radioactive release to the environment. In addition, there are also two scenarios that need to be considered for the containment isolation status. First, if the containment isolation system fails before the core melt accident, once release accidents occur, it can directly cause a large amount of leakage. Second, the isolation system can also fail after the core melt accident. For example, when a severe accident happens, the isolation valves function as the last barriers to confine the radioactive release and mitigate the core melt accident. However, if the pressure or temperature exceeds the design value of the isolation valves, the isolation failure will occur.

Different from containment bypass status and isolation status, the attribute for reactor coolant system status is designed to investigate the initiating events of the accident. In general, loss of coolant and transient events are considered as initiating events for containment failure accidents. Loss of coolant accident (LOCA) can vary from the small leak in the reactor coolant system boundary to the large rupture in the primary coolant recirculation system. Due to the different size of the rupture, the system response also 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 in case that there is no further safety injection applied. Because of the large break LOCA(LBLOCA), the fuel clad is quickly overheated and ruptured, the released fission products will be gradually transported to the containment. However, depending on the containment safety system status, the release amount of fission products to the environment can vary a lot. For example, if all safety systems are available upon the occurrence of the accident, the potential release consequence will be very small.

Different from large break LOCA, small break LOCA includes not only small pipe breaks but also the inadvertent opening of relief valves (Myerscough, 1992). When the small break LOCA happens, the primary system still remains high pressure, but it will lead to slow depressurization. If all safety injection systems are in operation, the safety status of the nuclear reactor will not be challenged very much. However, if no safety injection systems are available, it may also lead to the reactor rupture soon due to the overheating and over-pressurization issues. In the plant damage analysis, the amount of coolant loss should be well considered because they can lead to different accident progression scenarios. For example, the size of the break in the primary system will determine the pressure status of the whole system and the pressure status of the primary system is very important for defining success criteria for in-vessel injection status. In the plant damage logic diagram, the in-vessel injection status is a very important attribute because it decides whether the core melt accident will progress out of the reactor vessel or not. If the core melt accident can be stopped within the reactor, the containment safety systems will be less important. However, if the core melt accident progresses into the containment, the containment safety system status will be the key factor to determine the accident progression. The containment safety system is closely related to mitigation and prevention of accident progression within containment. Therefore, its functional status will largely impact accident consequence. Based on the above discussion about the attributes selection, the plant damage state logic diagram can be built accordingly. By following the logic of these PDSs attributes, eleven plant damage states are identified 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 that can be applied for categorizing plant damage states. One is to group plant damage states based on their similarities and select one typical event sequence to represent each group of plant damage states for further analysis (NEA, 2007). The other one is to use cut-off criteria based on the occurrence frequency of each PDS so that plant damage state screening schemes can be developed. In this research, the first approach is applied because the second approach is only applicable when the level 1 PSA can provide reliable data results. Considering the data availability and validity, the first approach is more robust and it is also recommended by most of the related studies. Therefore, by using the first approach, 5 plant damage state bins are finally obtained. The detailed classification steps can be viewed as follows. First of all, all 11 plant damage states can be classified into two clusters: the first cluster stands for the situation where the radionuclide release occurs due to the accident progression and the second cluster stands for the situation where the containment safety systems are bypassed or the containment loses 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 core cooled group is used to define those in which the initiating events occur and all the risk mitigation systems function to stop the accident progression within the primary system. In PDSs logic diagram, it can be found PDS1, PDS2, PDS5, and PDS6 all have the feature that the core is cooled in the vessel. Therefore, these 4 PDSs should be grouped into in-vessel core cooled group. The core melt with high pressure stands for those in which initiating events occur but emergency injection and depressurization systems fail to bring down the pressure in the reactor, meanwhile, the core melt progression is not stopped. The typical scenario can be the injection system and recirculation system both fail when the transient or small break LOCA takes place. Therefore, according to the PDSs logic diagram, PDS3 and PDS4 both belong to this category. For the group of core melt with low pressure in the primary system, it describes those PDSs in which the primary system is depressurized by the initiating events and safety injection systems fail to mitigate the core melt progression. The typical example is the large break LOCA induced core melt progression and PDS7 and PDS8 share this similar release features in the PDS logic diagram. However, different from the first cluster, the second cluster only consists of containment isolation failure and bypass failure. Their characteristics have been widely discussed in previous sections. Therefore, according to the PDS logic diagram, PDS10, PDS11 are grouped into bypass failure and PDS9 is classified as isolation failure. After the above detailed discussion about the grouping scheme, the final results for plant damage bins can be viewed as below.

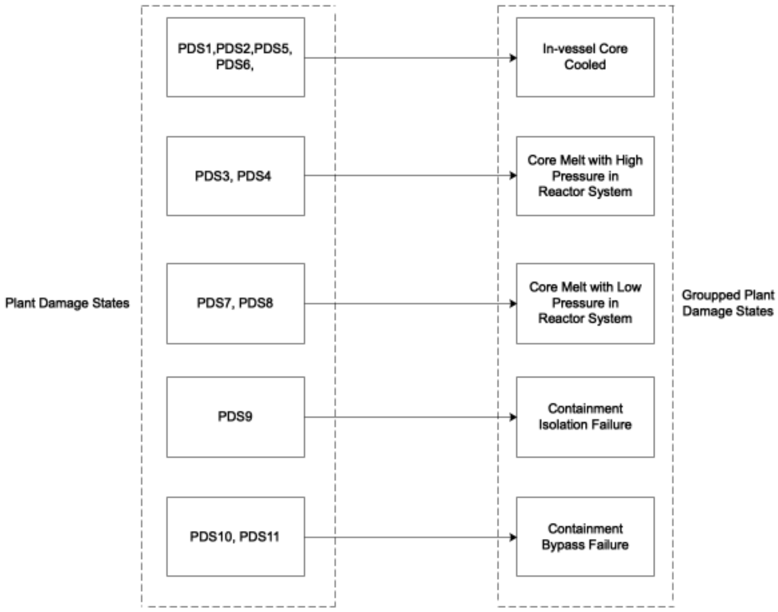

Figure 6. Plant damage sate grouping

3.4 Accident Progression Analysis

Accident progression analysis is used to analyze how core damage progression can impact containment integrity. As the key part of level 2 PSA, it investigates the containment safety system response under different accident scenarios. It also helps to present different radioactive release paths for identified PDSs. Based on the analysis results, it can help to optimize the risk mitigation measures in case of severe accidents. In order to perform the accident progression analysis, containment event tree (CET) is developed to examine how accidents could occur after severe initiating events and how these accidents can lead to the radioactive release.

As one of the key parts of accident progression analysis, it is recommended that an adequate number of nodes need to be defined in the CET so that the significant phenomena can be properly addressed in different time frames (NEA, 2007). Therefore, in this research, the CET is built through multiple event nodes and all these nodes follow a chronological sequence based on the accident progression. It starts from the core melt accident and ends at the end states of the containment failure in different time spans. In general, the core melt initiating events are the entry for the CET and events following the reactor vessel failure are used to analyze the containment response.

In principle, there are two approaches to build the CET. One is to create a big 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 fails to provide a general overview of how containment failure could occur in the whole system. Therefore, hereby a generic model is proposed in order to capture most of the scenarios that may impact the accident progression. In the model, eight top events have been selected to capture the significant phenomena during the accident progression, which include 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 eight top events represent the whole progression stage, which starts from a core melt incident through the reactor rupture status up to the late stage of radioactive control system response. They also present how different type of containment failures can occur during the accident progression because different event tree paths represent different containment failure scenarios. Moreover, due to the time effect, the containment condition also varies a lot in different accident progression stages. By following these eight top events and paths developed in the generic CET, there are 22 identified CET events in this research, which are presented in the graph below.

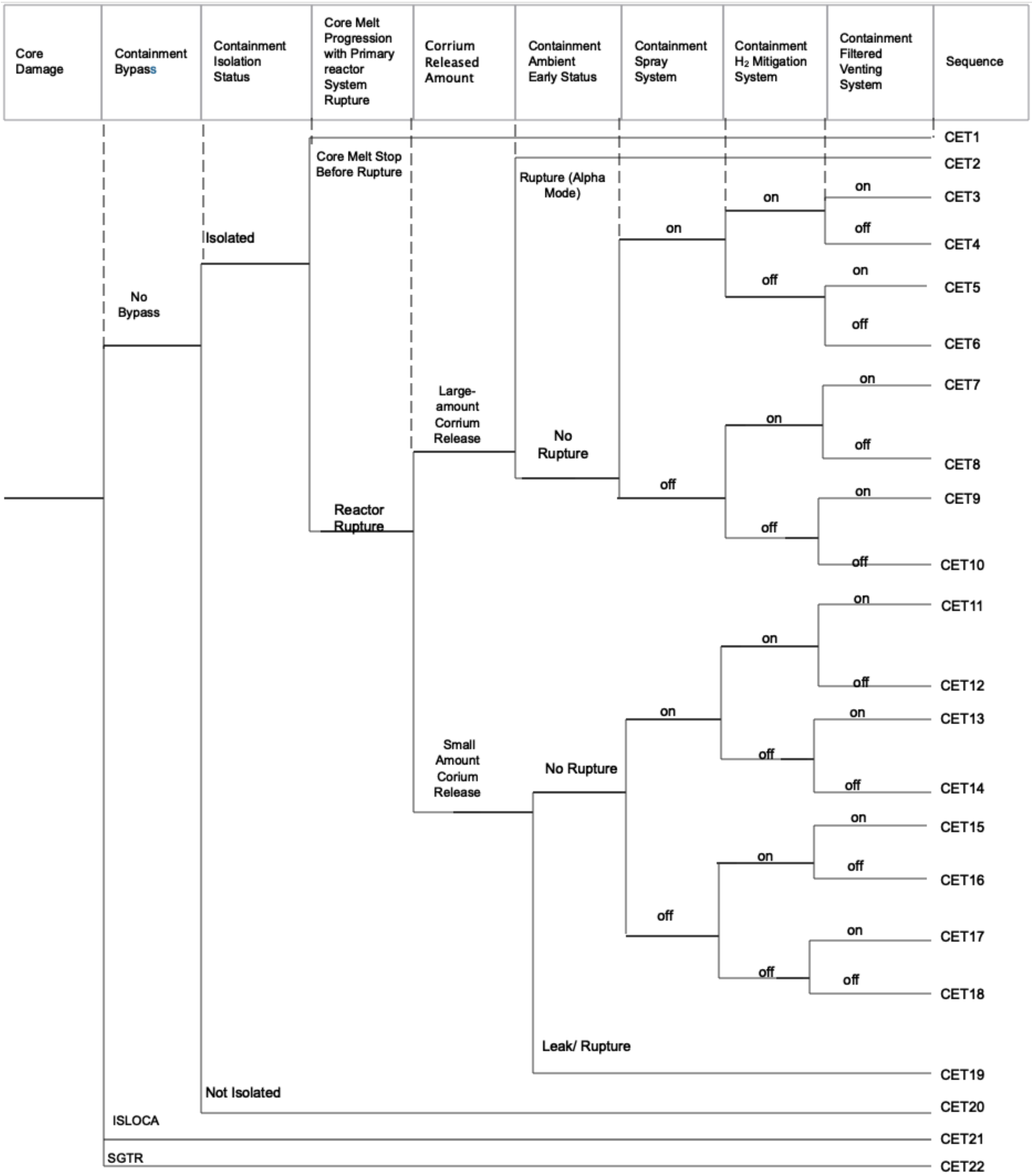

Figure 7. Generic containment event tree

Based on the analysis in the CET, it can be found containment bypass events are represented by CET21 and CET 22 because the typical containment bypass accidents are steam generator tube rupture (SGTR) and interfacing system loss of coolant accident (ISLOCA). For SGTR, it can be classified into spontaneous SGTR and consequential SGTR. The former one is normally caused by maloperation before the core melt accident (Song, et al., 2019). However, the later one is induced by the core melt accident. Hot gases from a damaged reactor core can overheat the tube and cause steam tubes ruptured (Song, et al., 2019). In recent years, several thermal and structural response studies have been made on the SGTR in order to well control this accident (USNRC, 2016). Different from SGTR, ISLOCA is perceived as the pipeline rupture at which the pipeline is linked to the reactor coolant system. If the radioactive substances get released to the primary system, ISLOCA can lead to the direct radioactive discharge to the environment without going through any filtering systems within the containment.

Containment isolation status is another big concern regarding the release consequence because it may cause the direct release of the radioactive substances without any filtering process. In the generic CET, it is represented by CET19 and CET20. In practice, 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. However, due to the very low probability of accidents during the operation, hereby the isolation failure is more referred to those valves and connecting system failures. If they fail to stop the radioactive release, there will be direct health and environmental impact on the local communities.

If the containment is not bypassed and it is isolated at the beginning, the only factor that may greatly influence the radioactive release will be the availability of the containment heat removal system and the containment safety system. Therefore, except for CET19, CET20, CET21, and CET22, the rest CET event paths are all used to represent the accident progression scenarios under different risk mitigation measures. Core melt accident occurs mainly because the energy generated from the fission reaction exceeds the heat removal capacity of the nuclear plant. 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 availability of the containment heat removal systems, different accident progression paths can be developed accordingly. For instance, when there is a large release of corium meanwhile the containment safety systems are not available, it is very likely to lead to early containment failures. However, even if the containment safety systems are available to mitigate the early containment failure, it still may lead to another two scenarios. One is the accident stops before the penetration through the containment and the other is the late containment failure. On the contrary, when it speaks about a small amount of corium release, the release impact will be less severe compared with the large amount release scenarios. Moreover, if the incidents are controlled in time, the damage and radioactive substance might be confined within containment, which will have the least environmental damage among other containment failure scenarios. But if the containment safety systems stop working after a while, even small release events can still lead to the late containment failure due to the corium accumulation along with the time. Therefore, the small release scenario also can result in the late containment failure.

From the above analysis, it can be seen containment safety system plays a vital role to mitigate the accident progression especially when the core melt accident progresses out of the vessel. Therefore, in this research, spray system status, hydrogen mitigation system status, and filtered venting system status are very important for the CET. By investigating the accident behavior under different safety system status, different release paths and accident characteristics can be obtained. By following through these different event paths in the CET, 22 containment end-state events are obtained in total, which basically cover most of the accident scenarios and release paths. These containment end-state events will be further classified into release categories so that the accident consequence can be well estimated in order to optimize the mitigation management.

3.5 Release Category Identification

Release category identification serves as the interface between the accident progression analysis and the source term analysis. In the previous section, 22 containment end-state events have been identified from the CET analysis. Based on accident progression characteristics, these events can be further classified into five different release categories, which are containment intact, early containment failure, late containment failure, bypass failure, and isolation failure. Hereby it is imperative to point out that not all core melt accidents would lead to containment failures. According to NUREG-1150 report, most PWRs have a relatively high probability to remain intact after the severe accident (USNRC, 1990). The grouping scheme of CET events and detailed descriptions regarding release categories can be viewed as below.

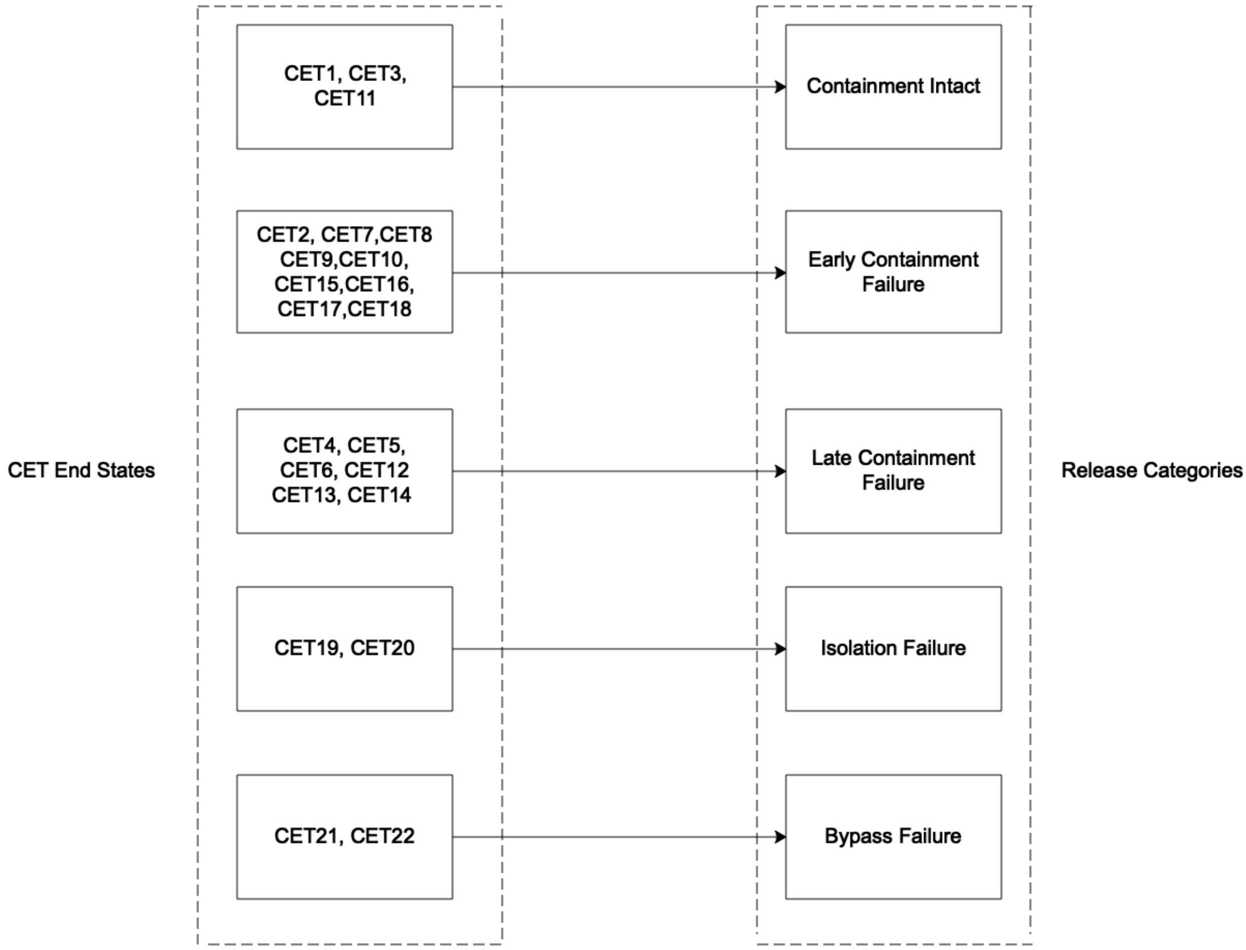

Figure 8. Containment source term release category grouping

Based on the developed CET and the accident progression analysis, it can be found CET1, CET3, and CET11 should be classified into the containment intact group. Because under these three event states, core melt accidents are well managed within the containment. CET9 and CET 10 should be grouped as isolation failure because they are induced by isolation problems. In addition, CET21 and CET22 should be classified as bypass failure considering they both are typical examples of bypass accidents. For the rest CET end-state events, CET2, CET7, CET8, CET9, CET10, CET15, CET16, CET17, and CET18 all share a similar feature as the early containment failure. For example, CET2 represents the rocket effect of the accident progression, which is very early containment failure. CET7, CET8, CET15, and CET16 represent the direct containment heating (DCH) due to the malfunction of the spray systems. Moreover, CET9, CET10, CET17, and CET18 represent the early hydrogen combustion because both spray systems and hydrogen mitigation systems fail to stop the accident. Considering all these three phenomena are typical examples of the early containment failure, that is the reason why those CETs should be grouped into the early containment failure.

However, if the spray system only functions at the early stage, there is still a high probability to have the containment failure, which is the late containment failure. For example, CET4, CET5, CET6, CET12, CET13, CET14 are classified into late containment failure. Because slow over-pressurization, late hydrogen burn and basemat melt through are three typical examples of late containment failure, the above 6 CETs should be grouped into the same release category. If the venting system fails, it may result in slow over-pressurization. That is the reason why CET4, CET6 CET12, and CET14 are selected. In addition, late hydrogen burn and basemat melt-through can also lead to the late containment failure, which are featured by CET16 and CET 5. In order to better understand the characteristics and release mechanism of each release category, the following sections are designed to provide a deep insight into different release scenarios.

3.5.1 Early Containment Failure

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

3.5.1.1 Direct Containment Heating

Direct containment heating (DCH) is one of the major causes that can lead to early containment failures. Under this scenario, the radioactive substances will be directly released to the environment because of safety barrier failures. It allows a very short time interval for taking any emergency actions. The heat exchange between metal particles and containment atmosphere and the energy released from hydrogen combustion are two energy sources that are accounted as major contributors to this phenomenon (Bal, 2012). However, in-vessel pressure will influence the energy release forms of direct containment heating. For example, if the in-vessel pressure is much higher than the containment pressure when small LOCA or station blackout accidents occur, it is likely to induce the high-pressure melt ejection into the containment cavity. Due to the oxidation of melt debris, the generated heat will immediately pressurize the containment. These above complex processes are typical examples of the DCH and that is the reason why DCH is perceived as one of big threats for containment integrity.

3.5.1.2 Steam Explosion

Steam explosion is another type of early containment failure incident. Because of the rapid fragmentation of molten fuel, the released energy quickly transfers to the coolant, which will generate a lot of steam and induce shock waves. However, if there is a large amount of molten fuel getting fragmented in a very short time interval within the confined reactor vessel, the missile effect will get generated. The energy that is released from these explosions could damage the containment and result in a very early release of radioactive substances. But based on experts' judgments, the probability of this kind of in-vessel explosion accident is quite low so that it is not considered as a credible threat for the containment failure.

Different from very early containment failure, another steam explosion scenario is the ex-vessel steam explosion, which is aroused due to the incident that the molten fuel dropped into the water outside the vessel. The generated shock wave can progress through the water and damage the containment mechanical structure. But the damage scenarios vary a lot among different types of nuclear power plants and it seems these potential explosion scenarios have less impact on PWR large dry containment.

3.5.1.3 Hydrogen Combustion

Hydrogen combustion is another big threat to containment integrity because it can result in the rapid temperature and pressure rise in the containment. Once the increase exceeds the containment design boundary, the radioactive substances will be directly released to the environment. During the core melt accident, the hydrogen mainly comes from the oxidation process between the steam and zirconium. Especially when the reactor core is uncovered, the zirconium is very likely to be heated to a very high temperature so that the hydrogen concentration will quickly climb up.

In addition to the hydrogen generation, hydrogen mixing and transport mechanisms are also crucial for the combustion analysis because different transport mechanisms 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 can be widespread around the containment. On the contrary, if the mixing process is relatively slow, it is likely to result in the localized gaseous mixture and localized burning. Therefore, the hydrogen release rate may determine the combustion type. Due to the overheating and over-pressurization issues, the hydrogen combustion can pose a great threat to the containment structure if the gaseous mixture gets ignited. Two typical hydrogen combustion types are deflagration and detonation. Deflagration is the phenomenon that unburned gases get quickly heated up due to the conduction. One of the important characteristics of deflagration is that the generated combustion waves travel subsonically, which will create static loads on the surrounding structures. However, detonation is the combustion phenomenon that the gaseous mixture gets instantly compressed and heated up. Therefore, its combustion waves travel supersonically, which can induce both static and dynamic loads on structures. Direct initiation and flame acceleration are two ways to induce detonation. But direct initiation normally needs high energy to start, for the containment structure, it is very difficult to induce this amount of energy. Therefore, in severe accidents, only flame acceleration is likely to result in detonation. As flame acceleration is dependent on the system geometry, the detonation likelihood will also vary a lot due to different containment designs.

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 refers to the failure that takes place in a late fashion after the core molten debris are released from the reactor vessel. Gradual/ slow over-pressurization, late combustible gases burning, and basemat melt through are three typical late containment failures.

Slow over-pressurization takes place when the containment heat removal systems stop working or venting systems fail to balance the containment pressure. It is mainly induced by the steam generation and non-condensible gas emission. Non-condensible gases and steams are normally generated from the molten core and concrete interaction (MCCI). During the MCCI, the core debris interacts with containment concrete, which will generate steam and carbon dioxide. However, MCCI is a complex process, which depends on multiple factors including the amount of water in the containment cavity and the coolability of debris. If the cavity is dry at the beginning, the core debris could remain hot to continue interacting with the concrete and non-condensible gases will get released. If the cavity is flooded before the vessel rupture, the scenario will be different because the fallen molten debris will be cooled and fragmented very fast. If coolable crusts get formatted first, it will cause very little gas emission. Because boiling water absorbs all the decay heat, the core concrete interaction is prevented due to the low temperature. But if the water flow cannot be continuously provided, later on, the core debris will continue interacting with concrete and emit gases and steams. Therefore, it must ensure that containment pressure control systems are always available on demand. In addition to the slow over-pressurization, basemat melt-through is another big concern for late containment failure, especially speaking of the underground water contamination induced by this accident (Bal, 2012). When spray systems function to mitigate the early containment issues, the large amount of fission products will be dissolved into the containment water. Thus, if the containment basemat melt-through occurs, there will be a great fear that the fission products will get released into the underground water.

3.5.3 Bypass Accident

Containment bypass accidents refer to release accidents that occur outside the containment or bypass the containment safety systems. Even though the containment building is intact, fission products can still get released to the environment. Therefore, the release consequence of this type of accident is relatively high due to the direct release. Steam generator tube rupture (SGTR) and interfacing system loss of coolant accident (ISLOCA) are two typical examples of bypass accidents.

3.5.3.1 SGTR

Steam Generator Tube Rupture (SGTR) is the accident that the heat transfer tube ruptures in the containment. But it leads to the radioactive release bypassing the containment safety systems. SGTR is characterized as an accident which has a high frequency of occurrence and big consequence of radioactive release (Zhang & Chen, 2017). When SGTR occurs, the shear fracture will make multiple tubes ruptured in a very short time. Due to the pressure difference between the primary system and secondary system, a large amount of coolant will flow into the system with lower pressure. Compared with the primary system, the secondary system has a relatively slow flow rate and lower pressure, therefore, the water in the secondary system can easily be contaminated by the coolant. In this way, radioactive substances may get released into the environment after the core melt accident. In order to estimate the radioactive release from the SGTR accident, two factors are suggested to be taken into consideration. One is the core fraction which is released to the secondary system through the steam generator. The other is the core fraction which is released to the environment through the relief valves (Song, et al., 2019).

3.5.3.2 ISLOCA

ISLOCA is another type of bypass accident, which normally occurs outside of the containment. Low-pressure piping isolation valve failure is a typical example of ISLOCA. When the failure occurs at the interfacing between the reactor coolant system and the low-pressure supporting system, the supporting system will be quickly over-pressurized. It may lead to direct radioactive release to the environment. Especially when the safety injection is impossible during the recirculation phase and no further RCS isolation is applied, it is very likely to cause a large radioactive release accident. Therefore, it is imperative to increase the reliability of interfacing provisions between reactor coolant systems and the low-pressure supporting system so that the potential risks of direct release can be reduced.

3.5.4 Isolation Failure

When the accident occurs, the containment isolation systems are designed to confine the radioactive release and prevent any potential releases to the environment. The containment isolation systems include valves, actuators, filters, and piping. The common isolation failures are the containment breach or malfunction of the isolation components. They are normally induced by either mechanical problems (e.g. isolation valves fail to open and close) or human errors (valves are forgotten to be closed). If the isolation failure cannot be mitigated on time, once the radioactive substances leakage rate exceeds the expected value, it will cause big damage to the public health and the environment. Therefore, it is very important to take isolation failure into account.

3.6 Source Term Analysis

Source term analysis is one of important components in Level 2 PSA, which is normally used to determine the radioactive substance of the accident. As the interface between level 2 PSA and Level 3 PSA, it represents various release categories and their release consequences. In order to determine the release consequence, one of the key parameters to be considered is the fission product release amount, which includes the isotopes and inventory fractions. Therefore, it is important to understand what kind of fission products will be released and their transport mechanism.

When the cladding failure happens, the volatile fission products will gradually escape from the fuel rods and pellets to the primary system. If the core melt is initiated, nearly most of the volatile products will be released to the containment and the released mass of fission products will be very large. Taking the French 900 MWe PWR analysis as an example, the fission product release mass could even reach 1500 Kg (IRSN, 2015). During the radioactive release, the major volatile fission products are Iodine [I], Caesium [Cs], Bromine [Br], Rubidium [Rb], Tellurium [Te], Strontium [Sr] and noble metals [Ru]. Among these volatile fission products, Iodine should be given special attention because of the radiological consequence, which is mainly determined by the physical and chemical conditions. It is important to understand the physical and chemical forms of iodine when it is released after the accident. After the core melt accident, the chemical forms of the iodine are molecular iodine (I_2), organic iodine ([CH_3I]), and caesium iodide ([CsI]). When these iodine substances get released from the primary system to the containment, only the gaseous molecular iodine (I_2) can be quickly adsorbed by the containment wall. After a short reaction, molecular iodine will be transferred into organic iodine. However, the organic iodine is very difficult to be absorbed even by using the filtration system (IRSN, 2015). Therefore, it is very important to monitor organic iodine transportation after a severe accident. However, besides the Iodine, there might be other radioactive substances that may impact the environment as well, hereby the major radioactive substances after the release accident 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 so that the release consequences can be better estimated. Hereby, the data from TMI unit 1 level 2 PSA are taken as the data source for radionuclide release consequence calculation. The TIM Unit 1 level 2 PSA was performed in 2007 aiming at better investigating the large dry PWR containment response during the accident progression (USNRC, 2007). In the report, it includes the failure probability of systems, plausible accident scenarios with the estimated frequency. It also investigates the core degradation physical process and fission products release scenarios. In order to develop the release category, all accident sequences which contribute to the core damage are grouped separately according to their release characteristics. In total, it creates nine release categories for PWR plants. For each release category, it consists of multiple different event sequences and the detailed description of these sub-event sequences for each release category can be viewed in Appendix A. The summarized 9 PWR release categories of TMI Unit 1 level 2 PSA are developed 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 for release categories, each release category proposed by TMI Unit 1 level 2 PSA has been compared with the release categories that have been proposed in the generic level 2 PSA model. According to the generic model, five release categories are developed in total, which include: the containment intact, early containment failure, late containment failure, isolation failure and bypass failure. Therefore, by comparing the scenario description with the release categories in TMI Unit 1 analysis, it is found that the previous nine release categories can be further grouped into these newly proposed five release categories. The release category mapping scheme are developed as follows. For early containment failure, PWR5 is RC1. For late containment failure, PWR6, PWR7, and PWR8 are grouped as RC2. For isolation failure, PWR3 and PWR4 are grouped as RC3. PWR1 and PWR2 are grouped as RC4 because they both represent bypass failure. In addition, for the containment intact scenario, PWR9 is represented by RC5. The table below provides a general overview of the release category mapping scheme.

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 are used to describe 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.

In order to apply 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 are provided. The rest inventory fractions and occurrence frequency are only available 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 in this research. 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	1.94E-07	9.20E-01	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	1.17E-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 previous proposed method, the obtained fraction of radioactive release group of TMI Unit 1 Level 2 PSA can be used to calculate the fractions for new release categories. The calculated results are presented in the table below and the detailed calculation steps 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 above table, it can be found the noble gas is released immediately after the core melt accident. By comparing late containment failure and early containment failure scenarios, it can be seen that the release factions of Caesium and Tellurium decrease a lot along with the time progression. Moreover, among five release categories, containment bypass failure can lead to the largest amount of radionuclide release regarding the release magnitude. But containment intact scenario has the least amount of radionuclide release among others. It is also interesting to find out that the fraction of iodine does not vary a lot under different failure scenarios, which verified the previous analysis that organic iodine cannot be filtered by the current filter systems. Therefore, regardless of the time span, the release amount of iodine will not vary a lot.

3.6.2 Release Frequency Quantification

In addition to the identification of source term fractions under different release categories, it is also important to analyze the occurrence frequency of each release category so that the accident consequences for different release paths can be analyzed. The calculated consequences can be further used to investigate the vulnerability part of the containment safety systems so that accident mitigation management can be continuously optimized. To calculate the occurrence frequency for each release category, the probability of basic CET events needs to be calculated first. Hereby, component failure rate and event occurrence frequency are calibrated based on the data collected from the literature review, the detailed frequency calibration method can be viewed in Appendix D. The table below describes the occurrence frequency of containment end states in the proposed 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 beginning
CET21	1.83E-06	ISLOCA
CET22	2.47E-06	SGTR

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

3.6.3 Consequence Result 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. Within the same category, by adding up corresponding containment end sate frequency together, the occurrence frequency of 5 release categories 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. 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, compared with early containment failure, late containment failure is much more likely to occur. In order to have better insight regarding the frequency of release categories, the release category frequencies from studies such as WASH-1400, French 1300 MWe level 2 PSA, and TMI Unit 1 level 2 PSA are taken to compare with those obtained from the 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 14 summarized the descriptions of these 9 release categories.

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 the 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 the release from gaps, containment is intact

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

Table 15 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 in order to provide a general overview of release scenarios and their consequences.

Release Category	Probabili-ty 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	0.7	0.4	0.4	0.05	0.4
WH 2	8×10^{-6}	170	0.9	0.7	0.5	0.3	0.06	0.02
WH 3	4×10^{-6}	6	0.8	0.2	0.2	0.3	0.02	0.03
WH 4	5×10^{-7}	1	0.6	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}
WH 5	7×10^{-7}	0.3	0.3	0.03	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 the generic model 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 a 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 show that 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 are able to represent the results obtained from the study which used large event trees for analysis.

In addition to WASH-1400, the results from level 2 PSA of French 1300 MWe PWR were also taken to cross-compare with the research results. 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 comply with each other very 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, SGTR, IS 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 serve 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 in advance. These information together at least could provide a preliminary view on risk significant events of the target plants.

3.7 Major Accident Mitigation Management

As mentioned in the previous section, the release consequence results obtained from the generic model of level 2 PSA can be used to investigate the vulnerabilities of the nuclear power plant. If those plant vulnerabilities can be improved, the existing risk mitigation measures can be further optimized. In addition, because the major containment failure mode can be identified through the generic model, the mitigation and emergency plan can be developed so that different accident scenarios can be dealt with. According to the level 2 generic model, it can be seen that the most likely scenario is that the containment still remains intact. Although there are still some radioactive releases through normal penetration, the release impact on the environment and on public health is very small. Under this scenario, accident mitigation measures are very effective so that potential major accident development can be well controlled. Mitigation measures are to be applied after a core damage accident in order to prevent containment failure. By controlling radioactive substance transportation, the ultimate goal of mitigation measures is to reduce offsite consequences as much as possible.

According to the defense-in-depth strategy, there are several barriers to mitigate major accident development after core damage. The recovery of emergency core cooling systems (ECCS) serves as the first barrier to stop the core melt development and progress within the reactor, which includes a high-pressure safety injection system and a low-pressure safety injection system. If the core melt can be stopped before the reactor rupture, the impact is the least severe among other scenarios regarding the radioactive release amount. By using recovered ECCS to inject the makeup water into the reactor coolant system or the reactor, it can help to cool down the reactor and stop the core melt progress. Therefore, the reliability and effectiveness of the ECCS recovery are critical for risk mitigation during the in-vessel phase. In addition to using the recovered ECCS, another method to prevent the reactor rupture after the core melt is to flood the cavity. By submerging the reactor vessel in the water, the decay heat from the corium pool can be transferred through the vessel wall into the cavity. In order to successfully cool down the vessel by flooding the cavity, it needs the abundant coolant inventories and time window to fill the coolant into the cavity. Compared with the ECCS mitigation approach, it may also cost a huge amount of money for repairing the plant after the accident. Therefore, the recovery of ECCS should be given high priority in terms of risk mitigation after a core melt.

However, sometimes ECCS recoveries are not available after a core melt and it may be too costly by flooding the cavity. The core melt may continue and lead to the vessel breach. As depicted in the containment event tree model, reactor rupture scenarios can be classified into two categories: the reactor rupture under high pressure and the reactor rupture under low pressure. The reactor rupture under high pressure is the scenario that the vessel ruptures when the primary system pressure still remains high. Depending on the amount of released energy and availability of the containment safety systems, the reactor vessel rupture under high-pressure can lead to different containment failure scenarios, which include alpha mode failure, early containment failure, and late containment failure. Among these three types of containment failures, the alpha mode failure is perceived as the most severe containment failure scenario because it can lead to the missile effect. Under this scenario, the released energy can easily exceed the design boundary of the containment, which will cause the direct radioactive release to the environment. Because this failure mode will bypass all the containment safety systems, the only effective mitigation measure is to depressurize the reactor vessel before it ruptures. The depressurization strategy can be achieved through the means of opening pressurizer relief valves or using external depressurization stations. If all these depressurization measures are not available to prevent the accident, the emergency plan for public relocation should be timely activated. Speaking of the emergency plan preparation, social acceptance and ethical acceptability issues need to be well considered. Compared with the alpha mode failure, the early containment failure is relatively less severe because the containment safety systems can still help to mitigate the accident consequence. For example, if the containment spray system functions well, the accident consequence caused by direct containment heating can be well mitigated. In addition, if the hydrogen control system works during the accident, the hydrogen combustion induced containment failure can also be avoided. The scale of hydrogen combustion is highly dependent on the containment geometrical conditions and hydrogen concentration. Therefore, the optimal mitigation solution to reduce the hydrogen risk can be the combination use of passive autocatalytic recombiners and igniters. By implementing this strategy, the local combustion can be avoided through the deliberate ignition and the hydrogen concentration can be reduced through the catalytic recombiners. It can be seen that the availability of containment safety systems plays an important role in major accident mitigation.

Even though the accident can be mitigated by the safety systems at the early stage of the accident, there are still possibilities that will lead to the late containment failure because of the ex-vessel accident progression. In order to prevent “late containment failure”, the containment safety systems should be available during the whole accident progression period. For example, the hydrogen recombiner or ignition system should be always available in case of the late hydrogen burns. As described in the containment event tree, the typical example of “late containment failure” is the slow over-pressurization. Due to the oxidation of the molten corium and its interaction with concrete, the pressure in the containment can gradually build up. To prevent slow over-pressurization induced containment failure, the recirculation mode of the spray system must be guaranteed so that it can cool down the corium and the containment. In addition, the filtered venting should also be well maintained because it can relieve the containment pressure to the outside environment. Meanwhile, it can help to reduce the ex-vessel source term release. However, filtered venting should not be heavily relied on to mitigate the slow over-pressurization induced containment failure. To force venting the containment atmosphere, it requires a very large capacity of the filters. Also, it may cause some direct radioactive release through the venting system in case that radioactive filter does not function well. Therefore, the recirculation of spray systems should be set as the first priority to mitigate the over-pressurization issue in the accident mitigation plan. Different from the reactor rupture under high pressure, the reactor rupture with the low-pressure scenario is less severe because the primary system pressure was already very low before the reactor ruptures. The only concern for this scenario is whether the accident progression can lead to containment failure or not. If the reliability of the safety systems can be improved or more engineered safety barriers can be implemented, it is very unlikely to result in the large containment failure under low-pressure reactor rupture scenario.

Although the reactor rupture induced containment failures are major concerns for containment safety management, there are still other types of containment failures that may challenge the containment integrity. According to the results obtained from the generic level 2 PSA model, it can be seen the containment bypass failure has the highest radioactive release magnitude among all other types of containment failures. Its occurrence frequency is also relatively high compared with others. Therefore, the mitigation management plan should also take this scenario into consideration. ISLOCA and SGTR are two common types of bypass failure. The redundant design of isolation valves in the interface between the reactor coolant system and the low-pressure piping may be an effective way to mitigate the ISLOCA accident. To mitigate the SGTR accident consequence, radioactive filters are needed to filter the coolant in the secondary system so that even if the secondary system would be contaminated during the accident, it can still get quickly purified before it releases to the environment. By continuously improving the current safety design, the accident consequences are expected to be mitigated as much as possible to an acceptable level.

The above mitigation strategies are all based on the concept that the safety systems should have the least human intervention. Because once human interaction gets involved in normal operation, human error may lead to potential accidents. For example, if the maintenance operators forget to close the isolation valve after the normal check, it may cause the containment isolation failure. In this case, all safety designs are ineffective because of human errors. Therefore, even though the system needs to be designed to require as little human intervention as possible, the plant operators still need to get proper training regarding normal operation and emergency management. If the required operation training has been well performed, human error induced accidents can be reduced. Even if the accident occurred, the accident consequence also would be mitigated because operators would have been trained to deal with different emergency situations.

The aim of the improvement of safety systems and the enhancement of emergency training performance is to limit the accident impact within the nuclear facility. However, sometimes the safety systems and the onsite emergency plan may fail to stop the accident progression. Therefore, in order to develop a comprehensive accident mitigation plan, the worst-case scenario still needs to be considered, which is the case that all safety systems and emergency operation training are in-effective and the radioactive substances get released to the environment. Hereby, the offsite emergency plan is needed, which includes the relocation of the communities and further treatment of the contaminated areas, etc. The offsite mitigation strategy should well consider social and ethical issues because the accident consequences are closely related to people's everyday life. For the offsite accident mitigation plan, the social acceptance and ethical acceptability need to be discussed in order to prevent potential social conflicts that might be generated due to the mitigation management.

Because the offsite mitigation management needs the wide support from the public and the society, the public acceptance towards the mitigation plan should be well taken care of. Therefore, in order to enhance the public acceptance and minimize the accident consequence, the first priority for accident mitigation management is to establish a good communication mechanism with the public, which means all the information related to nuclear risks should be available to the affected parties. A good risk communication mechanism is needed during the whole life cycle of risk management, which requires a proper consideration of the societal and ethical issues in the initial development.

Risk communication is the risk information exchange process among all interested parties regarding the nature of the risk, the risk significance, and control of the risk (Covello, 1992). A good risk communication mechanism should function as a process to enable the public and affected parties to understand and internalize the risk message (Sato, 2016). It should also serve as the tool to facilitate the consensus establishment in order to manage the potential risks in a collaborative way. Moreover, risk communication should focus on both the post-accident phase and the pre-accident phase. For example, if the public has been well informed about the risk exposure and potential accident consequences, it will be much easier to mobilize the affected public to implement the offsite mitigation plans because they are already aware of the accident consequence. However, sometimes the government just overlooked the importance of timely communication about potential risks during the plant operation, which makes the mitigation plans less effective as it was designed for.

In the Fukushima accident investigation report, Tokyo Electric Power Company (TEPCO) admitted misconduct by falsifying data to cover up nuclear risks for nearly 200 periodic safety checks on the three TEPCO nuclear power plants (including Fukushima Daiichi) in the years between 1977 and 2007 (Xiang & Zhu, 2011). For these identified safety issues, they did not take actions to timely communicate with affected parties and correct them. Instead, they chose to cover up these risks so that they could avoid potential disruptions of the operation. If they could obey the safety regulations and inform the affected parties about the risks, the Fukushima Daichi accident probably can be avoided or at least the accident consequence can be reduced. Because the company and the government did not release the full fact to affected parties, it makes the public gradually lose their trust in the government's risk mitigation capability, which would further reduce the implementation effectiveness of mitigation plans. Trust is an important factor that influences the social acceptance of mitigation management. In practice, it is not possible for all individuals to acquire a similar level of knowledge about nuclear risks, many people relied on the statement or opinions made by nuclear technology experts or authorities to decide their acceptance level (Liu et al., 2008). This willingness to follow the authorities' opinions are perceived as trust. Normally It takes a long time to build trust, but it is very easy to pull it down. The Fukushima accident mitigation issues are good examples to illustrate how easily trust can be torn down due to the inaccurate information release (Roh, 2017). After the accident, Japanese nuclear authorities are heavily criticized by the public, which results in the big distrust from the public to the government. If the public cannot reach a wide consensus regarding the accident mitigation plan, the opposition from the society may cause the potential obstacles for the smooth implementation. For example, the resident evacuation activities did not go very well due to poor risk communication. The radius of evacuation zones has been changed three times, which made the affected public have to relocate for more than six times (Hasegawa, 2016). These problems all reflected how public trust can influence mitigation effectiveness. The misleading and contradictory risk information not only reduced public trust but also could intensify the public panic towards accident consequences. Therefore, the ethical way to deal with these issues is to objectively inform the public about the risks with accurate information in a timely manner. A good communication mechanism and proper training on emergency preparation can help to improve the evacuation efficiency and smooth the public panic during the accident mitigation management.

In order to set up a good risk communication mechanism, several suggestions have been made accordingly. First of all, all the risks related to nuclear plant operation should be properly addressed and explained to the public. Corresponding risk mitigation and emergency plans should be developed together with the affected communities. For example, once the radioactive substances get released to the surrounding neighborhoods, the relocation of the affected communities may be needed. Under this circumstance, the affected communities should be well taken care of, especially their emotions towards the relocation plan as well as their potential economic loss. In addition, during the collaborative decision-making process, all affected parties should be treated fairly without any discrimination on their identities, beliefs or social status so that the potential conflicts can be reduced. Secondly, in case of major accidents, a timely emergency communication platform should be set up so that the affected public can have a knowledge of the risk magnitude and prepare themselves for the evacuation plans. This communication platform should be designed and made before the plant operation by taking all affected parties' opinions into account. Furthermore, post-accident communication should be ensured so that concerned parties can have access to the accident information, which could help to reduce the public panic regarding accident impact. Last but not least, all uncertainties existed in the risk analysis should be discussed in a collaborative manner and limits of current knowledge should be properly addressed. A transparent decision-making process can improve the credibility of the authorities and make the strategy implementation more smoothly.

Speaking of the implementation of mitigation strategies, there are several protective measures that have been recommended in order to reduce the potential radiological impacts on the public. In the event of a major nuclear accident, the radioactive release consequence varies a lot depending on accident characteristics and weather conditions. These factors jointly determine the size of ground contamination, the potential exposure imposed on people who live around nuclear plants, and the areas that will be affected by protective actions (Qu, 2003). Among different protective actions, relocation of the affected communities is perceived as one of the effective ways to minimize the radiological exposure after the accident, which has been applied in both Chernobyl and Fukushima accident mitigation management (Ashley, et.al., 2017). However, it also brings about a lot of social and ethical issues, which also need to be carefully considered. According to UNHCR (2014), relocation activity can be classified into evacuation and planned relocation. Evacuation refers to the rapid movement of the affected parties to a safer place in case of imminent risk. Under this scenario, there are several issues that need to be properly addressed. For example, questions like whether the evacuation plan is made under the free and informed consent of the affected communities are major concerns. Different communities may hold different values towards the evacuation plan and they also have different capabilities to implement the plan. Therefore, their interests and concerns should be well discussed during the collaborative decision-making process so that the burden and benefits can be distributed fairly. In addition, during the evacuation strategy development, decision-makers tend to overlook the protracted displacement issues after the evacuation, especially the place of the origin is not suitable to live anymore (UNHCR, 2014). If these issues cannot be well managed, it may lead to a deep reflection on whether the development of the nuclear plant is the morally right thing or not for the local communities because it seems the local communities' interest is sacrificed a lot for the sake of national interest. It will also bring the question of whether the benefits and burdens are distributed fairly in the spatial dimension. Therefore, these potential social and ethical issues should be well discussed during the evacuation plan development.

Different from the emergency evacuation approach, the planned relocation refers to the situation that the affected communities are relocated and integrated to another place, which is normally planned before the accident. Considering this relocation activity may bring about a lot of social and ethical problems, it should be used as the last resort. According to the Chernobyl investigation report published by WHO (2006), it was found relocation has brought deeply traumatic experience to many people. Because of the disruption of social networks, it is very difficult for ‘exposed people’ to return back to their previous life and many of them also suffered a social stigma associated with being ‘exposed people’. This psycho-social issue is also found in the relocation activity after the Fukushima Daiichi accident. It was shown that relocation has caused a fall in the notional life quality of the residents from contaminated areas, as measured by the life quality index (Waddington, et. al., 2017). Therefore, during the development of the relocation plans, it is very important to ensure the culture and identity of the relocated communities can be easily integrated to the host communities so that the recognition justice can be guaranteed. If two communities hold a big divergence regarding interests and beliefs, it will harm both sides and later on result in a lot of social problems as mentioned above. However, the selection of the host community for resettled communities can also provoke other justice issues. For example, the benefits generated from the plant operation are distributed at a national level, while the burdens are only taken by certain communities. Therefore, it is necessary to reach a national consensus regarding how to compensate these affected parties so that benefits and burdens can be fairly distributed. Besides justice issues, public emotion is another concern that influences the relocation process. Both resettled communities and host communities should be appeased so that the accident induced disruption and dislocation impact can be reduced as low as possible. In this way, it will be less difficult to implement the corresponding relocation plan. Even though the relocation of affected communities can significantly reduce the inhaled dose of radioactive substances, decision-makers still need to bear in mind that the social and ethical issues induced by this measure is huge. In order to well manage these issues, a participative decision-making approach can be an effective way to enable all affected parties to express their concerns and exchange ideas. In the end, a consensus regarding how to develop a comprehensive relocation plan can be reached so that the accident induced health impact can be well mitigated.

In addition to the health impact, the accident induced environmental issue is another big concern for risk mitigation management. Different from the property loss or operation disruption, the damage to the environment and the surrounded wildlife is uncountable. The contaminated area may need hundreds of years to recover, which will bring the extra burdens to future generations. Therefore, the accident mitigation plan also needs to think about intergenerational justice issues. Once the accident happened, the contaminated area should be controlled as small as possible so that the accident induced environmental issues can be reduced. Also, the treatment plan towards the contaminated areas needs to be prepared in advance. For example, part of turnovers from the nuclear plant operation should be continuously used to invest the basic research and technology which can tackle the nuclear accident induced environment issues. Or the special funds should be set aside in order to tackle the contamination problem. By using these ways, it may help to reduce the burdens that are imposed on future generations due to environment recovery issues. However, in addition to the intergenerational justice issues, recognition justice issues also need to be considered. For example, the future generation may possess different values and beliefs towards contamination treatment plans and they also have their own rights to decide whether or not continue developing nuclear technology. Therefore, the ultimate goal for the accident management should be to leave as few burdens to future generations as possible.

From the above discussion, it can be seen the accident mitigation management should take both technical risk and societal risk into consideration. The generic level 2 PSA model can be used to develop mitigation plans to tackle different nuclear accidents from the engineering perspective. Based on the release frequency and release amount obtained from the model, different mitigation approaches can be generated for different release scenarios. The generic level 2 PSA model is mainly used to analyze how to improve the performance of containment safety barriers in order to stop the core melt accident as soon as possible. However, the level 2 PSA model cannot deal with the accident induced societal and ethical issues, which are mainly brought by the radioactive release. Therefore, societal acceptance and ethical justice issues are discussed to complement the mitigation strategy. In order to well integrate both technical risk and societal and ethical discussion into risk mitigation management, a new nuclear risk governance framework is proposed based on the IRGC risk governance principles (IRGC, 2017).

For nuclear power development, the risk is embedded in the situation with high complexity and uncertainty. Therefore, a multidisciplinary and multi-stakeholder involved approach should be used. The newly proposed risk governance framework includes four steps, which are risk identification, risk assessment, risk solution evaluation, and implementation. During each step, the communication among associated parties and stakeholder engagement are actively involved. In risk identification, the associated risks are identified and the risk governance boundaries will be defined. In the risk assessment step, the technical risk is assessed based on

PSA results and related societal and ethical issues are discussed through different indicators, which may include risk perceptions from different stakeholders, justice concerns and socio-economic impact of the mitigation measures. In risk solution evaluation step, risk acceptance and ethical acceptability of the affected parties are evaluated. The feasibility and practicality of the mitigation strategy are examined. In the last step, the optimal risk management solutions are developed and continuous improvement is made based on practice feedback. The newly proposed risk governance framework can be viewed in the graph below. By applying this risk governance framework, it may better facilitate the decision-making process regarding nuclear risk management so that mitigation strategies can be continuously optimized.

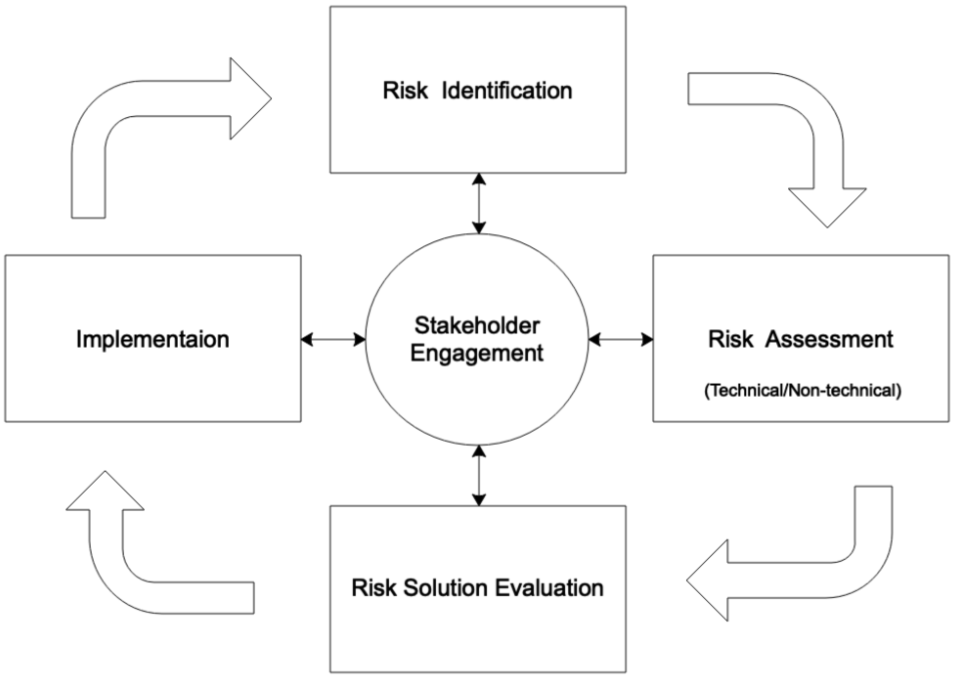

Figure 9. The proposed nuclear risk governance framework

Chapter 4 Conclusion & Reflections

This chapter is designed to conclude the final research findings based on the reflection of the whole research development. In the research, a generic level 2 PSA model has been developed, which was used as the reference to analyze different accident scenarios. Based on the analysis results obtained from the model, the corresponding risk mitigation measures have been discussed. By properly taking the social and ethical issues into consideration, a comprehensive nuclear risk mitigation plan has been developed so that nuclear risk management can be further improved.

One of the good ways to reflect the research development is to revisit research questions. At the beginning of the research, three sub-questions have been proposed in order to better guide the research development. Therefore, it is imperative to revisit them to see whether the developed model and mitigation plan can handle these questions and what answers can be provided based on the research results. By revisiting these questions, it can also help to check whether the research outcomes meet the research goals. After the reflection on research questions, another section has also been designed in this chapter aiming at providing the summarized creative points of the research, which can point 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 are also discussed at the end of this chapter.

4.1 Revisiting the Research Questions

Sub-question 1: *What steps should be followed in order to create a generic level 2 PSA model?*

This question is designed to come up with the research outline for the development of generic level 2 PSA model. In order to build up the generic model, several things need to be considered. 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. Through the broad literature review, the studies related to level 2 PSA have been discussed and the research motivation to create the generic model has been analyzed. The second thing is to select the reference reactor type to perform the analysis. Because level 2 PSA can be tailored and applied for various types of nuclear plants, a reference reactor type is needed as the research basis. Through the qualitative study and literature review, the pressurized water reactor has been selected as the reference reactor. The common design types of containment safety systems have also been summarized. Based on the above information, 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. By creating generic event trees, the major release paths have been identified and the release frequency and its release consequence have been quantified.

Sub-question 2: *How can the generic level 2 PSA model contribute to optimizing risk mitigation within a NPP?*

The generic level 2 PSA model is designed to analyze the severe accident progress and containment response. By creating the generic containment event tree, it can represent various nuclear accident scenarios as well as estimate their accident consequences without need to check specific design details. Based on the release category difference, the different accident progress scenarios can be figured out. Because different release scenarios have different radioactive release amount, the severity of the accidents could be identified, which can be further used to optimize accident mitigation management.

Based on the analysis results obtained from the model, the accident mitigation plan has been developed in the research. According to the defense-in-depth concept, different mitigation measures towards different accident scenarios have also been proposed. The ultimate goal for risk mitigation management is to reduce the accident consequence as low as possible. Therefore, by using the generic model to analyze accident progress characteristics, the optimal risk mitigation solution can be developed.

Sub-question 3: *How should societal issues be addressed in order to complement the probabilistic safety analysis?*

Probabilistic safety analysis is used to analyze the nuclear risks from the technical perspective. By investigating the accident progress and release consequence, nuclear plant safety can be assessed. Based on the analysis results, the corresponding risk mitigation measures can be proposed. However, due to the limitation of the probabilistic safety analysis, those proposed risk mitigation measures mainly focus on the technical and operational side, which neglect the societal aspects. Therefore, it is important to include societal and ethical discussions regarding risk mitigation management so that nuclear risk can be well addressed from different aspects. In the research, the social acceptance and ethical acceptability regarding risk mitigation management have been selected to evaluate the practicality of mitigation management. By taking the risk communication and community relocation mitigation measures as an example, risk perception, public emotions, and trust issues, etc. are chosen as indicators to analyze the societal challenges regarding the risk mitigation management. The corresponding solutions to tackle these issues have also been recommended. Moreover, by discussing the distributional justice, procedure justice, and recognition justice issues towards the risk mitigation management, the moral acceptability regarding mitigation measures has been addressed and countermeasures to deal with these justice issues are also recommended. In this way, both technical aspects and non-technical aspects of nuclear risks can be analyzed so that risk mitigation management can be comprehensively improved. By using the proposed risk governance framework, it may better facilitate the decision-making process regarding nuclear risk management.

Based on detailed answers provided for the above three sub-questions, it is clear that nuclear risk mitigation management should be developed by taking both technical and social aspects into consideration. The generic level 2 PSA model can provide a good reference regarding how to improve and design the engineered accident mitigation measures. The societal and ethical discussion can make the mitigation strategy more comprehensive and applicable to those societal issues.

4.2 Creative Points in the Research

Through the broad literature review and detailed qualitative analysis, it was found current level 2 PSA studies are mostly performed for the specific design type of plants. The results obtained from these specific analyses may be less valuable for other plants to take as the reference. Too many details make it hard for non-experts to study and understand. Therefore, the first creative point in this research is the development of a generic level 2 PSA. By creating the generic level 2 PSA model, the whole accident progression scenarios, as well as key plant damage states, have been taken into consideration. By identifying potential release paths and estimating corresponding consequences, the analysis results could be used as a good reference to develop the accident mitigation management plan.

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 more generic, various PWRs in different countries have been studied. The classification tables can serve as good references for future containment safety system studies. They together provide 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 a generic containment event tree. As mentioned in previous chapters, the containment event tree is one of the key elements in level 2 PSA. 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. For example, if there were 40 plant damage states, there would be 40 containment event trees, which would lead to a huge number of event tree end sates. It will be very difficult for others to quickly understand release paths in the plant by going through 40 event trees. Later on, although some improvements have been made to reduce the size of containment event trees, they gradually became very detailed and plant-specific, which makes 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. In this research, a generic containment event tree has been 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. In this research, by just using one generic containment event tree, different release scenarios have been represented, which greatly reduces the complexity and redundancy among end-state events.

The last but not least creative point is the development of the mitigation management plan by properly taking societal and ethical issues into account. Based on the results obtained from the generic model, most of the mitigation measures are designed to improve the current safety systems or implement more safety barriers to control the accident progress. However, these measures are ineffective to tackle with societal and ethical problems, which also need to be well considered in order to mitigate the accident consequence. Therefore, by discussing the mitigation measures from societal and ethical perspectives, it makes mitigation management more comprehensive and effective. By proposing a new risk governance framework, it may better help the development of nuclear risk mitigation management.

4.3 Limitations and Future Work

Although the PWR generic level 2 PSA model has been successfully developed and the accident mitigation plan has been discussed from both technical and societal and ethical perspectives, there are still some limitations in the research. In the plant damage state analysis and containment event tree development, all accidents are assumed to occur at the full power operation, which have neglected the off-power situation. For the containment safety system analysis, all safety systems are assumed independent from each other, therefore, the data accuracy issues regarding the reliability of these systems need to be improved in the future. Also, because it is very difficult to cover all design types of nuclear plants, in this study, only pressurized water reactors are selected as the research base. In addition, during the establishment of the plant damage state logic diagram, only key attributes for the accident progression have been taken into consideration. Besides these model development issues, there are also some limitations in the data quantification process. 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 event frequency, and system failure rate are also simplified by just taking the mean value obtained from the literature as the data input. 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 considered. Therefore, if a better data source can be provided, it can better improve the release consequence estimation. In addition, although the accident mitigation measures have been discussed from social and ethical aspects, societal issues are still very difficult to deal with. All those proposed measures are mainly aimed to reduce the potential accident consequences in a more socially and ethically acceptable way. Last but not least, the economic aspect regarding the risk mitigation measures is not fully addressed because the appropriateness to use cost benefit analysis to value the environmental and social impact is questionable.

As mentioned above, there are several limitations in the research, therefore, some future works are expected to 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 accordingly. If these reactor types 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 safety evaluation.

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 robust. 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. The third aspect of future work development is to continue improving the accident mitigation management strategies by including the economic feasibility discussions. Based on the improved model results, more robust mitigation measures can be brought up. Also, for societal issues discussion, more open debates regarding how to develop mitigation measures in a socially and ethically acceptable way should be encouraged so that the proposed accident mitigation strategies could be more comprehensive and applicable in practice.

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Appendix A: Descriptions of Release Subevents 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 release of fission products, with fission product scrubbing
1-02	PWR1	Containment bypass, outside the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing
2-01	PWR2	Containment bypass, to the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing
2-02	PWR2	Containment bypass, to the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing
2-03	PWR2	Containment bypass, to the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing
2-04	PWR2	Containment bypass, to the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing
3-01	PWR3	Large isolation failure, to the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing
3-02	PWR3	Large isolation failure, to the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing
3-03	PWR3	Large isolation failure, to the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing
3-04	PWR3	Large isolation failure, to the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing
3-05	PWR3	Large isolation failure, outside the auxiliary building, without ex-vessel release of fission products
3-06	PWR3	Large isolation failure, outside the auxiliary building, with ex-vessel release of fission products
4-01	PWR4	Small isolation failure, to the auxiliary building, without ex-vessel release of fission products, with fission product scrubbing
4-02	PWR4	Small isolation failure, to the auxiliary building, without ex-vessel release of fission products, without fission product scrubbing
4-03	PWR4	Small isolation failure, to the auxiliary building, with ex-vessel release of fission products, with fission product scrubbing
4-04	PWR4	Small isolation failure, to the auxiliary building, with ex-vessel release of fission products, without fission product scrubbing
4-05	PWR4	Small isolation failure, to the environment, without ex-vessel release of fission products, with fission product scrubbing
4-06	PWR4	Small isolation failure, to the environment, without ex-vessel release of fission products, without fission product scrubbing
4-07	PWR4	Small isolation failure, to the environment, with ex-vessel release of fission products, without fission product scrubbing
4-08	PWR4	Small isolation failure, to the environment, with ex-vessel release of fission products, without fission product scrubbing
5-01	PWR5	Early containment failure, without ex-vessel fission product release
5-02	PWR5	Early containment failure, with ex-vessel fission product 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 catastrophic containment failure, without ex-vessel fission product release, without re-vaporization, with fission product scrubbing
6-02	PWR6	Late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, without re-vaporization, without fission product scrubbing
6-03	PWR6	Late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, with re-vaporization, with fission product scrubbing
6-04	PWR6	Late overpressurization, with catastrophic containment failure, without ex-vessel fission product release, with re-vaporization, without fission product scrubbing
6-05	PWR6	Late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, without re-vaporization, with fission product scrubbing
6-06	PWR6	Late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, without re-vaporization, without fission product scrubbing
6-07	PWR6	Late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, with re-vaporization, with fission product scrubbing
6-08	PWR6	Late overpressurization, with catastrophic containment failure, with ex-vessel release of fission products, with re-vaporization, without fission product scrubbing
7-01	PWR7	Late overpressurization, with benign containment failure, without ex-vessel fission product release, with fission product scrubbing
7-02	PWR7	Late overpressurization, with benign containment failure, without ex-vessel fission product release, without fission product scrubbing
7-03	PWR7	Late overpressurization, with benign containment failure, with ex-vessel release of fission products, with fission product scrubbing
7-04	PWR7	Late overpressurization, with benign containment failure, with ex-vessel release of fission products, without fission product scrubbing
8-01	PWR8	Containment failure from basemat melt-through, with ex-vessel release of fission products
9-01	PWR9	No containment failure, without ex-vessel fission product release, with fission product scrubbing
9-02	PWR9	No containment failure, without ex-vessel fission product release, without fission product scrubbing
9-03	PWR9	No containment failure, with ex-vessel fission product release, with fission product scrubbing
9-04	PWR9	No containment failure, with ex-vessel fission product release, without fission product scrubbing

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 are taken from TMI Unit 1 level 2 PSA report (USNRC, 2007).

Sub-release event sequences	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
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

Sub-release event sequences	Release category	Release fractions				
		Iodine (I)	Caesium (Cs)	Tellurium (Te)	Strontium (Sr)	Ruthenium (Ru)
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
6-07	PWR6	2.0E-02	2.0E-02	4.0E-03	1.0E-06	4.0E-05

Sub-release event sequences	Release category	Release fractions				
		Iodine (I)	Caesium (Cs)	Tellurium (Te)	Strontium (Sr)	Ruthenium (Ru)
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

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

$$4.57E\text{-}07+1.59E\text{-}06 = 2.05E\text{-}06$$

PWR2

$$1.81E\text{-}07+0+1.27E\text{-}08 = 1.937E\text{-}07$$

PWR3

$$9.07E\text{-}11+ 9.07E\text{-}11+1.90E\text{-}10+2.88E\text{-}10= 6.59E\text{-}10$$

PWR4

$$3.90E\text{-}08+1.46E\text{-}08+8.54E\text{-}09+3.16E\text{-}07= 3.78E\text{-}07$$

PWR5

$$7.39E\text{-}07+1.66E\text{-}07 = 9.05E\text{-}07$$

PWR6

$$2.20E\text{-}08+2.36E\text{-}10+2.08E\text{-}11+8.00E\text{-}08+1.43E\text{-}08 = 1.16557E\text{-}07$$

PWR7

$$2.25E\text{-}07+2.75E\text{-}09+7.45E\text{-}07+2.89E\text{-}07 = 1.26E\text{-}06$$

PWR8

$$3.19E\text{-}06$$

PWR9

$$1.20E\text{-}05+1.69E\text{-}08+2.3E\text{-}06 +1.91E\text{-}08= 1.44E\text{-}05$$

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 account, 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 are 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)

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

Strontium (Sr)

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

Ruthenium (Ru)

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

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)

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

Caesium (Cs)

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

Tellurium (Te)

$$\frac{1.2\times10^{-5}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.0\times10^{-9}+\frac{1.69\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 1.0\times 10^{-6}+\frac{2.36\times10^{-6}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.8\times10^{-6}+\frac{1.91\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 1.4\times10^{-5}\ =4.84\text{E-}07$$

Strontium (Sr)

$$\frac{1.2\times10^{-5}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 4.0\times10^{-9}+\frac{1.69\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.5\times 10^{-7}+\frac{2.36\times10^{-6}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 5.0\times10^{-8}+\frac{1.91\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.5\times10^{-7}\ =1.23\text{E-}08$$

Ruthenium (Ru)

$$\frac{1.2\times10^{-5}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 2.0\times10^{-9}+\frac{1.69\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 1.0\times 10^{-6}+\frac{2.36\times10^{-6}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 1.4\times10^{-6}+\frac{1.91\times10^{-8}}{1.2\times10^{-5}+1.69\times10^{-8}+2.36\times10^{-6}+1.91\times10^{-8}}\times 7.0\times10^{-6}\ =2.43\text{E-}07$$

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

Release fraction data calibration for new proposed release categories is 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 proposed release categories.

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)

$$\frac{1.17\times10^{-7}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times 10^{-1}+\frac{1.26\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times 10^{-1}+\frac{3.19\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times 10^{-1}=$$

5.62E-07

Ruthenium (Ru)

$$\frac{1.17\times10^{-7}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times 10^{-1}+\frac{1.26\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times 10^{-1}+\frac{3.19\times10^{-6}}{1.17\times10^{-7}+1.26\times10^{-6}+3.19\times10^{-6}}\times 1.3\times 10^{-1}=$$

9.71E-06

RC3:

Noble Gas (Xe)

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

Iodine (I)

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

Caesium (Cs)

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

40

Tellurium (Te)

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

Strontium (Sr)

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

Ruthenium (Ru)

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

RC4:

Noble Gas (Xe)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 9.2\times 10^{-1} = 9.93\text{E-}01$$

Iodine (I)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1.09\times 10^{-2}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 8.5010^{-1}= 8.34\text{E-}02$$

Caesium (Cs)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1.09\times 10^{-2}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 8.5010^{-1}= 8.34\text{E-}02$$

Tellurium (Te)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1.64\times 10^{-3}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 1.71\times 10^{-1}=$$ 1.63E-02

Strontium (Sr)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 7.58\times 10^{-6}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 8.58\times 10^{-2}=$$ 7.41E-03

Ruthenium (Ru)

$$\frac{2.05\times10^{-6}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 3.96\times 10^{-5}+\frac{1.937\times10^{-7}}{2.05\times10^{-6}+1.937\times10^{-7}}\times 6.3\times 10^{-1}=$$ 5.44E-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 focus 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 from the NUREG-1150 report.

The below table summarizes 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

Table 22.The mean frequency of CDF under different PDS groups (NUREG-1150, USNRC, 1990)

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

(Because we only have core damage frequency induced by 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, and 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 (USNRC, 1990).

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
Taken from Reliability analysis for passive systems – A case study on a passive containment cooling system (Silvonen, 2011).

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]= 1.46E-07

CET3-1

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

CET4-1

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

CET5-1

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

CET6-1

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

CET7-1

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

CET8-1

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

CET9-1

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

CET10-1

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

CET 11-1

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

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 the reference, hereby a is given 0.1 in this research.

CET12-1

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

CET13-1

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

CET14-1

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

CET15-1

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

CET16-1

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

CET17-1

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

CET18-1

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

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 is 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 contribute 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

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 is given 0.01 in this research. (EPRI, (1981). A study of the risk due to accidents in nuclear power plants. Palo Alto,)

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

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

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

CET6-2

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

CET7-2

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

CET8-2

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

CET9-2

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

CET10-2

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

CET 11-2

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

CET12-2

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

CET13-2

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

CET14-2

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

CET15-2

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

CET16-2

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

CET17-2

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

CET18-2

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

CET19-2

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

CET20-2

For the isolation failure occurred before the core melt, data from TMI Unit 1 level 2 PSA ias 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 is 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

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 is given 1 in this research. (EPRI, (1981). A study of the risk due to accidents in nuclear power plants. Palo Alto,)

CET19	1.68E-07
CET20[8]	3.78E-07
CET21	1.83E-06
CET22	2.47E-06

Table 23. The calculated CET end-state release frequency

D.3 Release Category Frequency Calculation

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

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