Identifying the dynamic effect of manual responses on the progression of a steam generator tube rupture accident – A case study[1]

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Abstract: Probabilistic safety/risk assessment (PSA/PRA) has been widely used for many decades as a tool for securing the operational safety of nuclear power plants (NPPs). One of the typical information available from PSA results is the catalog of accident scenarios leading to potentially undesirable consequences (e.g., core damage) that are combined by the failures of systems (e.g., malfunctions of automatic responses) and the degradation of manual responses (e.g., human errors or delayed response times) involved in the operation of NPPs. In current PSA technique, in general, the catalog of accident scenarios should be largely dependent on the analysis results of a thermal-hydraulic (TH) code that emulates the physical responses of key process parameters (e.g., pressure and temperature) in a given accident condition. It is evident that various kinds of automatic responses and manual responses can directly and/or indirectly influence the progression of an initiating event. This implies that a large number of TH code runs is required because an NPP consists of many kinds of components of which the status will directly or indirectly affect the trends of key process parameters in the course of an initiating event progression. In order to address this issue, it is necessary to clarify at least the following two aspects: (1) the effect of manual responses on the accident progression and (2) practical way to overcome the curse of TH code runs (i.e., requiring a lot of TH code runs). In dealing with the first aspect, in this study, over 14 million simulation conditions are randomly generated based on the combinations of 79 manual responses that are identified from the SGTR EOP of a reference NPP. As a result, it is observed that about 14.7% of the simulation conditions are related to the stuck open of a main steam safety valve (MSSV) that can result in an undesirable consequence. This implies that the actual path of an accident scenario can be significantly affected by the dynamic combinations of manual responses. Accordingly, in order to properly incorporate this dynamic effect into PSA, it is necessary to figure out practical solutions to deal with the curse of TH code runs.

Keywords: PSA, Scenario progression, Manual responses, Dynamic effect

1. INTRODUCTION

For many decades since the occurrence of TMI-2 accident, probabilistic safety/risk assessment (PSA/PRA) technique has been widely used in many countries operating nuclear power plants (NPPs) because of many benefits, such as answering the question of “What can go wrong?” In other words, as PSA allows us to identify the catalog of accident scenarios leading to an undesirable consequence (e.g., core damage) based on the promising combinations of failed systems (e.g., malfunctions of automatic responses) and the degradation of human performance (e.g., human errors), it is possible to significant enhance the safety of NPPs based the identified accident scenarios. To this end, the use of a thermal-hydraulic (TH) code is indispensable.

The role of the TH code is to provide information (e.g., the trends of key process parameters such as temperature and pressure) that can be used for analyzing the physical behavior of an NPP in a specific accident condition. Accordingly, the input of the TH code includes the status of both crucial components/systems and human operators who are responsible for the operation of an NPP. This implies that a large number of TH code runs is needed because the NPP consists of many components/systems of which the status will directly or indirectly affect the trends of the key process parameters. The number of required TH code runs will drastically increase when the performance of human operators come into play as the input of the TH code. Unfortunately, although the progression of an accident scenario can be significantly changed by the dynamic effect originated from the combinations of both automatic responses and manual responses,

[1]*A large portion of descriptions and case study results in this paper can be found from the paper of “A case study to address the limitation of accident scenario identifications with respect to diverse manual responses” that is under review in Reliability Engineering and System Safety (Park and Kim, 2022). More detailed explanations on simulation conditions are given in this reference with their promising contributions to the conventional PSA/PRA.*

it seems that the current PSA did not fully consider this dynamic effect. In other words, due to the limitation of available resources, the number of TH code runs should be reduced to an affordable range by introducing several assumptions such as “a manual response that should be conducted at a specific time point without any human errors”. Therefore, many researchers claimed that the identification of accident scenarios would be insufficient unless manual responses are properly considered [Bogumil, 1982; Mosleh, 2014; Parhizkar et al., 2021; Verma et al., 2016].

From this concern, the purpose of this paper is to elaborate the contribution of the dynamic effect on the safety of an NPP. To this end, as a case study, a total of 14,348,907 simulation conditions were randomly generated from the combinations of 79 manual responses that are identified from the emergency operating procedure (EOP) of the steam generator tube rupture (SGTR) being used in a reference NPP. As a result, it is observed that about 14.7% of the simulation conditions are related to the stuck open of a main steam safety valve (MSSV) that can result in unexpected consequences such as the initiation of a small break loss of coolant accident (SBLOCA) or the provision of a direct pathway for the release of a radioactive material to the environment.

This paper is organized as follows. In Section 2, the catalog of representative manual responses related to an SGTR accident is briefly explained. Then, Section 3 provides detailed descriptions on how this case study is conducted with its result. After that, Section 4 suggests a promising solution that can be used for supporting the identification of accident scenarios with affordable amount of resources (e.g., time and manpower).

2. REPRESENTATIVE MANUAL RESPONSES WITH RESPECT TO AN SGTR

According to MacDonald et al. (1996), it is reported that an SGTR is one of the most frequent initiating events experienced in a pressurized water reactor (PWR) type because of the existence of two circulation loops (i.e., primary and secondary circulation loops). The main functions of a steam generator (SG) in the PWR are twofold: (1) produce steam through heat exchange between the primary and secondary loops and (2) provide a physical boundary to confine radioactive materials generated from the core of the PWR. In order to fulfill these functions, the SG usually contains many thin tubes in it. It is evident that the use of thin tubes is effective from the perspective of the steam production. Unfortunately, due to the harsh operation environment of the SG (e.g., very high pressure and temperature), they are apt to be ruptured. This implies that the second function of the SG could be challenged because any defects in the SG tubes could be a direct pathway that allows the release of radioactive materials from the primary to the secondary circulation loop. Therefore, each NPP prepares a dedicated emergency operating procedure (EOP) that specifies detailed responses to cope with the SGTR (i.e., what should done and how to do it).

For example, Shin and Jae (2021) pointed out the catalog of important tasks that can significantly affect the progression of an SGTR. Table 1 exemplifies a part of the important tasks that require manual responses. In addition, Fig. 1 shows a part of EOP descriptions related to the first task of Table 1.

Table 1. Important manual tasks to cope with an SGTR

Task	Meaning
HPI	High pressure injection
MSGP	Maintain the broken SG pressure
SHR	Secondary heat removal by intact SG
RCSPCON	RCS (reactor coolant system) pressure control

Instructions	Contingency actions
4. **IF** pressurizer pressure less than 124.5 kg/cm^2, **THEN** verify SIAS* and CIAS** are automatically actuated.	4. **IF** SIAS and CIAS are **NOT** automatically initiated, **THEN** manually actuate SIAS and CIAS.

[1]SIAS Safety injection actuation signal [2]CIAS Containment isolation actuation signal

Figure 1. A part of an SGTR EOP related to the first task of Table 1

From the comparison between the important tasks and the contents of an SGTR EOP, it is possible to identify the list of manual responses to be carried out when an SGTR occurred. For example, two automatic responses and three manual responses can be distinguished from the comparison between Table 1 and Fig. 1.

- (Manual - Monitoring) Verify pressurizer pressure is greater than 124.5kgf/cm2
- (Automatic) Verify automatic actuation of SIAS
- (Automatic) Verify automatic actuation of CIAS (containment isolation actuation signal)
- (Manual - Control) Manually initiate SIAS
- (Manual - Control) Manually initiate CIAS

Here 'Manual – Monitoring' means a manual response that requires the verification of a specific component or process parameter while 'Manual – Control' denotes a manual response that actually manipulate a specific component. Based on these manual responses, a case study was conducted with respect to the stuck open of an MSSV followed by an SGTR.

3. CASE STUDY

3.1. Definition of the MSSV stuck open

The primary function of the MSSVs is to provide a pathway to relieve the high pressure of a main steam line with its cycles between open and closed positions by spring force. That is, if the pressure of a main steam line exceeds the spring force of an MSSV, then it will automatically open, and vice versa. Unfortunately, according to the operation experience of NPPs, numerous events pertaining to the failure of an MSSV have been reported over several decades [NRC, 1986; NRC, 1996]. For this reason, when it requires a series of cyclic operations, the stuck-open failure of an MSSV has been considered within the scope of traditional PSA. In the current case study, it is assumed that the stuck-open failure of an MSSV will occur if its cyclic operations are repeated over five times.

3.2. Determination of important manual responses

In order to extract important manual responses that are crucial for coping with an SGTR, the SGTR EOP of a reference NPP was analyzed in detail. As a result, a total of 79 manual responses (both monitoring and control responses) were identified. Table 2 shows a part of manual responses with their characteristics.

Table 2. A part of manual responses with their characteristics

ID	Task description	Manual response in the case study	Remark
1	Start Pump03	Start Pump03	Dichotomous control
2	Start Pump04	Stop Pump04	
3	Open CV-524 100%	Open CV-524 100%	Control with a specific value
4	Open CV-524 50%	Open CV-524 50%	
5	Maintain the pressurizer level between 28% to 38% using CV-240	**IF** (CV-240 valve position < 90%) **THEN** CV-240 valve position + 10% **ELSE** Open CV-240 100%	Adjust a specific component in order to maintain a target process parameter
6		**IF** (CV-240 valve position > 10%) **THEN** CV-240 valve position – 10% **ELSE** Close CV-240 0%	

From Table 2, the first and second manual responses correspond to dichotomous control with two fixed (or predetermined) states, such as Start/Stop and Open/Close. In contrast, the third and fourth manual responses are related to the setting the state of a target component into a specific value (e.g., 100% and 50%). Comparing with these four manual responses, the last two manual responses are somewhat complicated because the required task is to maintain a target process parameter by manipulating a specific component. For example, if the pressurizer level can be controlled by using CV-240 valve, human operators have to control CV-240 in order to maintain the pressurizer level between 28% and 38%. In this case, if the current value of the pressurizer level is 10% then it is necessary to repeat the fifth manual response several times until the pressurizer level is within 28% to 38%. This means that the performance of the fifth manual responses is usually combined with one or more manual responses related to monitoring (e.g., the openness of CV-240).

3.3. Variability of manual responses

It is natural to expect that the performance of each manual response should have a variability in terms of time and error. For example, for the manual response of ‘Start Pump03’, there could be a time distribution when human operators have to conduct it. In this regard, three levels of manual response timings are assumed in this case study based on the time distribution of manual responses obtained from a full-scope simulator of Korean domestic NPPs [Park et al., 2005; Park and Jung, 2007]. The three levels are Fast, Average, and Slow. Table 3 summarizes the meaning of each level with their values. In addition, the existence of omission error is considered to consider the variability of manual responses. For example, the omission error of ‘Start Pump03’ denotes the situation in which human operator did not start Pump03.

Table 3. Three levels of manual response timings

Level	Meaning	Value (s)
Fast	Human operators completed a manual response immediately.	0
Average	A manual response is conducted with an average speed.	22
Slow	A manual response is accomplished very carefully and meticulously.	94

It should be noted that the timing of each manual response was treated independently in this case study. At glace, it seems to be reasonable to assume that the manual response timing of human operators would be consistent during the performance of emergency operating procedures. That is, human operators who showed Fast response in the early phase of emergency situations will quickly conduct manual responses that are required at the later phase of emergency situations. However, in reality, it is more realistic to expect that the timing of manual responses is subject to be changed with respect to the nature of an emergency situation at hand. In other words, human operators are apt to try to quickly accomplish the required manual responses if they are urgent while human operators will carry out the required manual responses if they have to be completed with a meticulous manner. In order to address this characteristic of human operators, it is necessary to assign the level of manual response timings after evaluating the urgency of individual manual responses. Unfortunately, as a lot of additional efforts is indispensable to evaluate this urgency, the timing of individual manual responses was randomly assigned when they are activated during the progression of an accident scenario.

3.4. Analysis tool

In order to analyze the effect of manual responses on the occurrence of the MSSV stuck open followed by an SGTR, Nuclear Plant Analyzer (NPA) is used. The NPA was developed by Korea Institute of Nuclear Safety (KINS) in 2000. One of the significant advantages expected from the NPA is its calculation speed because it adopted a very simple TH model [KINS, 2000]. For this reason, the NPA can quickly generate diverse trends of key process parameters that are critical for determining the occurrence of the MSSV stuck open. Although it seems that the overall accuracy of NPA results could be degraded compared with precise but slow TH codes, many researchers reported that NPA results NPA are credible as well as sufficient for understanding the nature of diverse scenario progressions [Song et al., 2005; Yoon and Lee, 1992]. For this reason, a total of 14,348,907 simulation conditions that with randomly generated based on the variabilities of the 79 manual responses were analyzed by using the NPA.

3.4. Analysis results

Each condition randomly generated with the consideration of variabilities (i.e., the existence of omission errors and execution timings) is provided as an NPA input in order to observe the trends of the key process parameters. Since the stuck open of the MSSV connected to the ruptured SG will occur if the MSSV repeats its cyclic operations over five times, it is crucial to observe the pressure and level trends of the ruptured SG. Figure 2 shows the trends of several key process parameters obtained from the NPA analysis.

As can be seen from Fig. 2, the pressure of SG2 fluctuated (repeated increase and decrease in SG2 pressure), bounded by a specific setpoint. This trend indicates that the MSSV of SG2 started its cyclic operation in order to relieve pressure. Accordingly, since the number of cyclic operations exceeds five, it is expected that the stuck-open failure of the MSSV will occur. If we focus on the trend of SG2 level, it is possible to assume that the stuck-open failure of the MSSV causes an uncontrollable release of radioactive materials to the environment because the SG2 level remains near 100% when the cyclic operation of the MSSV begins. In this way, the results of all cases obtained from NPA were thoroughly investigated. Consequently, it is observed that 2,109,289 conditions (about 14.7%) are related to situations in which there is a chance for the release of radioactive materials to the environment.

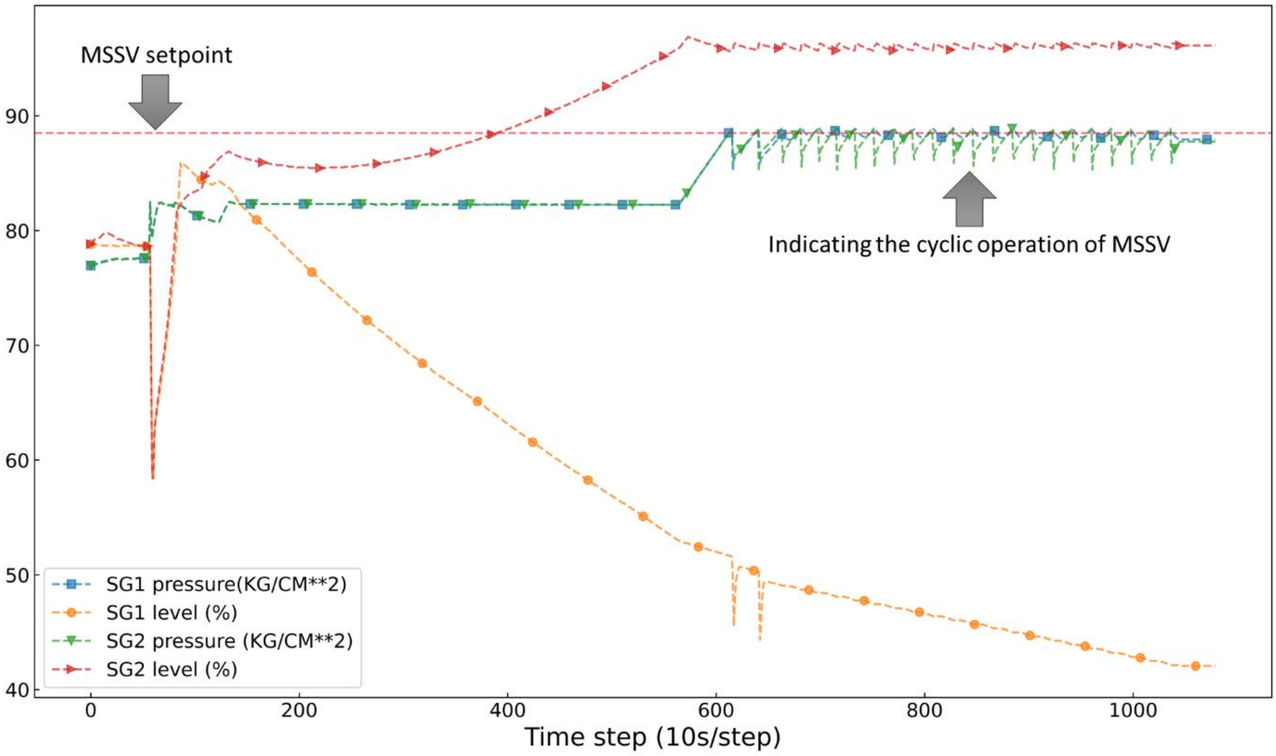

Figure 2. NPA analysis result: Example

4. DISCUSSION AND CONCLUSION

It is obvious that the identification of accident scenarios is crucial for enhancing the quality of PSA results. Unfortunately, complicated process control systems such as NPPs consist of many components that are operated through both automatic and manual responses. This implies that the search space of scenario progressions that could result in undesirable consequences is too large to exhaustively explore using a TH code that typically requires a couple of hours or more to get analysis results. For this reason, in current PSA, catalogs of accident scenarios are generally identified from a finite number of TH code runs with two assumptions: (1) accident scenarios can be represented by combining a limited number of automatic/manual responses that govern the status of safety-significant components, and (2) automatic/manual responses can be separately combined for determining scenario progressions because they are independent of each other.

The two abovementioned assumptions seem to be problematic because of the dynamic characteristics of NPPs. In short, a catalog of accident scenarios should be determined by analyzing a TH code run of each scenario progression that is changed by various kinds of interactions (or interdependencies) among the automatic/manual responses needed for dealing with the given initiating event [Maidana et al., 2023]. This implies that, in order to properly extract the accident scenarios with respect to a given initiating event, it is vital to handle a large number of scenario progressions by combining diverse automatic/manual responses with their interactions.

In order to corroborate this claim, as a case study, 79 manual responses that are supposed to be carried out following an SGTR occurrence in the reference plant were selected from the contents of the related EOP. From these responses, a total of 14,348,907 simulation conditions were generated based on combinations of plausible variabilities that are expected during the performance of the manual responses (i.e., the existence of omission errors with three levels of execution timing). In this light, it is possible to say that the identification of accident scenarios should be conducted in accordance with the consideration of diverse interactions among manual responses. Indeed, as a technical countermeasure for addressing this issue in the current PSA, dynamic PSA/PRA techniques (or simulation-based PSA/PRA) have been proposed by many researchers.

If so, it is inevitable to figure out a promising way to resolve the curse of TH code runs. In other words, although about 14.7% of the simulation conditions in this work could be regarded as novel accident scenarios that are worth spending additional resources on, it is unrealistic to run a slow TH code over a million times. Accordingly, regardless of whether current PSA or dynamic PSA is considered, the common technical challenge is to unravel the curse of TH code runs with a groundbreaking approach.

Recently, researchers have proposed a few interesting approaches [Park and Lee, 2022; Rahman et al., 2018; Suo et al., 2021], one of which is to use a reduced order model (ROM) based on up-to-date artificial intelligence (AI) techniques including deep learning [Maidana et al., 2023]. The basic idea behind combining an ROM with AI or deep learning techniques is to train a certain ROM using diverse conditions of automatic/manual responses with associated final states (e.g., CD or not) so that it can quickly distinguish the final state of scenario progression paths that are combinations of arbitrary conditions with respect to specific automatic/manual responses [Lee et al., 2020; Mandelli et al., 2017; Mandelli et al., 2021]. If we have an appropriate ROM that is able to distinguish the final state of a given scenario progression with a relatively high accuracy, it is strongly expected that the amount of computational resources required will be drastically reduced.

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References

Bogumil, R. J. (1982) *Limitations of Probabilistic Risk Assessment*. IEEE Technology and Society Magazine, 1(3), 24–28.
Korea Institute of Nuclear Safety (KINS). (2000) *Development of the Nuclear Plant Analyzer for Korean Standard Nuclear Power Plants*, KINS/HR-346, Daejeon, Republic of Korea
Lee, J. H., Yilmaz, A., Denning, R., Aldemir, T. (2020) *An online operator support tool for severe accident management in nuclear power plants using dynamic event trees and deep learning*, Annals of Nuclear Energy, 146, 107626.
MacDonald, P. E., Shah, V. N., Ward, L.W., Ellison, P. G. (1996) *Steam Generator Tube Failures,* NUREG/CR-6365, US Nuclear Regulatory Commission.
Maidana, R. G., Parhizkar, T., Gomola, A., Utne, I. B., Mosleh, A. (2023) *Supervised dynamic probabilistic risk assessment: Review and comparison of methods*, Reliability Engineering and System Safety, 230, 108889
Mandelli, D., Parisi, C., Alfonsi, A., Maljovec, D., Boring, R., Ewing, S., St. Sermain, S., Smith, C., Rabiti, C., Rasmussen, M. (2019) *Multi-unit dynamic PRA*, Reliability Engineering and System Safety, 185, p. 303-317.
Mosleh, A. (2014) *PRA: A perspective on strengths, current limitations, and possible improvements*, Nuclear Engineering and Technology, 46(1), p. 1-10.
Parhizkar, T., Utne, I. B., Vinnem, J. E., and Mosleh, A. (2021) *Dynamic probabilistic risk assessment of decision-making in emergencies for complex systems, case study: Dynamic positioning drilling unit,* Ocean Engineering, 237, 109653.
Park, J. and Jung, W. (2007) *OPERA – A human performance database under simulated emergencies of nuclear power plants*, Reliability Engineering and System Safety, 92(4), p. 503-519.
Park, J. and Kim, H., (2022). A case study to address the limitation of accident scenario identifications with respect to diverse manual responses. Under review in Reliability Engineering and System Safety.
Park, J. W., Lee, S. J. (2022) *Simulation optimization framework for dynamic probabilistic safety assessment*, Reliability Engineering and System Safety, 220, 108316.

Park, J., Jung, W., Kim, J., and Ha, J. (2005) *Analysis of human performance observed under simulated emergencies of nuclear power plants*, Korea Atomic Energy Research Institute, KAERI/TR-2895/2005, Daejeon, Republic of Korea.

Rahman, S., Karanki, D. R., Epiney, A., Wicaksono, D., Zerkak, O., Dang, V. N. (2018) *Deterministic sampling for propagating epistemic and aleatory uncertainty in dynamic event tree analysis*, Reliability Engineering and System Safety, 175, p. 62-78.

Shin, H., and Jae, M. (2021) *Realistic evaluation of source terms for steam generator tube rupture accidents*, Annals of Nuclear Energy, 164, 108559.

Song, I., Sohn, S., Huh, J., Lee, G., Sohn, J., and Kim, E. (2005) *Performance analysis of Westinghouse 3-loop plant with CENTS code*, Transactions of the Korean Nuclear Society Autumn Meeting, October 27-28, Busan, Republic of Korea.

Suo, W., Wang, L., Li, J. (2021). *Probabilistic risk assessment for interdependent critical infrastructures: A scenario-driven dynamic stochastic model*, Reliability Engineering and System Safety, 24, 107730.

Verma A. K. (2016) *Dynamic PSA*, In: Reliability and Safety Engineering (Eds: Verma, A. K., Ajit, S., and Karanki, D. R.), Springer Series in Reliability Engineering, p. 373-392.

Yoon, H. Y., and Lee, W. J. (1992) *Verification of a transient simulation code CENTS for all PWR design application*, Proceedings of the Korean Nuclear Society Spring Meeting, May 29-30, Kori, Republic of Korea.