Feasibility on the Use of Quantitative DID Information for the Design of a New NPP

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1. Introduction

Pilot Application of Defense in Depth (DID) characterization methodology [1,2] has been performed for a reference nuclear power plant (NPP). It aims at the investigation of the feasibility on the use of quantitative DID information for the design of a new NPP including its use on the safety issues of the current operating NPP.

5 Levels of DID which have been proposed in the INSAG [3] are estimated using probabilistic safety assessment (PSA) model. Level 1 and 2 PSA model for internal event are used to characterize 4 levels of DID.

Initiating event (IE), core damage event sequence, and containment failure event sequence are used to characterize 4 levels of DID. The number of DIDs and the DID toughness in terms of barrier failure probability are the major measures to investigate the DID structure of an NPP.

2. DID Estimation Process and Results

In this section, methods for estimating each DID level and results are described with the overall results in the final subsection.

2.1 Estimation of DID Level 1 and 2

DID level 1 and 2 are related to the occurrence of initiating event (IE) which is the starting point of level 1 PSA model. To estimate DID level 1 and 2, the IE should be disassembled into root cause of IE and mitigation of the root cause. As discussed in the DID characterization methodology [2], current level 1 PSA model does not have detailed information structure of IE to identify DID revel 1 and 2 directly. To handle this difficulty, we applied fault tree method for IE caused by supporting system failure discussed in the previous study [2], event decomposition for loss of coolant accident (LOCA) used in the NUREG/CR5750 [4], and simple geometric average for IE with no information for its generation.

Table 1 shows the results of DID 1 and 2 in terms of barrier failure probability from the IE of level 1 PSA model for reference NPP.

Table I: DID level 1 and 2 failure probability of a reference NPP

IE	DID1	DID2
Large LOCA	2.35E-04	1.23E-02
Medium LOCA	7.86E-03	2.06E-02
Small LOCA	1.87E-02	1.87E-02
SGTR	7.01E-02	7.01E-02
Reactor Vessel Rupture	1.85E-04	1.85E-04
Interfacing System LOCA	3.96E-04	2.54E-05
General Transient	8.40E-01	8.40E-01
Loss of Feedwater	2.02E-01	2.02E-01
Loss of Condenser Vacuum	2.72E-01	2.72E-01
Partial Loss of Component Cooling Water	7.01E-02	7.01E-02
Total Loss of Component Cooling Water	9.65E-01	2.17E-04
Inner Containment MSLB	1.87E-02	1.87E-02
Outer Containment MSLB	8.56E-02	8.56E-02
Consequential SGTR by MSLB	7.67E-03	5.00E-02
ATWS	7.52E-01	1.98E-06
Consequential SGTR by ATWS	1.49E-06	3.40E-02
LODCA	4.96E-02	4.96E-02
LODCB	4.96E-02	4.96E-02
LOKVA	6.57E-02	6.57E-02
LOOP	1.54E-01	1.54E-01
SBO by DG Starting failure	2.36E-02	3.45E-04
SBO by DG running failure	2.36E-02	1.56E-03
Consequential SLOCA by PSV reseating failure	1.43E-01	6.76E-05

2.2 Estimation of DID Level 3 and 4

The overall role of each phase of DID in an NPP is to maintain the plant's key safety functions. The main key safety functions of an NPP is as follows:

- Reactor reactivity control
- Reactor coolant system pressure control
- Maintenance of reactor coolant system inventory
- Decay heat removal
- Containment pressure control

DID level 3 is a stage in which, when an IE occurs, it performs a function to prevent the IE from developing into a severe accident. In the design of a nuclear power plant, most of the safety resources are allocated to this DID level. So, there may be more than one safety systems to maintain whole key safety functions by which DID level 3 can be divided into sublevel of DID. The reference NPP has two independent safety systems which can preserve whole key safety function except for reactivity control. One is the Auxiliary feedwater system and the other is the ECCS with safety depressurization system. To estimate the detailed features of DID level 3, the DID level 3 is subdivided into two levels of DID, DID level 3.1 and 3.2 in this pilot study. Table 2 shows the two sub-level of DID level 3. As shown in the table, some IEs can have only one DID level because the condition for the system operation are not met with the accident progression. For example, IE group of LOCA can only DID3.2 since AFW cannot be used to remove decay heat.

Table 2: DID level 3 and 4 failure probability of a reference NPP

IE	DID3.1	DID3.2	DID4
Large LOCA	1.00E+00	2.42E-03	1.43E-06
Medium LOCA	1.00E+00	2.17E-03	3.96E-01
Small LOCA	1.00E+00	8.66E-04	2.48E-02
SGTR	1.00E+00	4.12E-05	6.76E-03
Reactor Vessel Rupture	1.00E+00	1.00E+00	1.71E-02
Interfacing System LOCA	1.00E+00	1.00E+00	1.00E+00
GTRN	2.42E-07	4.01E-01	2.74E-01
LOFW	5.69E-06	3.10E-02	1.09E-03
LOCV	5.69E-06	3.11E-02	1.12E-03
Partial LOCCW	2.93E-04	3.19E-02	5.18E-01
Total LOCCW	2.24E-03	1.00E+00	6.94E-01
Inner Containment MSLB	7.70E-04	1.65E-02	1.98E-02
Outer Containment MSLB	4.94E-04	2.55E-02	3.36E-02
Consequential SGTR by MSLB	4.66E-03	7.91E-03	8.13E-01
ATWS	3.31E-01	1.20E-01	1.69E-07
Consequential SGTR by ATWS	4.78E-03	8.05E-02	5.14E-04
LODCA	2.22E-04	2.39E-01	4.27E-01
LODCB	1.76E-04	1.70E-01	4.13E-01
LOKVA	2.90E-04	3.91E-02	5.08E-01
LOOP	2.08E-05	4.22E-02	8.42E-04
SBO by DG Starting failure	1.73E-01	1.80E-01	2.14E-02
SBO by DG running failure	1.43E-01	8.73E-02	2.92E-02
Consequential SLOCA by PSV reseating failure	2.41E-02	7.94E-02	5.29E-01

DID level 4 can be estimated by using the containment large release frequency which is the main risk measure from the level 2 PSA [2] as shown in Table 2.

2.3 Overall DID Structure of a reference NPP

To represent the overall DID structure of a reference plant, we used two DID measures developed from the previous study [2]. One is the DID toughness and the other is the total number of DID barriers used in the defense of an IE.

DID toughness is defined as follows:

$$\boldsymbol{T(Df_i) = \log\left(p(Df_i)\right)^{-1}} \qquad (1)$$

Where T() is the DID toughness function and Df_i is the DID failure event. For a binomial distribution of B(n, p), if the mean value is equal to 1, the number n can be expressed as the inverse of the failure probability which can be interpreted as the expected number of challenges for the DID level be failed. For example, if a DID level have a value of 3 as the DID toughness, it means that it needs 3000 challenges of DID level to obtain 1 expected DID barrier failure.

Figure 1 shows the overall DID structure of a reference NPP. As shown in the figure, the reference NPP has sound DID structure for some IE, in which the total DID toughness value exceed 10 and the each DID level make a good balance in their relative values (see the Large LOCA case). However, in some IEs, DID structure has some weakness in terms of toughness and the balance among each DID level.

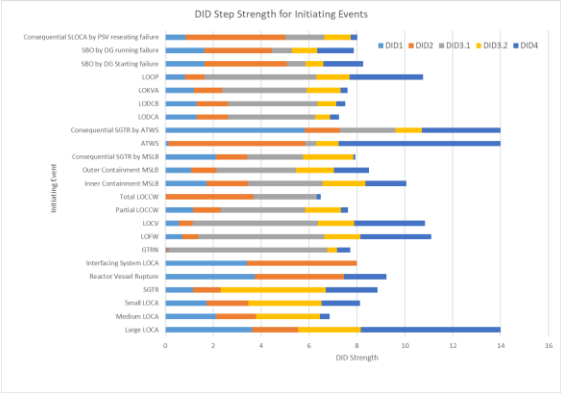

Fig. 1. DID structure of a reference NPP

The new design for a small modular reactor requires a core damage frequency less the 1.0E-9 as their top tire requirement (TTR), which can be converted into more than 9 DID toughness for all IEs. From a DID structure of an NPP design, one can easily identify the design weaknesses and can help to find an alternative to reinforce the design weakness.

3. Conclusions

The pilot application of DID characterization methodology was performed for a reference NPP. Each level of DID can be quantified using information from a PSA model. DID characterization can represent the detailed safety structure of an NPP, by which safety weakness of an NPP can be effectively improved. We expect that the present DID characterization methodology can be used in the new reactor design with high level of safety goal.

REFERENCES

[1] H. G. Lim, Quantification of Defense in Depth using Risk model, Transactions of the Korean Nuclear Society Spring Meeting Jeju, Korea, May 13-14, 2021

[2] H. G. Lim, Practical Application of PSA Model for the Evaluation of Defense in Depth, Transactions of the Korean Nuclear Society Spring Meeting Jeju, Korea, May 19-20, 2022.

[3] Assessment of Defense in Depth for Nuclear Power Plants, Safety Report Series No. 46, IAEA, Vienna 2005.

[4] NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants: 1987 – 1995, USNRC, 1998.