Nuclear Engineering and Technology 50 (2018) 1234–1245

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Topical Issue Article

Multi-unit Level 2 probabilistic safety assessment: Approaches and their application to a six-unit nuclear power plant site

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A R T I C L E I N F O

Article history:
Received 20 September 2017
Received in revised form
21 March 2018
Accepted 3 April 2018
Available online 13 April 2018

Keywords:
Large Early Release Frequency
Level 2 Probabilistic Safety Assessment Method
Multi-unit Probabilistic Safety Assessment
Seismic Event
Site Containment Failure Probability
Site Risk
Tsunami Event

A B S T R A C T

The risk of multi-unit nuclear power plants (NPPs) at a site has received considerable critical attention recently. However, current probabilistic safety assessment (PSA) procedures and computer code do not support multi-unit PSA because the traditional PSA structure is mostly used for the quantification of single-unit NPP risk. In this study, the main purpose is to develop a multi-unit Level 2 PSA method and apply it to full-power operating six-unit OPR1000. Multi-unit Level 2 PSA method consists of three steps: (1) development of single-unit Level 2 PSA; (2) extracting the mapping data from plant damage state to source term category; and (3) combining multi-unit Level 1 PSA results and mapping fractions. By applying developed multi-unit Level 2 PSA method into six-unit OPR1000, site containment failure probabilities in case of loss of ultimate heat sink, loss of off-site power, tsunami, and seismic event were quantified.

1. Introduction

The risk of multi-unit nuclear power plants (NPPs) at a site, i.e., multi-unit risk, has received considerable critical attention recently, because approximately 70% of the plant sites worldwide have two or more NPPs [1]. Moreover, the Fukushima accident in 2011 raised concern regarding multi-unit risk. In South Korea, 23 NPPs are currently operating at just four plant sites: five NPPs at the Gori site and six NPPs each at the Wolsong site, Hanul site, and Hanbit site. Because of the large number of the NPPs at a site and the high population density near the sites, which are mostly due to the geographical characteristics of South Korea, the public acceptance of nuclear power has rapidly decreased in recent years. Accordingly, the regulatory body and government are demanding assessments of multi-unit risk.

Multi-unit risk was first considered for the Seabrook two-unit NPP after the Chernobyl accident [2]. Subsequently, multi-unit risk became an issue again in 2005 due to a new plant licensing policy in the United States [3]. Because the Fukushima accident has been the only multi-unit NPP accident at a site, research on multi-unit probabilistic safety assessments (PSAs) is rapidly progressing worldwide [4–13]. However, most studies in multi-unit PSA have only focused on the multi-unit Level 1 PSA scope, which is the quantification of the site core damage frequency (CDF).

In the multi-unit Level 1 PSA, one of the key research questions is how to calculate and get a high number of core damage accident sequences by considering every status combination of each NPP. When we assume that there are "n" NPPs at a site and that the number of accident sequences of core damage event trees (ETs) for a single-unit NPP is "x," the number of combinations of core damage accident sequences is theoretically $\sum_{k=1}^{n} x^k$, where x^k is the total number of combinations of core damage accident sequences, given that k NPPs have core damage. Accordingly, the next question arises: how many accident sequences we should consider for a multi-unit Level 2 PSA? It is obvious that the number of accident sequences of multi-unit Level 2 PSA increases as much as the number of severe accident sequences. By assuming there are six NPPs at a site, 100 core damage accident sequences, and 10 severe accident sequences, about 1.0E+18 accident sequences (or scenario) need to be considered in a multi-unit Level 2 PSA.

However, the current Level 2 PSA procedure and computer code do not support multi-unit Level 2 PSA because the traditional PSA structure is mostly used for the quantification of single-unit NPP risk. To solve this problem, the main purpose of this study is to

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https://doi.org/10.1016/j.net.2018.04.005

develop a multi-unit level 2 PSA method and apply it to real multi-unit NPPs in South Korea. For this application, full-power operating six-unit OPR1000 was considered. The site containment failure (CF) probabilities in case of two loss of ultimate heat sink (LOUHS), loss of off-site power (LOOP), tsunami, and seismic event were quantified.

The overall structure of this article has five sections, including this introductory section. Section 2 begins by describing a new method for a multi-unit Level 2 PSA. The third section presents the application results of the method to the six-unit OPR1000 at a site. In Section 4, the authors suggest the graphical plant damage distribution maps to better understand the results obtained by multi-unit Level 1 and Level 2 PSA. Finally, the conclusion gives a brief summary and critique of the findings.

2. Multi-unit Level 2 PSA method

Fig. 1 shows the overall structure for the calculation of the multi-unit Level 2 PSA. There are three steps for conducting the multi-unit Level 2 PSA. The first step is to develop a single-unit Level 2 PSA and calculate the CF probability for a single unit. In the second step, mapping fractions from the single-unit Level 2 PSA results are extracted. Finally, in the third step, multi-unit Level 2 PSA results are calculated based on multi-unit Level 1 PSA results, with the mapping fractions of single-unit Level 2 PSA results. Sections 2.1, 2.2, and 2.3 give detailed information for each step.

For the development of the multi-unit Level 2 PSA method, the major assumption is as follows:

- Inter-unit physical interaction is negligible during severe accident progression. This means that when one unit is experiencing severe accident progression, the other units are not physically affected by that.

The PSA requires technical steps (i.e., Level 1, 2, and 3). The statuses of technical interface with multi-unit Level 2 PSA are as follows:

- Multi-unit Level 1 PSA: For the multi-unit Level 2 PSA, the multi-unit Level 1 PSA results, such as the number of NPPs in which core damage has occurred, core damage accident sequences, and frequency of each sequence, are required. The model and quantification results of multi-unit Level 1 PSA were developed and described in the article written by Kim et al. [14].
- Multi-unit Level 3 PSA: For a multi-unit Level 3 PSA, multi-unit Level 2 PSA should provide CF probability and radioactive materials release fraction. However, the multi-unit Level 2 PSA method developed in this study only suggests the CF probability estimation of multi-unit NPPs. Because the source term calculation method for a single unit is already well established, multi-unit source term calculation could be quantified by adding source term of the each unit in which CF has occurred. The environmental effect caused by a multi-unit source term is the main aspect dealt with in the multi-unit Level 3 PSA. The multi-unit Level 3 method was developed and described in the article written by Kim et al. [15].
- Software Packages: For the multi-unit Level 2 PSA method, computer code calculation is needed due to a tremendous number of accident scenarios. In the article written by Han et al. [16], the software packages for multi-unit PSA are introduced.

2.1. Single-unit Level 2 PSA method

The single-unit Level 2 PSA has been, or is being, used for most NPPs worldwide. The main steps of the single-unit Level 2 PSA method are already well established. Thus, the general method which is commonly used [17] was adopted for the single-unit Level 2 PSA in this study.

There are four steps to quantify CF probability. First, the ET of a Level 1 PSA is extended to a plant damage state ET (PDS ET). Level 1 ET provides information of the accident sequences that lead to core damage. However, they do not include the status of the containment system that mitigates the effects of severe accidents. For example, a containment isolation system is not considered in a Level 1 PSA because it does not affect core integrity. However, it should be considered and designed in a Level 2 PSA because functionality of the containment system is one of the factors that determines the amount of radioactive material released and the timing of that release. Thus, accident sequences developed by Level

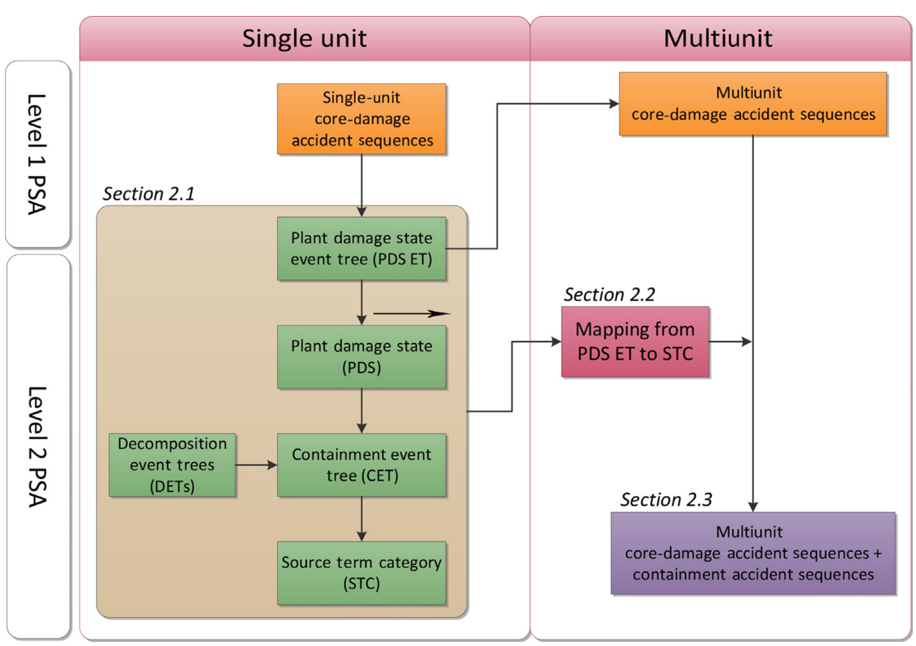

Fig. 1. Overall flowchart for multi-unit Level 2 PSA method.
PSA, probabilistic safety assessment.

1 PSA need to be extended to consider mitigation features of severe accidents.

Secondly, each accident sequence of PDS ET is grouped into specific plant damage state (PDS). Because the number of accident sequences of PDS ET is huge, they need to be grouped into PDSs based on similarities in the plant conditions that determine further accident progression. To do this, a PDS logic diagram and grouping rules are developed.

Third, a containment ET (CET) is developed, and the likelihood of every severe accident sequence is analyzed. To develop a CET, severe accident phenomena should be identified, and the probability of severe accident phenomena should be assessed by using expert judgment or on a reasonable basis such as computer code simulations and experimental data. To quantify the likelihood of every severe accident sequence, each PDS is used as the initial condition of the CET analysis.

Fourth, all end states of the CET are grouped into specific source term categories (STCs). Because the number of end states of the CET are too many to effectively quantify the amount and timing of radioactive material released to the environment, all end states of the CET need to be grouped into several STCs, based on similarities in the amount and timing of radioactive material released.

2.2. Mapping fractions from PDS ET to STC

The relationship between accident sequences of PDS ET and PDS is one-to-one function. This means that one accident sequence of PDS ET is classified into just one PDS. Because the number of accident sequences of PDS ET is much larger than the number of PDSs, the function is surjective, and several accident sequences of PDS ET can be classified into one PDS.

The relationship between PDS and CET is a discrete likelihood function. This means that one PDS is distributed into every CET having a likelihood. The summation of every likelihood originating from one PDS should be equal to one.

The relationship between CET and STC is also one-to-one function. This means that one accident sequence of a CET is classified into just one STC. Because the number of accident sequences of a CET is larger than the number of STCs, the function is surjective, and several accident sequences of a CET are classified into one STC.

The above three relationships are illustrated in Fig. 2, in which the number of accident sequences of PDS ET, the number of PDSs, the number of accident sequences of the CET, and the number of STCs are n, m, k, and l, respectively. The first accident sequence of PDS ET, i.e., PDS ET-1, is classified into PDS-1. PDS-1 is distributed into every CET having each likelihood, i.e., ν_{1k} to ν_{mk}. The summation is equal to one, as shown in Eq. (1). Each accident sequence of the CET is classified into one STC. CET-1, -2, and -3 are mapped to STC-2, -l, and -1, respectively.

$$\sum_{x=1}^{k} \nu_{ix} = 1, \quad \text{where i} = 1, 2, \ldots, \text{m} \tag{1}$$

Using these logical relationships between PDS ET, PDS, CET, and STC, the mapping fractions between PDS ET and STC can be calculated.

An example of mapping fractions is as follows. There are assumptions made about the relationships between PDS ET, PDS, CET, and STC, as shown in Table 1. The number of PDS ET, PDS, CET, and STC are 10, 5, 9, and 5, respectively. A one-to-one function of PDS ET—PDS and CET—STC are randomly assumed. We can identify two things from this. In the first example, PDS ET-4, -9, and -10 are classified to PDS-1. This means that PDS ET-4, -9, and -10 have the same plant damage characteristics as the initial condition of severe accident progression analysis. Accordingly, mapping fractions of three accident sequences of PDS ET into every CET are the same, i.e., $\nu_{11} - \nu_{19}$. Second, although CET-2, -6, and -7 are different accident sequences of severe accident progression, they are classified into one CF mode, i.e., STC-2. This means that the accident sequences of CET-2, -6, and -7 have the same CF mode.

From the example, we can easily calculate mapping fractions of each PDS into CF modes. The probability of STC-3 CF mode given that PDS-2 has occurred is the summation of ν_{21} and ν_{28}. The frequency of the STC-1 CF mode given that PDS ET-5 has occurred is $f_5{}^*(\nu_{45}+\nu_{49})$.

2.3. Combination of core damage accident sequences and PDS-STC mapping fractions

Fig. 3 shows the conceptual diagram for obtaining the multi-unit Level 2 PSA results. To accomplish this, multi-unit Level 1 PSA results and PDS—STC mapping fractions are required. First, multi-unit Level 1 PSA quantifies the cut-set and core damage frequencies using PDS ETs developed in the single-unit Level 2 PSA. Each accident sequence of PDS ETs has a specific PDS number. Multi-unit Level 1 PSA results consist of the a number of data sets, and each data set consists of an initiating event, all headings representing accident sequences, and a PDS number.

Second, each data set is distributed to every CET. In the PDS—STC mapping fractions table, the mapping fractions from PDS-n into every CET are ν_{n1} to ν_{nk}. The sum of ν_{n1} to ν_{nk} is equal to one. Thus, one core damage accident sequence of a multi-unit Level 1 PSA is divided into k CF accident sequences of a multi-unit Level 2 PSA. Frequency of the k^{th} accident sequence of the multi-unit Level 2 PSA could be quantified by multiplying the frequency of the multi-unit Level 1 PSA and mapping fractions ν_{nk}, i.e., frequency $\times$ ν_{nk}.

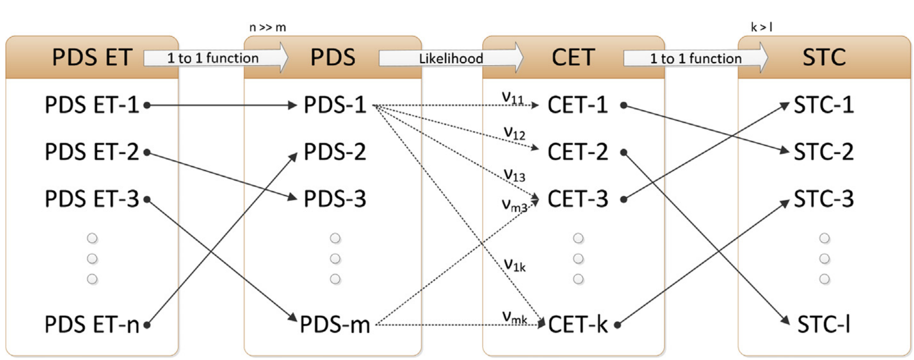

Fig. 2. Diagram on relationships between PDS ET, PDS, CET, and STC.
ET, event tree; CET, containment event tree; PDS, plant damage state; STC, source term category.

Table 1
Example relationship between PDS ET, PDS, CET, and STC (f_x: frequency of PDS ET-x, v_{yz}: mapping fraction from PDS-y into STC-z).

PDS ET (frequency)	PDS	STC-3	STC-2	STC-4	STC-5	STC-1	STC-2	STC-2	STC-3	STC-1
		CET-1	CET-2	CET-3	CET-4	CET-5	CET-6	CET-7	CET-8	CET-9
PDS ET-1 (f_1)	PDS-5	v_{51}	v_{52}	v_{53}	v_{54}	v_{55}	v_{56}	v_{57}	v_{58}	v_{59}
PDS ET-2 (f_2)	PDS-2	v_{21}	v_{22}	v_{23}	v_{24}	v_{25}	v_{26}	v_{27}	v_{28}	v_{29}
PDS ET-3 (f_3)	PDS-3	v_{31}	v_{32}	v_{33}	v_{34}	v_{35}	v_{36}	v_{37}	v_{38}	v_{39}
PDS ET-4 (f_4)	PDS-1	v_{11}	v_{12}	v_{13}	v_{14}	v_{15}	v_{16}	v_{17}	v_{18}	v_{19}
PDS ET-5 (f_5)	PDS-4	v_{41}	v_{42}	v_{43}	v_{44}	v_{45}	v_{46}	v_{47}	v_{48}	v_{49}
PDS ET-6 (f_6)	PDS-2	v_{21}	v_{22}	v_{23}	v_{24}	v_{25}	v_{26}	v_{27}	v_{28}	v_{29}
PDS ET-7 (f_7)	PDS-3	v_{31}	v_{32}	v_{33}	v_{34}	v_{35}	v_{36}	v_{37}	v_{38}	v_{39}
PDS ET-8 (f_8)	PDS-5	v_{51}	v_{52}	v_{53}	v_{54}	v_{55}	v_{56}	v_{57}	v_{58}	v_{59}
PDS ET-9 (f_9)	PDS-1	v_{11}	v_{12}	v_{13}	v_{14}	v_{15}	v_{16}	v_{17}	v_{18}	v_{19}
PDS ET-10 (f_{10})	PDS-1	v_{11}	v_{12}	v_{13}	v_{14}	v_{15}	v_{16}	v_{17}	v_{18}	v_{19}

CET, containment event tree; ET, event tree; PDS, plant damage state; STC, source term category.

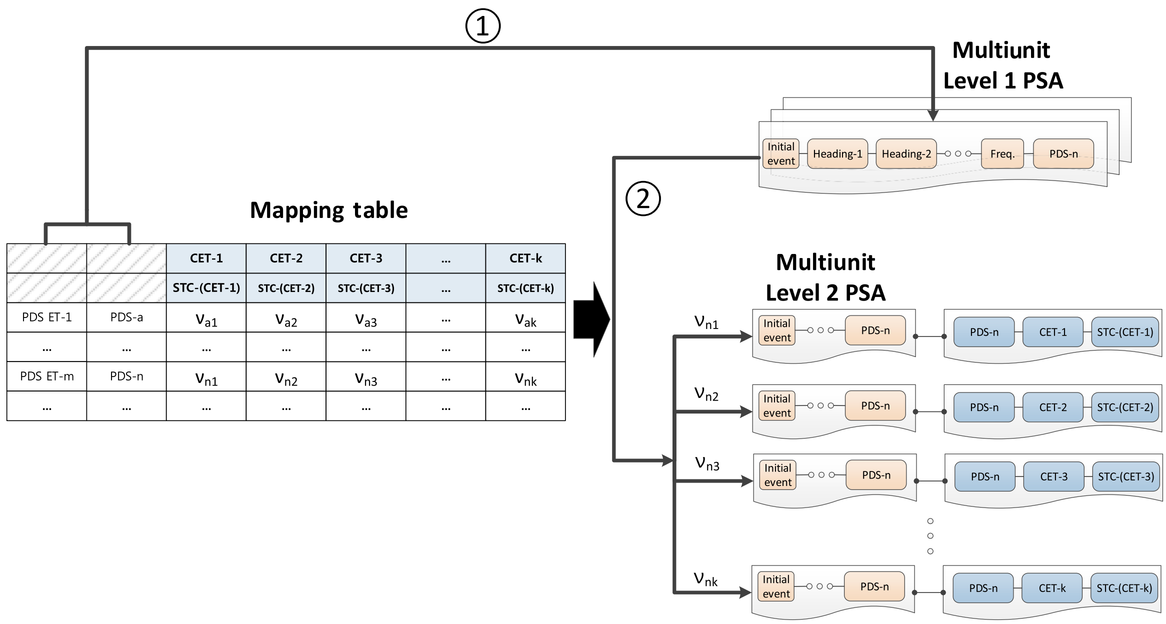

Fig. 3. Schematic of combination core damage accident sequences and PDS–STC mapping fraction table.
CET, containment event tree; PDS, plant damage state; PSA, probabilistic safety assessment; STC, source term category.

3. Application to six-unit OPR1000

The multi-unit Level 2 PSA method was applied to six units of the OPR1000, to quantify the multi-unit CF probability. First, the single-unit Level 2 PSA was developed and quantified (Section 3.1). Second, PDS–STC mapping fractions were derived (Section 3.2). Third, multi-unit Level 2 PSA results were quantified based on mapping fractions and multi-unit Level 1 PSA results (Section 3.3). Herein, LOUHS, LOOP, tsunami, and seismic event were considered as multi-unit common–cause initiators.

For application of the multi-unit Level 2 PSA method to six units of the OPR1000, the major assumptions are as follows:

- The six units of OPR1000 are at full-power operation.
- The six units of OPR1000 are identical in view of severe accident progression.

3.1. Single-unit Level 2 PSA results for OPR1000

To develop the single-unit Level 2 PSA model, PDS logic diagram, CET, and STC logic diagram were developed and are shown in Figs. 4–6, respectively. There are 39 PDSs, 100 end states of CETs, and 21 STCs. This means that each of the 39 initial conditions has a likelihood of undergoing 100 severe accident sequences, and 100 severe accidents sequences are grouped into 21 CF modes.

39 PDSs (see Fig. 4) are characterized by 10 major variables and consideration of dependencies between the variables. It consists of two initiating event types, five statuses of safety systems, and three plant physical characteristics. The two initiating event types are containment bypass status and accident type. The five status of safety systems are containment isolation status, in-vessel injection status, containment spray status, hydrogen igniter availability, and initial secondary cooling availability. The three plant physical characteristics are reactor coolant system (RCS) pressure and early and late cavity flooding conditions.

Rupture before core melt is one of the containment isolation status. Rupture before core melt means the containment building is first ruptured due to a gradual increase of containment pressure before the reactor core has been damaged. In the example of large loss of coolant accident (LOCA) cases, if safety injections are available except the containment spray system, then core is well-cooled; however, containment pressure gradually increases. Thus, the containment is damaged first and evaporation of safety injection water leads to core damage. In this case, at the time the core has been damaged, the containment is already ruptured, thus it has different characteristics to the “NOT ISOLATED” branch, which implies isolation valve failure.

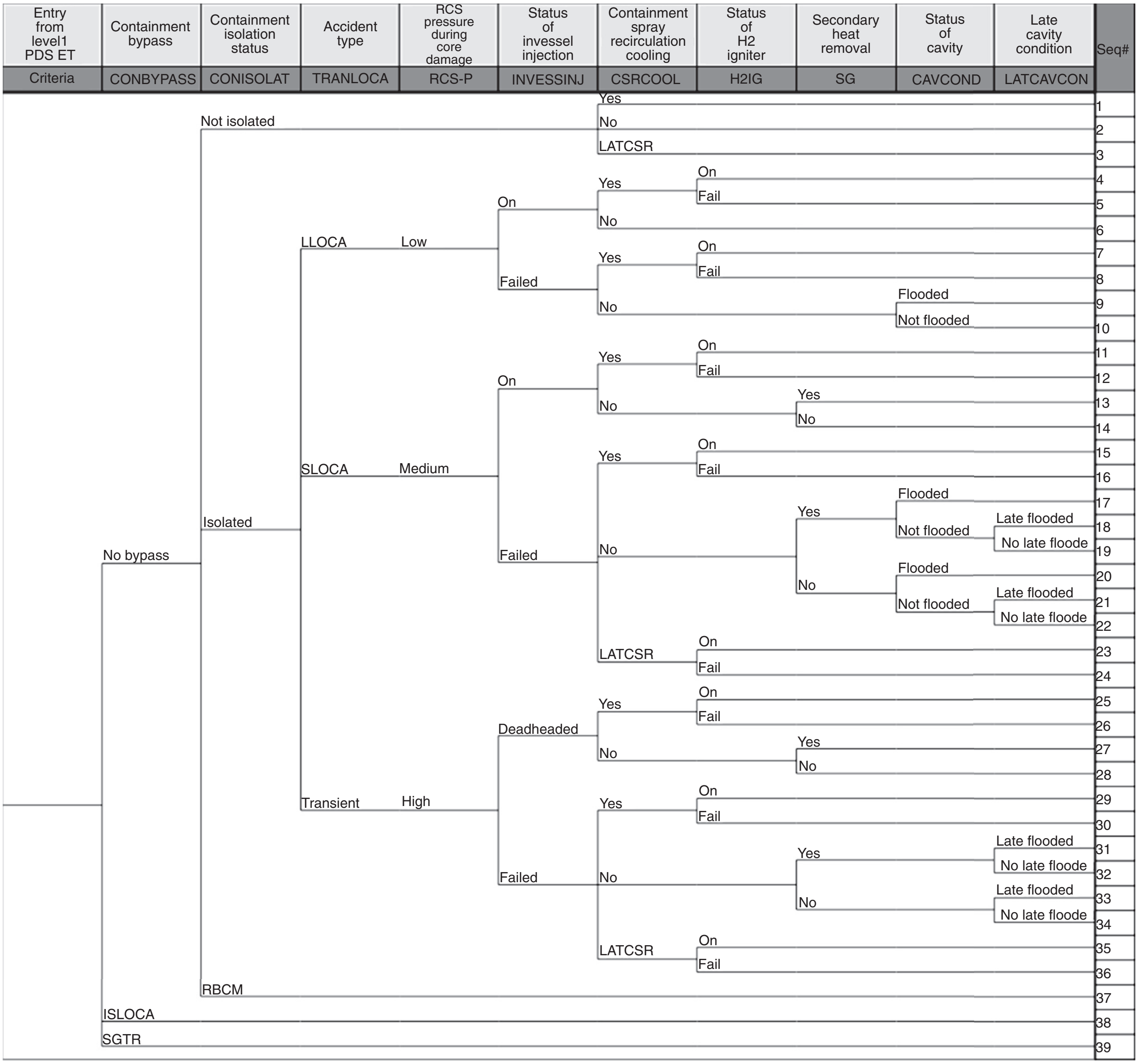

Fig. 4. Plant damage state logic diagram for OPR1000.
RCS, reactor coolant system.

In-vessel injection and containment spray are important mitigation functions for severe accident progression. Although in-vessel injection is not available due to high RCS pressure at the time that the core has been damaged, in-vessel injection could be available later when RCS pressure decreases due to hot-leg rupture or steam generator tube rupture during severe accident progression. Thus, it was considered as “DEADHEADED” branch. Containment spray could be also recovered later by recovering electricity in the case of a station blackout accident. Thus, it is considered the “LATCSR” branch.

One hundred end states of CET (see Fig. 5) were developed by 11 severe accident phenomena, with consideration of time dependencies between the major phenomena. It consists of two plant statuses that are already determined in a PDS logic diagram, five defense-in-depth (DID) integrities, and four major phenomena that affect DID integrity. The five DID integrities are reactor vessel failure, alpha mode failure, i.e., in-vessel steam explosion, early CF, late CF, and containment basement melt-through. The four major phenomena that affect DID integrity are RCS failure by natural circulation of high temperature steam, amount of corium ejection out of cavity, late containment spray availability, and ex-vessel corium coolability.

Twenty-one STCs (see Fig. 6) were characterized by eight major variables with consideration of dependencies between the variables. These variables are containment bypass status, containment isolation status, reactor vessel failure, alpha mode failure, CF timing, containment rupture area, corium debris coolability, and containment spray availability.

Table 2 shows the overall results of CF probability for an OPR1000 single unit. When an internal event has occurred and the core has been damaged, there is an approximately 58% chance of maintaining containment integrity, which means no radioactive material is released into the environment. The remainder, i.e., 42%, in the case that the core has been damaged after an internal event has occurred, means containment integrity has failed. Approximately 17% and 14% are the probabilities of late CF and CF before

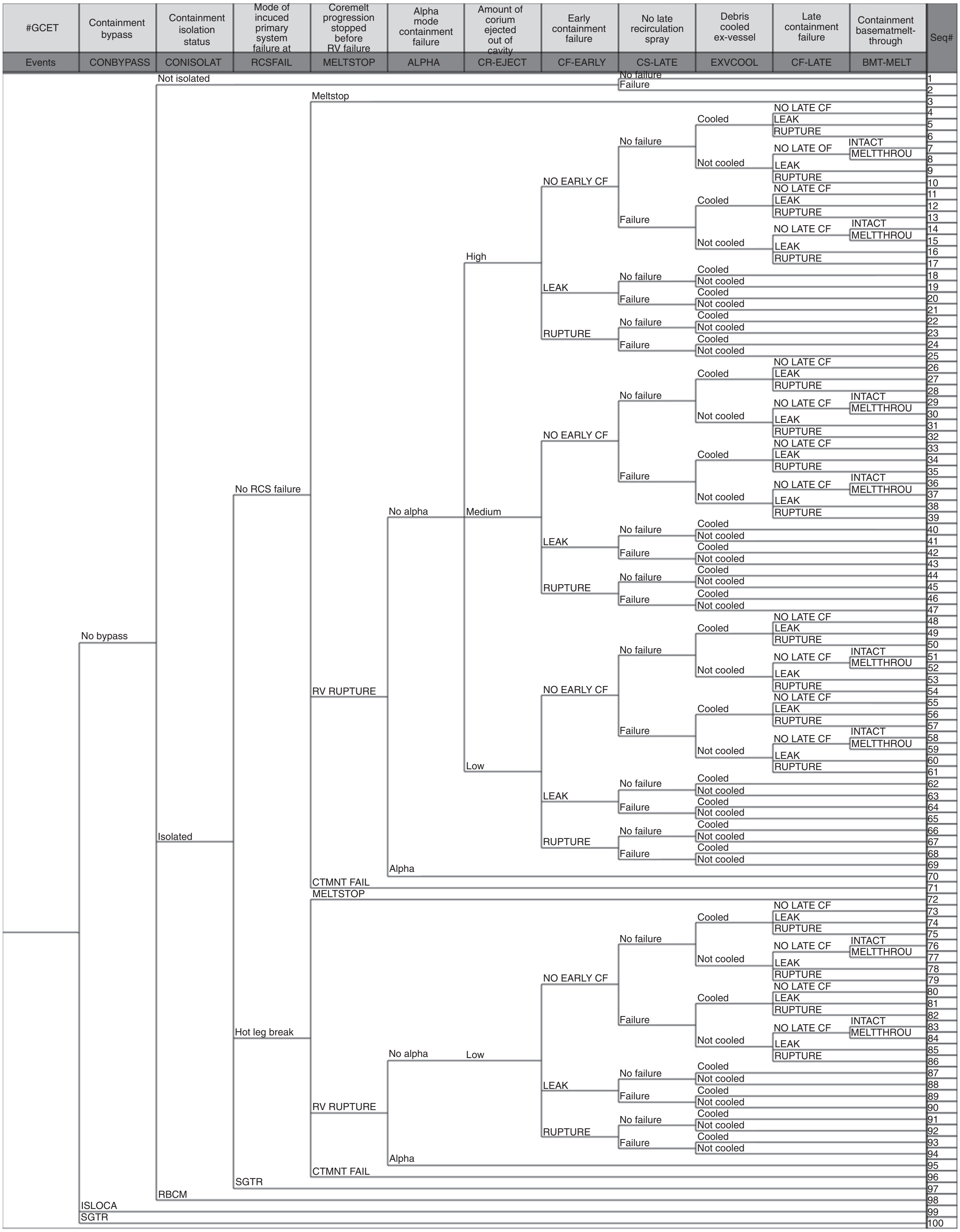

Fig. 5. Containment event tree for OPR1000.

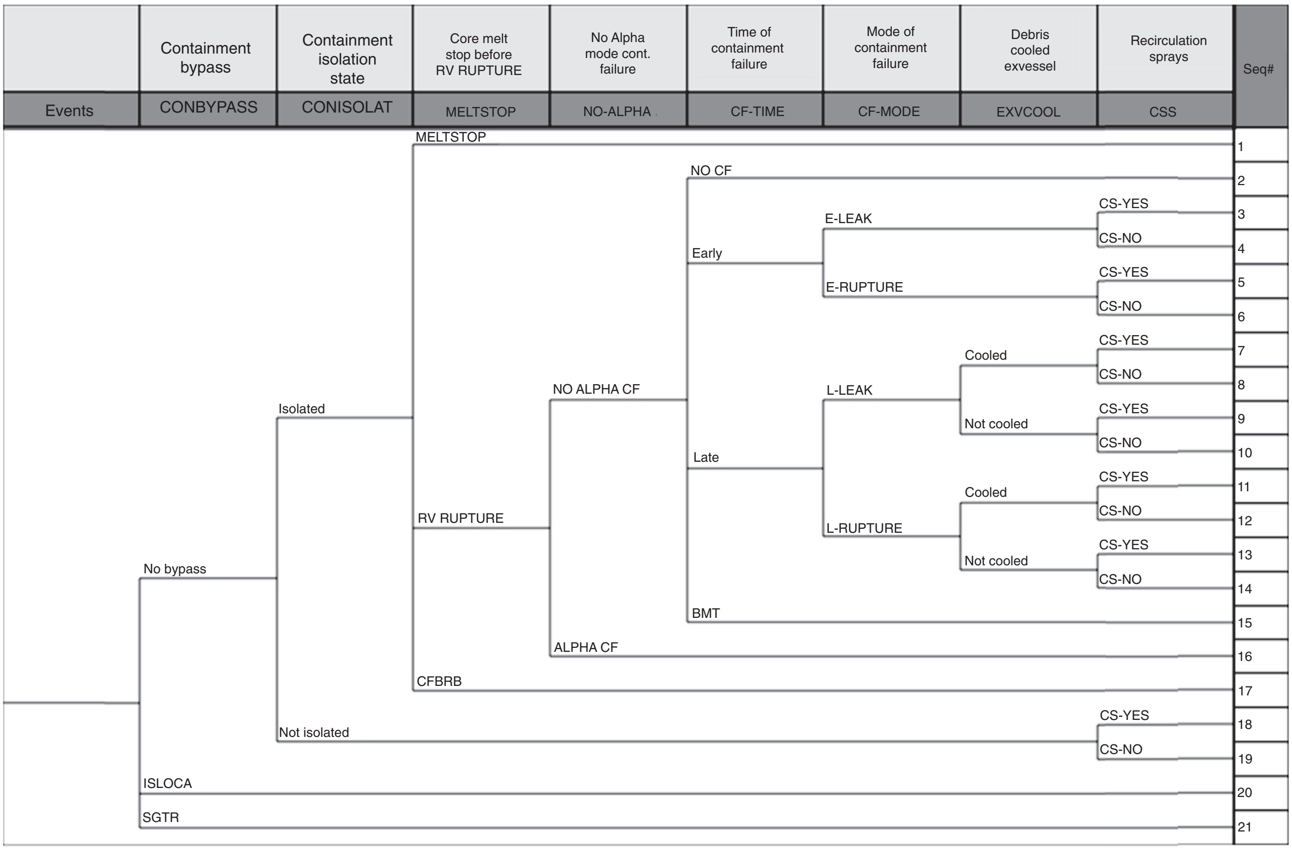

Fig. 6. Source term category logic diagram for OPR1000.

reactor vessel breach (CFBRB), respectively. The probability of bypass failure, which means steam generator tube rupture is the initiating event, is 8.4%. Because the probabilities of early CF (1.9%) and isolation failure (0.1%) are quite small, large early-release (LER) failure probability (10.4%) is mostly caused by bypass failure.

For quantification of a tsunami event risk, two categories were considered for the height of tsunami waves: under the yard and over yard. The CDF or CF probability of an overall tsunami event is calculated by summing each CDF or each CF frequency of two categories. When a tsunami event has occurred and the core has been damaged, there is a 21.5% chance to maintain containment integrity. The remainder (78.5%) is mostly caused by late CF (72.8%). The tsunami event basically assumes that total loss of component cooling water (TLOCCW) has occurred, and accordingly, containment spray is not available. Containment spray is the only function to mitigate gradual increase of containment pressure in OPR1000, thus, late CF probability is largely increased.

For quantification of seismic event risk, five categories were considered for the magnitude of peak ground acceleration (PGA): 0.2–0.4 g, 0.4–0.6 g, 0.6–0.8 g, 0.8–1.0 g, and 1.0–1.2 g. The CDF or CF probability of overall seismic events is calculated by summing each CDF or each CF probability of the five categories. When a seismic event has occurred and the core has been damaged, there is a 14.7% chance of maintaining containment integrity. The remainder (85.3%) is mostly caused by late CF (about 53%) and isolation failure (about 20%). Because the probability of containment valve failure increases as PGA increases, the probability of containment isolation failure rapidly increases in the high PGA region. In the region of

Table 2
Overall results of containment failure probability estimation for OPR1000 single unit.

Containment failure modes			Internal event	Tsunami event	Seismic event					
					Total	0.2–0.4 g	0.4–0.6 g	0.6–0.8 g	0.8–1.0 g	1.0–1.2 g
Level 1 PSA result	Core damage frequency (relative, assumed that CDF of internal event is 1.00)		1.00	0.05	4.87	0.56	1.67	1.61	0.71	0.32
Level 2 PSA results	No containment failure probability (%)		58.2	21.5	14.7	19.3	18.9	16.0	4.5	0.9
	Containment failure probability (CF, %)	Early CF	1.9	3.5	2.5	3.5	3.4	2.6	0.7	0.1
		Late CF	16.8	72.8	52.9	73.9	69.9	55.1	14.7	2.9
		CFBRB	14.0	0.0	8.2	1.3	5.9	15.1	7.2	0.3
		BMT-MELT	0.6	1.1	0.7	1.0	0.9	0.8	0.2	0.0
		Isolation failure	0.1	0.0	20.2	0.0	0.0	9.7	72.5	95.6
		Bypass failure	8.4	1.0	0.7	1.0	0.9	0.7	0.2	0.0
	Large early-release (LER) probability		10.4	4.6	23.4	4.5	4.3	13.1	73.4	95.8
	Total		100	100	100	100	100	100	100	100

BMT-MELT, BaseMenT MELT-through; CF, containment failure; CFBRB, containment failure before reactor vessel breach; PSA, probabilistic safety assessment.

1.0–1.2 g, containment isolation failure probability is 96%. Thus, overall isolation failure probability is quite large, at 20%.

3.2. Mapping fractions from PDS to STC

Mapping fractions from core damage accident sequences, i.e., PDS ET into CF modes, i.e., STC were obtained. Mapping fractions indicate the relationships between PDS ET, PDS, CET, and STC. Accordingly, the relationships were identified by using accident sequences of PDS ET, PDSs, accident scenario of CET, and STCs, which were developed in the single-unit Level 2 PSA for OPR1000 (see Section 3.1). Each of the 39 PDSs was distributed into every one of the 100 end states of CET, having each likelihood. Each end state of CET was classified into a specific STC from among 21 STCs.

Fig. 7 illustrates the mapping fraction of each PDS to CF modes. The Y-axis is the PDS number, which was identified by PDS logic diagram in the single-unit Level 2 PSA for OPR1000 (see Fig. 4). The X-axis is the mapping fraction to every CF mode for each PDS. CF modes were identified by the STC logic diagram in the single-unit Level 2 PSA for OPR1000 (see Fig. 6). CF modes consist of no CF, early CF, late CF, basement melt-through, alpha mode failure, CFBRB, isolation failure, and bypass failure.

The PDS numbers 1–3 are isolation failure, 37 is CFBRB, and 38–39 are bypass failure modes. The others were distributed into several CF modes. Particularly, the PDSs 4–5, 7–8, 11–12, 15–16, 23–26, and 29–30 mostly correspond to no CF. Because a common feature is that the containment spray system is available regardless of availability of other safety features as shown in Fig. 4, the containment spray system is a very important mitigation feature for containment integrity performance. The PDS numbers 6, 9, 13–14, 17, and 20 mostly correspond to the CFBRB, and characteristics include the cavity condition being flooded, whether initially or later, and containment spray system not being available. It is because that although cavity flooding increases the probability of corium cooling, however, the containment pressure gradually increases due to unavailability of the containment spray system. Accordingly, the containment has ruptured before a reactor vessel failure. For the PDS numbers 21, 28, and 33–34, late CF likelihood is more than 50%. They do not have a chance to initially cool RCS down by secondary side, and in-vessel injection and containment spray systems are not available. When there is no probability of cavity flooding later, such as the PDS numbers 10, 19, 22, 32, and 34, then containment basement melt-through could occur with a low likelihood.

3.3. Multi-unit Level 2 PSA results for six-unit OPR1000

Table 3 shows the overall results of CF probability estimation for a six-unit OPR1000 at a site. LOUHS, LOOP, tsunami, and seismic events were considered as multi-unit common–cause initiators. The results were obtained by using mapping fractions (Section 3.2) and multi-unit Level 1 PSA results. For multi-unit Level 1 PSA results, Kim et al.'s results [14] were referred. Major severe accident scenario results with core-damage accident sequences are described in Table 4.

3.3.1. Multi-unit LOUHS

Site CDF due to the multi-unit LOUHS is mostly from the cases that core damage on only one unit (98%). It means the results of two or more unit core damage cases do not meaningfully affect the site CDF. Especially, the frequency of core damage on six-unit is so small that it was truncated by conditional core damage probability cutoff value.

The most important core damage accident sequence is TLOCCW-4 which is TLOCCW without secondary side heat removal. This is

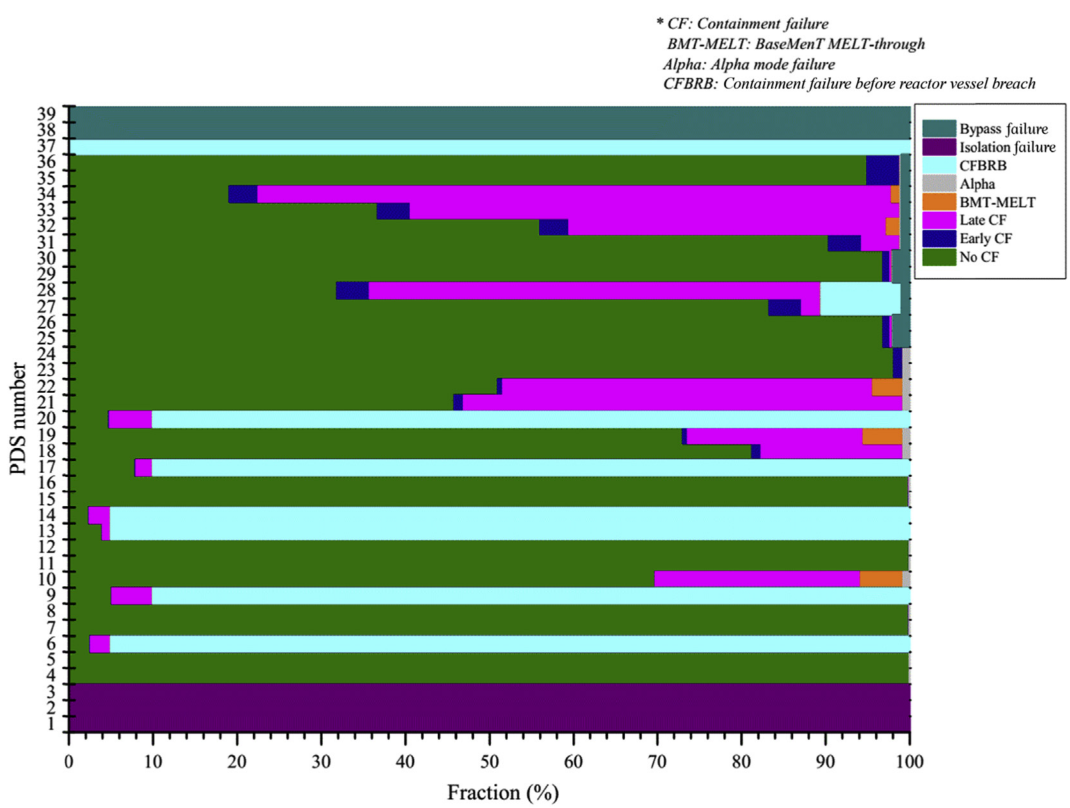

Fig. 7. Mapping fractions result between PDS and STC.
PDS, plant damage state; STC, source term category.

Table 3
Overall results of containment failure probability estimation for six-unit OPR1000 at a site.

Core damaged NPPs/total number of NPPs	Multi-unit loss of ultimate heat sink				Multi-unit loss of off-site power				Multi-unit tsunami				Multi-unit seismic			
	CDF fractions (%)	Containment failure probability (%)			CDF fractions (%)	Containment failure probability (%)			CDF fractions (%)	Containment failure probability (%)			CDF fractions (%)	Containment failure probability (%)		
		No CF	CF			No CF	CF			No CF	CF			No CF	CF	
			LER	Non-LER			LER	Non-LER			LER	Non-LER			LER	Non-LER
1/6	98.0	25.8	4.2	70.0	92.1	91.5	4.8	3.7	41.6	25.8	4.2	70.0	48.6	19.2	4.5	76.3
2/6	1.6	4.7	8.7	86.6	7.1	87.5	9.9	2.6	0.6	4.7	8.7	86.6	17.7	3.6	9.2	87.2
3/6	0.3	0.7	13.1	86.2	0.8	83.9	13.9	2.2	0.1	0.7	12.9	86.4	9.0	0.6	18.5	80.9
4/6	<0.1	0.1	15.6	84.3	<0.1	83.8	16.2	0.0	1.6	9.7	16.7	73.6	6.8	0.1	36.0	63.9
5/6	<0.1	0.0	19.2	70.8	–	–	–	–	0.2	6.0	16.1	77.9	7.3	0.0	50.4	49.6
6/6	–	–	–	–	–	–	–	–	55.9	0.2	21.6	78.2	10.6	0.0	81.1	18.9
Total	100.0	25.3	4.3	70.4	100.0	91.2	5.2	3.6	100.0	11.4	13.9	74.7	100.0	10.0	20.2	69.8

CDF, core damage frequency; CF, containment failure; LER, large early-release; NPP, nuclear power plant.

Table 4
Major accident scenario for six-unit OPR1000 at a site.

Initiating events		Multi-unit Level 1 PSA results [14]		Severe accident scenario	
		Major CDF distribution[a]	Major core-damage accident sequence	PDS (see Fig. 4)	Major containment failure modes (see Fig. 7)
Multi-unit LOUHS		1/6 (98.0%)	TLOCCW-4 (>85%)	PDS-34	Late CF (75%); No CF (19%)
			TLOCCW-2 (<15%)	PDS-19	No CF (73%); Late CF (21%)
Multi-unit LOOP		1/6 (92.1%)	SBOS/R-38 (>74%); SBOS/R-40 (>12%)	PDS-35; PDS-36	No CF (95%)
Multi-unit tsunami	Below yard 5–10 m (41.6%)	1/6 (99.5%)[b]	TS-L-5 (~84%)	PDS-34	Late CF (75%); No CF (19%)
			TS-L-3 (~12%)	PDS-19	No CF (73%); Late CF (21%)
	Over yard >10 m (55.8%)	6/6 (100%)	TS-H-11, 13, 15	PDS-34	Late CF (75%); No CF (19%)
			TS-H-5, TS-H-2	PDS-32	No CF (56%); Late CF (38%)
Multi-unit seismic	0.2 g–0.4 g (27.5%)	1/6 (96.7%)	LOUHS_S-1 (>81%)	PDS-34	Late CF (75%); No CF (19%)
	0.4 g–0.6 g (46.3%)	1/6 (47.1%)	LOUHS_S-1 (>64%); LOOP_S-54 (>15%)	PDS-34	Late CF (75%); No CF (19%)
		2/6 (34.8%)	LOUHS_S-1; LOOP_S-54	PDS-34	Late CF (75%); No CF (19%)
	0.6 g–0.8 g (17.3%)	5/6 (33.5%) 4/6 (27.7%)	LOUHS_S-1; VR_S–05	PDS-34; PDS–09	Late CF (75%); No CF (19%); CFBRB (90%); No CF (5%)
	0.8 g–1.0 g (6.1%)	6/6 (77.0%)	VR_S-18; VR_S-12; ABF_S-2; LOUHS_S-2	PDS–02	Isolation failure (100%)
	1.0 g-1.2 g (2.8%)	6/6 (85.7%)			

CDF, core damage frequency; CF, containment failure; CFBRB, containment failure before reactor vessel breach; LER, large early-release; LOUHS, loss of ultimate heat sink; LOOP, loss of off-site power; PDS, plant damage state; TLOCCW, total loss of component cooling water.
[a] n/6 (x %), n: the number of core-damage plants; x: CDF fractions among site CDF.
[b] Single-unit CDF is 99.5% among CDF caused by tsunami below yard, in other word, single-unit CDF is 41.4% (41.6%* 99.5%) among site CDF of tsunami event.

more than 83% (98% × 85%) among site CDF. The characteristics of this accident sequence at the time core has been damaged are that RCS pressure is high and safety injection is not available, and containment spray is not available. It maps into PDS-34 and it makes about 75% late CF and about 19% no CF.

Remainder is mostly due to TLOCCW-2 which is TLOCCW with reactor-coolant-pump seal LOCA. This is about 15% (98% × 15%) among site CDF. The characteristics are that RCS pressure is medium, safety injection is not available, containment spray is not available, and SG initial cooling is available. It maps into PDS-19 and it makes about 73% no CF and about 21% late CF.

Accordingly, for single-unit core damage which is dominant in multi-unit LOUHS, no CF probability, LER probability, and non-LER probability are 25.8, 4.2, and 70.0%, respectively.

3.3.2. Multi-unit LOOP

Similar with multi-unit LOUHS, single-unit core damage case is dominant in site CDF of multi-unit LOOP (92.1%). Five-unit and six-unit CDFs are so small that they were truncated by conditional core damage probability cutoff value.

The most important core damage accident sequence is that off-site power is recovered after core damage, and containment spray system is available. It maps into PDS-35 or 36, and it makes more than 95% no CF. Accordingly, for single-unit core damage which is dominant in multi-unit LOOP, no containment probability, LER probability, and non-LER probability are 91.5, 4.8, and 3.7%, respectively.

3.3.3. Multi-unit tsunami event

In case of tsunami induced multi-unit accident, fractions of single-unit CDF and six-unit CDF are 41.6% and 55.9%, respectively. Single-unit core damage cases are almost cause by tsunami wave below yard (99.5%), whereas six-unit core damage cases are cause by tsunami wave over yard.

For the single-unit CDF, the most important core damage accident sequence is TS-L-5 which is TLOCCW without secondary side heat removal (about 84%). Remainder is mostly from TS-L-3 which is TLOCCW with reactor coolant pump seal LOCA (about 12%). It maps into PDS-34 and PDS-19, respectively.

Accordingly, for single-unit core damage, no CF probability, LER probability, and non-LER probability are 25.8, 4.2, and 70.0%, respectively. For the six-unit CDF, PDS-34 and PDS-32 are dominant. Accordingly, for six-unit core damage case, no CF probability, LER probability, and non-LER probability are 0.2, 21.6, and 78.2%, respectively.

3.3.4. Multi-unit seismic event

For multi-unit seismic event, inter-unit correlation coefficients are very important. Kim et al. [14] showed that as the inter-unit correlation coefficient increases, the number of damaged unit increases. For this application, 0.3 was used for all categories as an inter-unit correlation coefficient. In case of seismic induced multi-unit accident, fractions of site CDF are 48.6%, 17.7%, 9.0%, 6.8%, 7.3%, and 10.6% for from single-unit CDF to six-unit CDF, respectively. As PGA increases, the number of core damage plants tends to increase. Especially, in high magnitude of PGA such as 1.0g–1.2g, all unit core damage cases are dominant. Moreover, CF mode is almost isolation failure because as PGA increases, containment isolation systems are likely vulnerable.

4. Discussion

For better discussion, we defined “core-damage plant” as the NPP unit in which the core has been damaged. Accordingly, “x/n core-damage plants” mean the number of NPP units in which the core has been damaged is x-units among n-units at a site. When multi-unit Level 1 PSA results are given for n units at a site, the multi-unit Level 2 PSA quantifies the likelihoods of all CF modes for from 1/n core-damage plant to n/n core-damage plants. In here, CF modes consist of no CF, early CF, late CF, CF before reactor vessel breach, basement melt-through, isolation failure, and bypass failure, as defined in Table 2. LER CF includes early CF, isolation failure, and bypass failure. We classified all CF modes into three types in view of site CF modes, as followings.

- No CF: there are no CFs in a site
- LER CF: there is at least one CF with large early-release CF
- non–LER CF: there are CFs with non–large early-release CF

Site CF frequency (or site large early-release CF frequency) is defined as the frequency at which one or more units at a site experience CF (or large early-release CF). Thus, site CF frequency for n units at a site could be calculated by Eq. (2)

$$\text{Site containment failure frequency} = \sum(i=1:n)CF_i{*}CDF_i \tag{2}$$

where, CF_i is the site CF probability given that i/n core-damage plants, CDF_i is the frequency at which i units experience core damage.

Accordingly, site CF probability is defined as Eq. (3)

$$\text{Site containment failure probability} = \sum(i=1:n)CF_i{*}CDF_i \Big/ \sum(i=1:n)CDF_i \tag{3}$$

Using Eq. (3) with Table 3, site CF probability for LOUHS, LOOP, tsunami, and seismic events are 74.5%, 8.8%, 89.0%, and 90.0%, respectively; in addition, site LER CF probability are 4.3%, 5.2%, 14.2%, and 20.2%, respectively.

If core damage accident sequence of each core-damage plant is exactly same as each other, site CF probability given that i/n core-damage plants, i.e., CF_i could be easily estimated using single-unit CF probability. For example, when single-unit CF probability is 30%, then site CF probability of six core-damage plants is 88% ($0.88 = 1-(1-0.3)^6$). Fig. 8 shows the CF probabilities estimated by single-unit CF probability using the Eq. (4).

$$y = 1 - (1-p)^n \tag{4}$$

where, y is the site CF probability given that n core-damage plants, p is the single-unit CF probability, and n is the number of core-damage plants.

However, because accident sequence of each core-damage plant is different from each other in the most cases and every sequences are grouped into different PDSs, thus, the site CF probability in practical case cannot be simply predicted by hand calculation like Eq. (4).

Another important thing we should consider is that we need to distinguish between “probability” and “frequency” of CF. Let us assume that the frequency of 1/6 core-damage plant and 3/6 core-damage plants are 1.0E-5 (/y) and 1.0E-6 (/y), respectively and the conditional CF probability of 1/6 core-damage plant and 3/6 core-damage plants are 45% and 90%, respectively. A simple comparison between CF probabilities may yield the conclusion that containment performance given 3/6 core-damage plants (90%) is two times worse than containment performance given 1/6 core-damage plants (45%). However, CF frequencies in the above example are 4.5E-6 (/y) (1.0E-5 × 0.45) and 9.0E-7 (/y) (1.0E-6 × 0.9) for 1/6 core-damage plant and 3/6 core-damage plants, respectively. This means that containment integrity given for 1/6 core-damage plant is five times more severe than containment integrity given for 3/6 core-damage plants.

For a better understanding of the results in view of risk, we suggest a diagram as shown in Fig. 9, which represents the multi-unit plant-damage distribution by integrating the Level 1 PSA results and Level 2 PSA results. The X-axis is the CDF distribution, which means each CDF fraction for the number of core-damage plants when the site CDF is one. In the example of the figure, the fraction of only one NPP in which the core has been damaged is 65%. In the case of two and three NPPs, the fractions are 20% and 15%, respectively. The Y-axis is the probability of three CF modes: No CF, LER CF, and non–LER CF. The green region (A) is the No CF and the red region (C) represents the LER CF. When the entire area of the diagram is one, each area A, B, and C are the specific probabilities

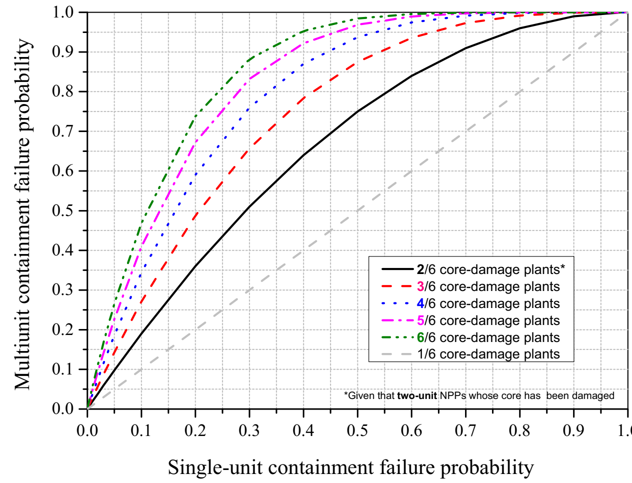

Fig. 8. Containment failure probability estimated by single-unit containment failure probability.
NPP, nuclear power plant.

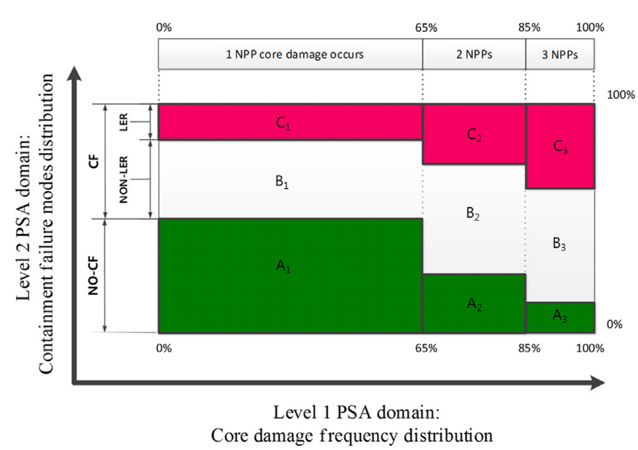

Fig. 9. Example of multi-unit plant damage distribution map.
CF, containment failure; LER, large early-release; NPP, nuclear power plant; PSA, probabilistic safety assessment.

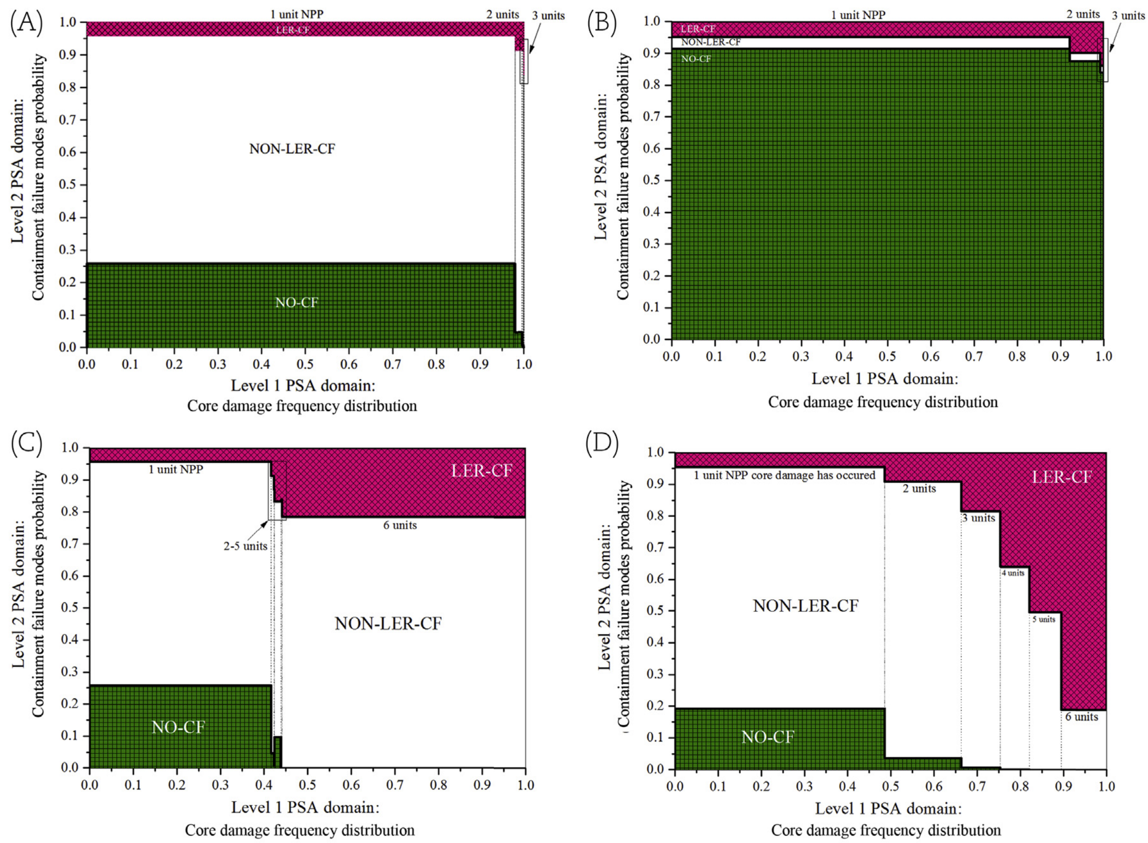

Fig. 10. Results of multi-unit plant damage distribution map for six-unit OPR1000. (A) Loss of ultimate heat sink. (B) Loss of off-site power. (C) Tsunami event. (D) Seismic event. CF, containment failure; LER, large early-release; NPP, nuclear power plant; PSA, probabilistic safety assessment.

given that core damage has occurred for at least one NPP. Area A_x, B_x, and C_x are the likelihoods of No CF, non-LER CF, and LER CF with x core-damage plants, respectively. Area $A_1+A_2+A_3$ means No CF probability, given that core damage has occurred for at least one NPP. Area $B_1+B_2+B_3+C_1+C_2+C_3$ means CF probability, given that core damage has occurred for at least one NPP. The width and height of each area represent the CDF and the probability of CF mode, respectively. Thus, the each specific area indicates conditional probability which is weighted by frequency.

Fig. 10A–10D show the multi-unit plant damage distribution map for a six-unit OPR1000. Fig. 10A–10D illustrate the results for LOUHS, LOOP, tsunami and seismic event, respectively.

Three types of CDF distribution were identified. First, CDF distribution is considerably concentrated in the only one core-damage plant, which is more than 90% (LOUHS 98% and LOOP 92%), whereas CDF distribution is somewhat divided from one to six core-damage plants in natural hazard, i.e., tsunami and seismic event. Second, the CDF in tsunami event is concentrated in the one core-damage plant and six core-damage plants having about 42% and 56% occurrence, respectively. Third, the CDF of a seismic event is well distributed in all numbers of core-damage plants. These three types of CDF distribution are mostly from the inter-unit dependencies considered in multi-unit Level 1 PSA. The tsunami event was divided into two categories by tsunami wave height. Tsunami wave over-yard was fatal to all NPPs at a site, thus the fully correlated inter-unit dependency was considered. Accordingly, when an over-yard tsunami has occurred, the primary auxiliary building of all six-unit NPPs has been damaged at the same time. Conversely, in a tsunami wave below the site-yard, inter-unit dependencies were considered for only inter-unit common cause failure, thus only one NPP was significantly damaged. According to the threshold effect of wave height, both side probabilities, i.e., one- and six-unit core-damage plant probabilities, are dominant in a tsunami event. For seismic events, as PGA increases, the initiating frequency decreases, however conditional core damage probabilities and the number of core-damage plants gradually increases.

Containment performance results in the LOUHS- and LOOP-induced multi-unit risk were not recognized as important when compared to single-unit risk because the CDF and CF probability distributions were concentrated in single-unit damage. This biased structure is mostly because of weak inter-unit dependency. However, containment performance results in the tsunami- and seismic-induced multi-unit risk are significantly important because strong inter-unit dependencies were considered.

5. Summary and conclusion

In this article, the multi-unit Level 2 PSA method was developed by introducing the mapping fraction, which is obtained by mapping the PDS into STC. The multi-unit Level 2 PSA method was applied to six-unit full-power operating OPR1000 for LOUHS-, LOOP-, tsunami-, and seismic events–induced multi-unit risk. The plant performance results were illustrated by plant damage distribution maps in the discussion section.

The following conclusions can be drawn in view of a single-unit Level 2 PSA:

- In internal events, no CF probability is about 58%. Among the CF case (48%), late CF and CFBRB mainly occurred having 17% and 14% probabilities, respectively.
- In tsunami events, two categories were considered for the wave height: wave height below the site-yard and over the site-yard.

Because tsunami assumes TLOCCW accidents, containment spray is not available without any recovery actions. Accordingly, CF probability is 78.5%, which is mostly from late CF (72.8%).

- In seismic events, five categories were considered along the magnitude of PGA: 0.2–0.4 g, 0.4–0.6 g, 0.6–0.8 g, 0.8–1.0 g, and 1.0–1.2 g. As PGA increases, CF severity dramatically increases. The overall CF probability (85.3%) is considerably higher, which is mostly due to late CF (approximately 53%) and isolation failure (approximately 20%).

The following conclusions can be drawn regarding the multi-unit Level 2 PSA for six-unit OPR1000:

- The each area of multi-unit plant damage distribution map (see Fig. 10) indicates the number of core damage plants and CF probability. The site CF probability is defined as probability that there is at least one CF among all core-damage plants. Accordingly, the site CF probability is calculated by weighting the each CF probability by CDF.
- In case of LOUHS and LOOP, site CF probabilities are 74.7% and 8.8%, respectively and site LER CF probabilities are 4.3% and 5.2%, respectively. Because CDF distribution is concentrated on one core-damage plant (i.e., 98% for LOUHS, 92% for LOOP), two or more units failure cases are not significant for site risk.
- In case of tsunami and seismic events, site CF probabilities are 88.6% and 90.0%, respectively and site LER CF probabilities are 14.2% and 20.2%, respectively. The natural hazard–induced (tsunami and seismic) multi-unit containment performance was significantly important because fraction of multi (i.e., two or more) core-damage plants is not negligible due to inter-unit fragility dependencies strongly considered in multi-unit Level 1 PSA.

The present study has the following limitations which should be considered for further study:

- During the progression of a severe accident in a specific NPP, the NPPs near the damaged NPP will be affected by the damaged NPP. The impact between inter-units will originate not only from the resource allocation problem but also from mechanical failure itself, caused by released radioactive material from the damaged NPP. However, at this stage, (1) we have no identifiable information about the inter-unit impact and (2) the single-unit Level 2 PSA was not constructed for modeling resource allocations such as operators and sharing mitigation systems. Thus, in this multi-unit Level 2 PSA method, inter-unit impacts were not considered at all.
- In this application, identical NPP type and full-power operation modes for all units were assumed. Various different types and operation modes should be considered for better insight.

Conflicts of interest

The authors declare there are no conflicts of interest.

Acknowledgment

This work was supported by the National Research Foundation of Korea (NRF) grant funded by the Korean government (MSIT: Ministry of Science and ICT) (No. 2017M2A8A4015287).

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