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Development of Long-term Safety Assessment Methodology for Multi-Module SMR by Using Simplified Model of Flooding Safety System

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1. Introduction

Small modular reactors (SMRs) have been highlighted with global policy aiming to achieve carbon neutrality by 2050. The most feasible SMRs share the form of integrated pressurized water reactor (IPWR), which integrates major components such as core, steam generators (SGs), pressurizer, and reactor coolant pumps (RCPs) into a reactor vessel thanks to the extensive experiences on development and operation of the large PWR reactors. However, the possibility of radioactive material release by core damage still remains although the power generation of IPWR-type SMRs is smaller than 300 MWe. Accordingly, proper safety systems are required to ensure safety even under extreme conditions.

Simplified system design of the SMRs is advantageous to install passive safety systems (PSS) adopting phase-change and naturally driven flow by gravity and pressure difference. In addition, the decay heat of the SMRs, which have to be removed during an accident, is also evaluated much lower than large reactor's as the thermal power is much lower. Accordingly, in most SMR design concepts, the generally proposed coping time of 3 days is ensured by using PSS even without any alternating current (AC) power sources and active systems. Furthermore, advanced concepts of the PSS for the SMRs, such as indefinite residual heat removal system (PRHRS) and flooding safety system (FSS), can achieve indefinite grace period with certain conditions [1, 2]. The innovative concepts securing indefinite grace period are not only remarkable engineering achievements but also powerful selling points in terms of nuclear acceptance.

The PSS concept whose grace periods were evaluated to be largely enhanced requires additional systems for sustaining cooling capability of emergency core cooling system (ECCS) and PRHRS. To assess the feasibility of the PSS, sufficient analyses are required to be carried out with specified modelling of target reactor and PSS by using safety assessment code such as MARS-KS code. However, the detailed analyses take a long time as utilizing the reactor input models with many nodes. Moreover, the time step needs to be sufficiently short owing to the frequent phase changes in the PSS.

The assessment of 72 hours was sufficient for the indefinite PRHRS because the water level in the emergency cooldown tank (ECT) was sustained, and air-cooling capability exceeded the decay heat within 72 hours [1]. However, the calculation time of 72 hours is insufficient to assess the grace period of the FSS. The FSS conditionally ensures indefinite core cooling. In other words, the cooling capability is not permanently sustainable but is largely enhanced in more practical cases. In addition, a certain concept of the collection ratio which is one of the most important factors is difficult to be adopted in the safety analysis code without modification of the source code. As the shortest grace period by using the FSS is expected to exceed a month, the long-term safety assessment by using the safety analysis codes such as MARS-KS code requires too much computation cost. The computation cost is expected to increase as the detailed design values are reflected in the input model.

Although the long-term analysis of more than a month is unrequired during the licensing process, the developers and researchers still need a method for long-term safety assessment in terms of achieving selling points and engineering completion. Thus, the objectives of this study are to propose methodology for long-term safety assessment that the PSS is expected to secure the grace period longer than a month and to adopt the proposed methodology for the FSS. To confirm the feasibility of the methodology, cross-verification between the calculations by using MARS-KS code and in-house code.

2. Methodology for long-term safety assessment

2.1 Reference reactor and passive safety system

The representative IPWR-type SMR, VOYGR^{TM}, proposed a PSS which ensures indefinite cooling capability as the ECCS and decay heat removal system (DHRS) passively operates during an accident for the IPWR-type SMRs employing metal containment vessel (MCV) [3]. However, several disadvantages such as difficult approach for management, more heat loss, and large inventory of the common pool for high thermal power owing to its design concepts such as permanently submerged reactor modules in a common pool (CP) even during normal operation. Accordingly, the FSS was developed to improve the operator convenience and long-term safety of multiple reactor modules configuration in a plant building [2]. As shown in Fig. 1, the FSS consists of a CP, 6 auxiliary pools (APs) which play a role of the ECT of the PRHRS, and 6 separate dry cavities (SDCs). The emergency coolant stored in the CP during normal operation is supplied into the SDC during an accident. To sustain emergency coolant supply capability from the CP to the SDCs, the air-cooled condenser installed on the ceiling of the plant building condenses the boiled-off steam on the MCV and the CP receives the condensate as a sump. More details are described in Ref. [2].

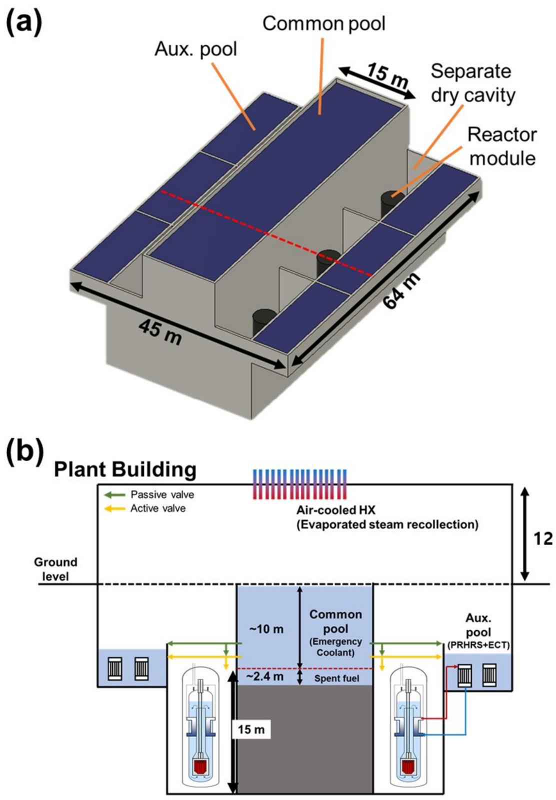

Figure 1. Flooding safety system configuration: (a) 3-dimensional drawing, (b) cross-sectional drawing by the red line in (a) [2].

To assess the long-term cooling sustainability of the FSS, an Autonomous Transportable On-demand Reactor Module (ATOM) was selected as the reference reactor module [4]. The thermal power and operation period refueling cycle were designed to be 450 MWt (150 MWe) and 2 years, respectively. The major parameters of the reference systems are summarized in Table 1.

Table 1. Major parameters of ATOM reactor module and flooding safety system for long-term safety assessment.

ATOM reactor module	
Height of reactor module	15 m
Diameter of metal containment	4.5 m
Thermal/electric power per reactor module	450 MWt / 150 MWe
Operation period for refueling cycle	2 years
Number of reactor modules in a plant building	6
Flooding safety system	
Cross-sectional areas of entire system/common pool	2,880 m^2 / 960 m^2
Volume of common pool	11,486 m^3
Water level/mass for emergency coolant	10 m / 9,016 t
Volume of a dry separate cavity	1,568 m^3
Heat transfer area of air-cooled condenser	260 m^2
Wall temperature of air-cooled condenser	370.00 K (960.85 °C)
Free volume in a plant building	50,000 m^3

2.2 Long-term safety assessment methodology

To confirm the feasibility of the PSS concept, system analysis codes such as MARS-KS for Design Basis Accident (DBA) and MELCOR for severe accident were used. The reference systems, ATOM and FSS, can be modelled with the detailed design values as shown in Fig. 2(a) for MARS-KS and MELCOR calculations. In the previous study on the FSS concept, passive supply of the emergency coolant from the CP could fill the SDCs before the pressure in the MCV exceeds the design pressure and core damage [5]. Accordingly, the FSS concept was deemed to be feasible. However, the detailed analysis of the PSS with the reactor module required a long time owing to many nodes in the input models. In addition, the vigorous boiling and condensation occurred in both plant building and connected system of RPV and MCV (RPV-MCV system). The time steps of the calculation became intensely short to satisfy the conservation during the phase changes. Accordingly, in terms of computation cost, the detailed analysis is more adequate for transient simulation during early phase of an accident and analyses whose calculation times are less than 72 hours. To reduce the computation cost with rationality, the simplified model and comparison with the detailed model were proposed.

From the detailed analysis, the heat transfer from the RPV-MCV system to the emergency coolant in the SDC was evaluated smaller than the decay heat from the reactor core owing to heat capacity of the MCV. In other words, the assumption that heat transferred from the RPV-MCV system is simplified by direct transfer of the decay heat is deemed conservative. In this study, instead of cross-verification with detailed analysis, the simplified model with the conservative assumption was adopted. The decay heat was calculated by Eq. (1) and data with time was adopted in MARS-KS calculation [6].

$$q_d(t) = 0.0622 q_o [t^{-0.2} - (t_o + t)^{-0.2}] \quad (1)$$

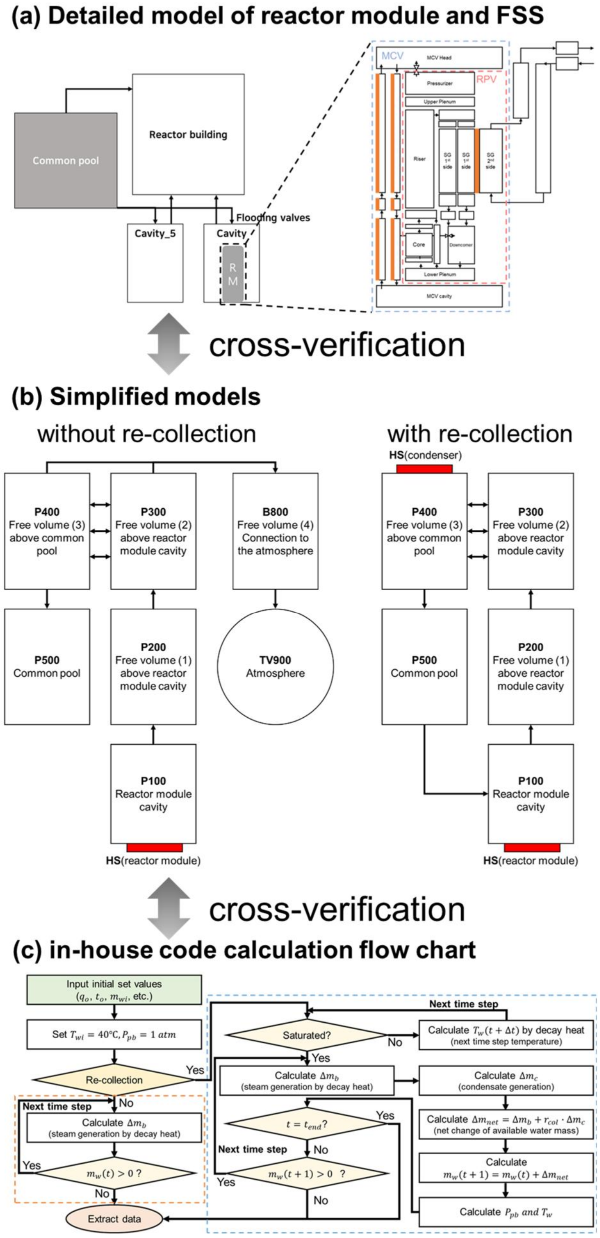

Figure 2. Cross-verifications for simplified models and in-house code (a) 3 days analysis for detailed model, (b) 1-month analysis for simplified model, (c) in-house code calculation flow chart for grace period longer than a month [2, 5].

By using the simplified model, the computation load can be significantly reduced. However, faster calculation method is still required as the FSS can secure a long grace period more than 30 days [2, 7]. In addition, as shown in Fig. 2(a), the boiled-off steam is released to the atmosphere or the condensate of the steam is completely collected to the CP in the cases without and with re-collection strategy, respectively. In more practical cases, the condensate can be stagnated somewhere in the plant building. In other words, the full re-collection of the steam is difficult to be realized. Accordingly, we defined a factor, collection ratio, which is the ratio between the collected condensate mass to the total condensate mass as expressed in Eq. (2). Because the certain definition of the collection ratio cannot be directly adopted to MARS-KS calculation without source code modification, the calculation algorithm described in Fig. 2(c) was developed and implemented by using MATLAB code.

$$r_{col} = \frac{m_{col}}{m_{con}} \quad (2)$$

To compare the results by MARS-KS and in-house code, the cases without and with re-collection strategy were evaluated by using both codes. Owing to difficult adoption of the certain collection ratio from 0 to 1 in the MARS-KS code, the condensate generated on the air-cooled condenser was assumed to be completely collected into the CP.

3. Results and Discussions

The objective of the in-house code was to reduce the computation cost with small relative error with MARS-KS analysis. The in-house code adopted several engineering assumptions. In the case without re-collection, the pressure in the plant building was assumed to maintain 1 atm as the steam was released to the atmosphere. In addition, the pressure by water level was neglected. Accordingly, boiling started when the emergency coolant in the SDCs reached the saturation temperature of 100 °C at 1 atm. However, the MARS-KS code considered the difference as pipes (Ps) were divided as multiple volumes in the nodalization shown in Fig. 2(b). Consequently, as shown in Fig. 3, the boiling start time and time to total depletion of the emergency coolant was evaluated earlier in the result of the in-house code than MARS-KS. However, both calculations showed that the FSS can achieve the grace period more than 40 days even without re-collection strategy.

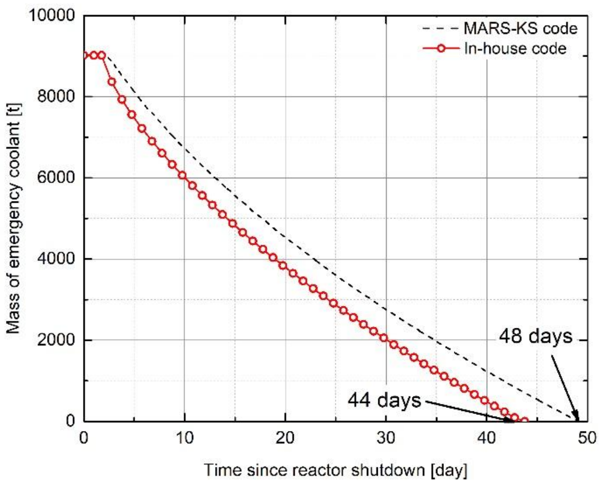

Figure 3. Comparison of the results calculated by MARS-KS code and in-house code without re-collection [2].

For the case with re-collection strategy, calculation time limit was set as 15 days because full re-collection was expected to secure indefinite grace period. The steam was assumed to be uniformly distributed in the plant building. However, the MARS-KS computed specified steam fraction in each volume in the plant building. In addition, to evaluate condensation heat transfer, the in-house code adopted Eq. (3) which is similarly expressed but different from the MARS-KS model and correlation [8, 9]. In addition, the same differences of the case without re-collection also existed. Consequently, as shown in Fig. 4, the results of the mass of available emergency coolant showed slight difference. However, both minimum values exceeded 99.15% of initial mass and the difference was less than 0.35%. This showed that the FSS can be deemed as a feasible PSS for IPWR-SMRs employing the MCV.

$$h_{co} = 0.185 D^{\frac{2}{3}}(\rho_{wall} + \rho_\infty)\left(\frac{\rho_{wall}+\rho_\infty}{\mu}\right)^{\frac{1}{3}} \ln\left(\frac{1-W_{s,wall}}{1-W_{s,\infty}}\right)\frac{h_{fg}}{T_\infty - T_{wall}} \quad (3)$$

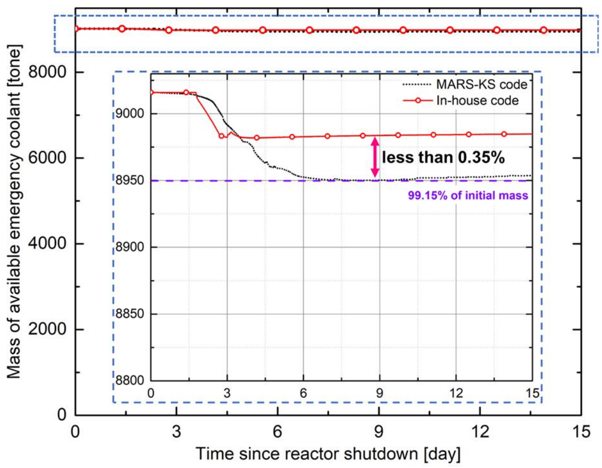

Figure 4. Comparison of the results calculated by MARS-KS code and in-house code with re-collection [2].

4. Summary and Conclusion

In this study, the methodology to assess the long-term cooling capability of the PSS which secure the grace period more than a month was proposed and the results of MARS-KS and in-house code calculations for the FSS were compared. From both calculation results of MARS-KS and in-house codes without and with re-collection, the feasibility of the FSS concept was confirmed as the grace period more than a month and increasing mass of available emergency coolant was observed. According to the engineering assumptions of the in-house code, the results in comparison with MARS-KS showed slight differences. To construct long-term safety assessment for grace period longer than a month, the in-house code will be improved based on the physical phenomena in the plant building as future works. Furthermore, the simplified input model for the MARS-KS will be compared with more detailed model with the reactor module.

Acknowledgments

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