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A Concept of Dynamic Emergency Operating Procedures Using Safety Margin Simulation

Jung Sung Kang, Seung Jun Lee*
Department of Nuclear Engineering, Ulsan National Institute of Science and Technology, 50 UNIST-gil, Ulju-gun, Ulsan, 44919, Republic of Korea
**Corresponding author: sjlee420@unist.ac.kr*

1. Introduction

A nuclear power plant is a facility where safety is very important like in the aviation and space fields. In the past, the device's defects in nuclear power plants were the main risk factors, but as the technology was advanced and experienced incidents caused by human errors, human errors were also accepted as a major risk factor.

Nuclear power plant operation is divided into three categories: normal, abnormal and emergency operation [1]. Emergency operation means operation in a status when a reactor trip is caused or necessary beyond the expected operating conditions, such as a loss of coolant accident (LOCA). In an emergency accident situation, variables change rapidly and numerous alarms occur in various ways, so an appropriate procedure design is very important to prevent operator error. The procedure used at this time is called the emergency operating procedure (EOP) and was initially used in the form of a paper-based procedure (PBP). PBPs cannot reflect all conditions of actual power plants due to the nature of the medium. This factor can also affect on potential factors for human error [2]. With the development of technology, the existing analog main control room (MCR) is changing to a digitalized MCR [3, 4], and the PBP is being developed in the form of a computer-based procedure (CBP). APR1400, standard nuclear power plant of the Republic of Korea, currently uses a CBP-type, computerized procedure system (CPS) [5]. CBP can share the progress of procedures between operators, and various support systems are being added to assist the operator. Currently, CBP reflects more real-time power plant data than PBP, but it is still limitedly applied. If it is compared with a map, it has evolved from paper maps of the past to computerized maps. If developed further, the concept of navigation can be extended to check the current state of the power plant and guide the optimal operation route. For this navigation system, two methodologies are required: a prediction method that predicts by reflecting the current state, and an evaluation method that can evaluate which route is better based on the predicted result. In recent years, prediction techniques using various neural network techniques have been proposed through the development of artificial intelligence technology. Technologies such as Long Short-Term Memory, Convolutional AutoEncoder, and Ensemble Quantile Recurrent Neural Network have been used to predict the state of the power plant [6-9].

2. Methods and Results

In this section, a brief structure of the EOP selected as a target, a description of the methodology, and the results of case studies through several scenarios based on the methodology.

2.1 Emergency Operating Procedure

The purpose of EOP is to mitigate the accident situation by diagnosing the emergency accident situation to the operator and suggesting appropriate measures according to the diagnosis result. The EOPs are utilized to protect critical safety functions (CSFs) and to prevent the core damage [1]. The core elements of EOP are composed of the following four [1].

- Immediate actions and diagnostic procedures
- Event related symptom based optimal recovery guidelines (ORGs)
- CSF restoration guidelines
- CSF status trees

If it is shown schematically, it is like the figure below.

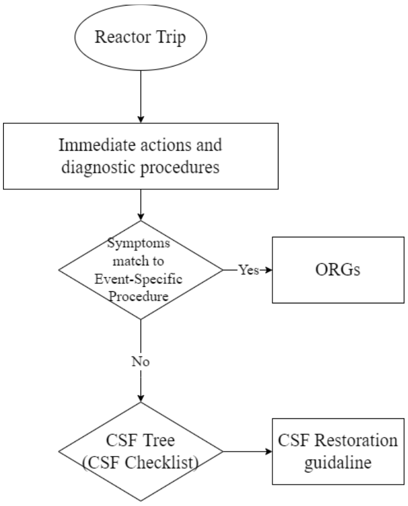

Fig. 1. The EOP Flowchart

Previously, an emergency initial operation support system that replaces the immediate actions procedure using multilevel flow modeling and performs the diagnosis procedure using a gated recurrent unit has been proposed [10, 11]. This paper proposes a methodology to cover the ORG and CSF restoration guideline areas as an extended version of the system.

2.2 Framework

The key to this methodology is to calculate the safety margin area and evaluate which driving is the optimal driving based on this result. The safety margin area can be calculated by obtaining the residual of the standard variable value (L) and the current variable value (P) of the key variable, and using the residual and the area of the unit time. The detailed formula is as below.

$$A = \int_0^{t_{end}} \{L_{key\_var} - P_{key\_var}(t)\} dt \qquad (1)$$

$$S = {}^{A}/_{t_{end}} \qquad (2)$$

$$S_{Total} = \sum W_{CSF_i} \cdot S_{CSF_i} \qquad (3)$$

The optimal operation is derived based on the assumption that the wider the safety margin area, the safer the operation. The safety margin area is calculated by using the standard value of the critical safety function key variables and the current value of key variables. The safety margin score is calculated based on safety margin area per time. Since the safety margin score requires evaluation of many variables, each score is calculated based on the critical safety functions. The total score is derived as the sum of the value obtained by multiplying the safety margin score for each CSF by weight. To calculate this, we need to derive key variables and key tasks. The detailed process is shown in fig 2.

After the key variable selection is finished and the main task is selected, the safety margin can be calculated based on this. Fig 3 below is an example of calculating the safety margin based on the core exit temperature, which is a key variable for core cooling.

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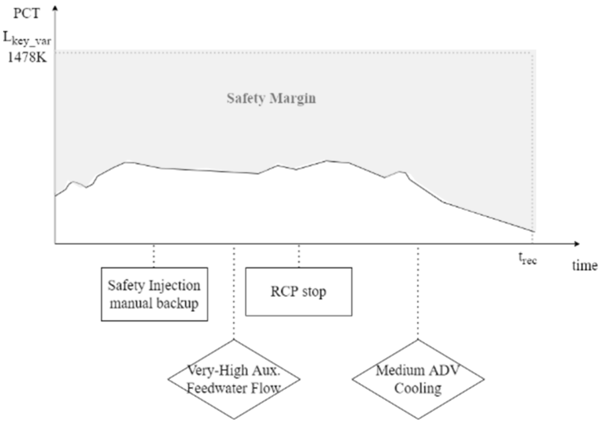

Fig. 3. CSF safety margin example (Core cooling)

Key variables for each critical safety function and major tasks are derived from the procedure analysis. The standard value of the key variable can be classified into three categories: maintaining the upper limit, maintaining the lower limit, or maintaining a limited area. The major tasks can be classified into two types: a single performing tasks and a continuous tasks. Among continuing tasks, control tasks are classified according to the level of control (ex. high, medium, low flow rate). Those key varables and major tasks are shown in table 1 and 2

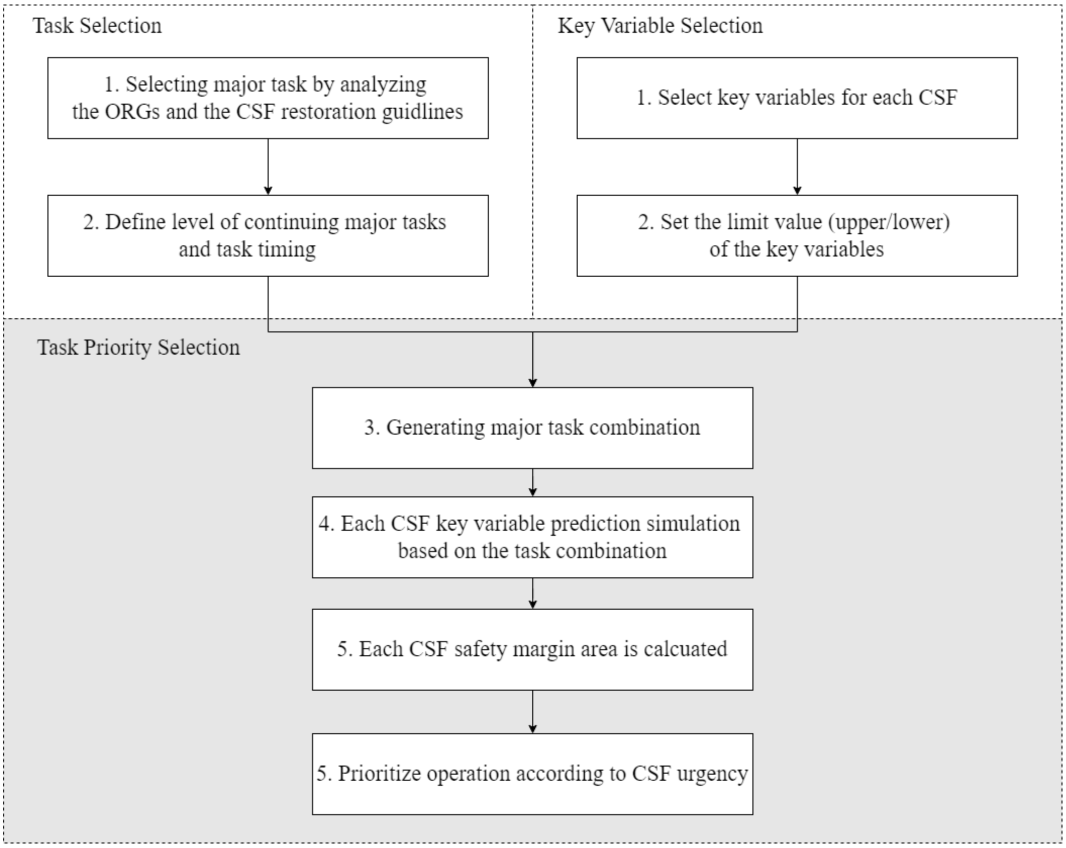

Fig. 2. Safety margin calculation process

Table I: CSF Key Variable

CSF	Key variable 1	Key variable 2
Subcriticality	Reactivity	-
RCS inventory	Pressurizer level	Pressurizer pressure
Core cooling	Core exit temperature	Peak cladding temperature
Heatsink	SG pressure	SG wide level
RCS integrity	Cooling rate	-
Containment integrity	Containment temperature	Containment pressure

Table II: Major Task Classification Example

Single performing tasks	Reactor coolant pump (RCP) stop
	Safety injection tank pressure setting value control
	Engineering safety feature actuation signal manual back-up
	Mains steam isolation signal setting value control
	Isolation of the failed steam generator
Continuing tasks	Safety injection flowrate control
	Pressurizer pressure control
	Cooling rate control using main steam line (atmospheric dump valve (ADV) control)
	Auxiliary feed water flow rate control

2.3 Case Study Result

To confirm this methodology, a case study was performed for several scenarios using the Compact Nuclear Simulator (CNS) developed by the Korea Atomic Energy Research Institute [12].

The first scenario is an RCP stop task in a LOCA situation. LOCA is cold-leg #1 fracture, and the size was given as 25cm2. Reactor Coolant Pump (RCP) stop tasks should be conducted when PRZ pressure is under 97kg/cm2, and at least 1 safety injection pump activated.

If RCP is stopped early, core cooling is degraded, but it can prevent aggravation of RCS inventory loss in LOCA situation. If LOCA occurs with a breaking size of 25cm2 in the first cold-leg, the RPC stop condition is satisfied in 45 seconds. The test compared the case where there was no RCP stop and the case where the RCP stopped at 1 minute. To compare this, the core outlet temperature and reactor vessel water level related to core cooling in CSF were measured. The variable comparison result is shown in fig 4 and 5. The safety margin score comparison can be found in table 3.

The RCP Stop task degrades core cooling but enhance reactor coolant system (RCS) inventory. The RCS inventory enhancement is larger than core cooling degradation. So the operator can make a decision making the RCP stop task is required to conduct.

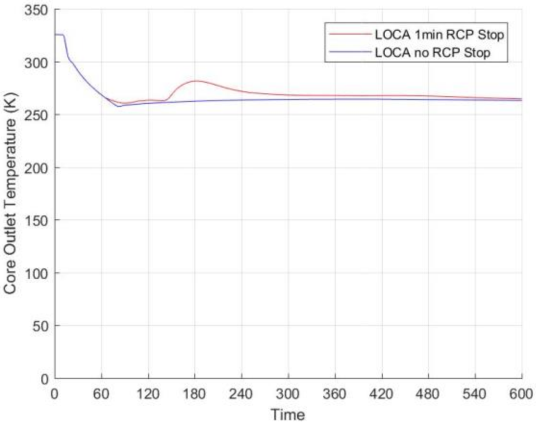

Fig. 4. Core Outlet Temperature Comparison (LOCA RCP scenario)

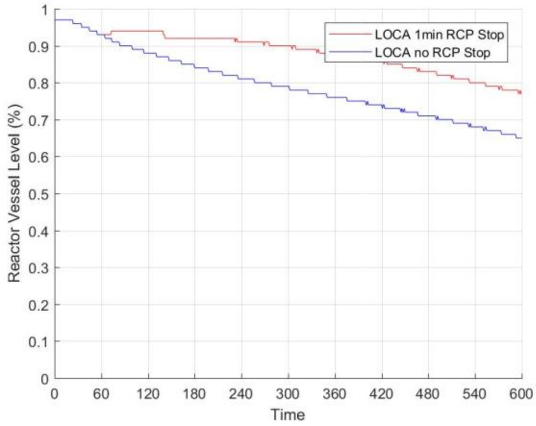

Fig. 5. Reactor Vessel Water Level Comparison (LOCA RCP Scenario)

Table III: Safety Margin Score Comparison (LOCA RCP Scenario)

Category	1min RCP Stop	No RCP Stop
Core Outlet Temp	729.12	**733.71**
Reactor Vessel Water Level	**0.884**	0.795

The second scenario is the steam generator (SG) isolation during a steam generator tube rupture (SGTR) accident. In the event of the SGTR accident, the faulty SG must be isolated. If the steam generator is quickly isolated, the secondary radiation leakage will be reduced, but the integrity of the steam generator may be compromised. In addition, since the isolated SG cannot participate in core cooling, the core cooling performance is also deteriorated. In case 2, a break of 60 cm^2 occurred in SG #1, and the SG isolation time was compared according to the early isolation time (30 seconds) and the general isolation time (90 seconds).

Table IV: Safety Margin Score Comparison (SGTR SG Isolation Scenario)

Category	Early SG Isolation (30s)	Normal SG Isolation (60s)
AVG Temp Score	730.6	**738.8**
SG1 Pressure Score	8.47	**9.69**
Secondary Radiation	**0.0083**	0.0069

If the SG is isolated early, it can be confirmed that the safety margin is lower than that of normal isolation because only two out of three steam generators are used. In addition, it can be seen that the pressure safety margin of SG also decreases due to the overpressure of SG1. On the other hand, for secondary radiation, it is confirmed that early isolation has a greater safety margin.

3. Conclusions

Although the current procedure has reached the stage of computerization, it has many characteristics of a paper procedure. The safety margin methodology that can provide risk information by reflecting the current status like navigation was proposed. This method focuses on transforming the prediction result into a form that is helpful for decision-making under the premise that power plant prediction is possible. To calculate the safety margin, key variables and major tasks were derived for each CSF from the procedure analysis. The score is defined with the safety margin divided by time, and it was assumed that the higher the score, the safer the operation. Scores for RCP stop and SG early isolation were compared for each CSF key variables. In the future, it is planned to perform an optimal operating evaluation by comparing safety margin scores for multi-task combinations.

4. Acknowledgement

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REFERENCES

[1] Agency IAE. Development and review of plant specific emergency operating procedures: Internat. Atomic Energy Agency; 2006.
[2] Oxstrand J, Le Blanc K. Computer-Based Procedures for Field Workers in Nuclear Power Plants: Development of a Model of Procedure Usage and Identification of Requirements. Idaho National Laboratory External Report. 2012.
[3] O'Hara JM, Hall RE. Advanced control rooms and crew performance issues: implications for human reliability. IEEE transactions on Nuclear Science. 1992;39:919-23.
[4] Park J, Lee D, Jung W, Kim J. An experimental investigation on relationship between PSFs and operator performances in the digital main control room. Annals of Nuclear Energy. 2017;101:58-68.
[5] Hong J-H, Lee M-S, Hwang D-H. Computerized procedure system for the APR1400 simulator. Nuclear Engineering and Design. 2009;239:3092-104.
[6] Bae J, Kim G, Lee SJ. Real-time prediction of nuclear power plant parameter trends following operator actions. Expert Systems with Applications. 2021;186:115848.
[7] Kim H, Arigi AM, Kim J. Development of a diagnostic algorithm for abnormal situations using long short-term memory and variational autoencoder. Annals of Nuclear Energy. 2021;153:108077.
[8] Kim H, Cho J, Park J. Application of a deep learning technique to the development of a fast accident scenario identifier. IEEE Access. 2020;8:177363-73.
[9] Ryu S, Kim H, Kim SG, Jin K, Cho J, Park J. Probabilistic deep learning model as a tool for supporting the fast simulation of a thermal–hydraulic code. Expert Systems with Applications. 2022;200:116966.
[10] Kang JS, Lee SJ. Concept of an intelligent operator support system for initial emergency responses in nuclear power plants. Nuclear Engineering and Technology. 2022.
[11] Choi J, Lee SJ. A sensor fault-tolerant accident diagnosis system. Sensors. 2020;20:5839.
[12] Kwon K-C, Park J-C, Jung C-H, Lee J-S, Kim J-Y. Compact nuclear simulator and its upgrade plan. 1997.