Major Regulatory Review Considerations on Safety Analysis of Small Modular Reactors

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1. Introduction

Various types of Small Modular Reactors (SMRs) are currently under development or have been completed both domestically and internationally. In Korea, the innovative light-water-based SMR known as i-SMR has been under development since 2021.

SMRs often adopt integrated designs and passive safety systems, which differ from large commercial reactors, necessitating different regulatory requirements and guidelines for safety analysis. Therefore, it is necessary to preemptively present applicable safety analysis-related regulatory requirements and guidelines for SMRs to ensure consistency in SMR safety regulation and to encourage applicants to prepare accordingly. First, the present study introduces an applicable regulatory requirement for the safety analysis of Small-Break Loss-Of-Coolant Accident (SB LOCA) of SMRs, and then, proposes safety regulatory guidelines applicable to develop Critical Heat Flux (CHF) correlations and to conduct related experiments to meet the identified SB LOCA requirement. Next, an safety issue caused by the adoption of passive safety systems, compared to active safety systems, is identified. And finally, the new conservative safety analysis methodology to address this issue is suggested as a regulatory guideline for conservative analyses of Design Basis Accidents (DBAs) and Anticipated Operational Occurrences (AOOs) involving passive safety systems.

2. Acceptance criteria for SB LOCA

For SMRs with an integral structure, large-diameter piping is excluded from their design stage, and therefore a Large-Break Loss of Coolant Accident (LB LOCA) is not considered as a DBA for SMRs. Only SB LOCA is treated as LOCA DBA.

In the case of SB LOCA unlike LB LOCA, SMRs core may remain submerged with coolant. As a result, acceptance criteria developed on the basis of LB LOCA—intended to protect the core against complete core uncovery and subsequent high-temperature oxidation due to steam—are not directly appropriate for SB LOCA in SMRs.

For example, in NuScale's case, considering these characteristics of SMR SB LOCA, the acceptance criteria (Figures of Merit: FOM) for SB LOCA were revised from the Peak Cladding Temperature (PCT) and other requirements under 10CFR50.46(b)(1)–(b)(4) to [1]:

① collapsed liquid water level in the core,
② critical heat flux ratio, and
③ containment pressure and temperature.

The background, especially NuScale applied the Critical Heat Flux Ratio (CHFR) as a new LOCA acceptance criterion is that both of Standard Review Plan (SRP)[2] and the Design Specific Review Standard (DSRS)[3] for NuScale require such acceptance criterion for LOCA of new reactor types. For example, section 15.6.5 of the DSRS for NuScale reads like below.

If core uncovery is not expected during the entire period of a LOCA, the staff should ensure that a significant number of fuel rods will not be damaged because of local dryout conditions. This may be demonstrated by showing that the limiting fuel rod heat flux remains below the critical heat flux (CHF) at a given pressure after depressurization has taken place. If, however, the heat flux exceeds the CHF, further analyses should be performed to estimate the amount of fuel damage expected from "burn-out" while the bulk of the core remains covered with water during the LOCA. Fuel damage and potential for radioactivity release to the environment must be consistent with 10 CFR Part 100. If such evaluations are not provided in the applicant's technical submittal, the reviewer requests that they be made.

Almost the same context is also described in the Korean Safety Review Guideline (SRG) for PWR in section 15.6.5.[4] On top of that, the US NRC clearly indicated in their Safety Evaluation Report that the application of the acceptance criteria from section 15.6.5 of the SRP and the DSRS for NuScale was made when NuScale LOCA was reviewed.[5]

From these observations, it is recommended that SMR should adopt the same acceptance criteria as NuScale's especially the CHFR for SB LOCA.

3. Regulatory guidelines for CHF

As described in the preceding section, the CHFR must be newly considered as an acceptance criterion for SB LOCA, in order to adequately reflect the unique design characteristics of SMR into the safety analysis. Moreover, the CHFR also serves as part of the acceptance criteria for non-LOCAs. Therefore, for the safety analysis of SMR, an accurate evaluation of the CHF is of paramount importance. Such evaluation—including the application and development of CHF correlation—must account for the distinct thermal-hydraulic characteristics inherent to SMR, which differ fundamentally from those of large commercial reactors. From these perspectives, some regulatory guidelines for CHF are proposed in this section, by incorporating unique thermal-hydraulic features of SMRs.

3.1 Use of a specific CHF correlation under low-pressure and low-flow condition

Globally, the majority of CHF correlations for large commercial reactors cover core pressure and core mass flux ranges of approximately 10.0–17.5 MPa and 1,695–4,747 kg/m²·s, respectively—commonly referred to as high pressure high flow conditions [6]. For instance, in the case of the domestically developed APR1400 reactor having the normal operating conditions of the core pressure of 15 MPa and the core mass flux of 3,500 kg/m²·s uses the CHF correlation (KCE-1) covering 9.61–16.65 MPa and 1,152.7–4,277.7 kg/m²·s. Although there are some special accidents such as LB LOCA and Steam Line Break (SLB) where the core may reach the low pressure low flow condition (≤10.0Mpa, ≤1,000 kg/m²·s), the need to use a specific CHF correlation tailored to the low pressure low flow is not that high [7] and as a result, a single CHF correlation covering the high-pressure high-flow seems enough for large commercial reactors.

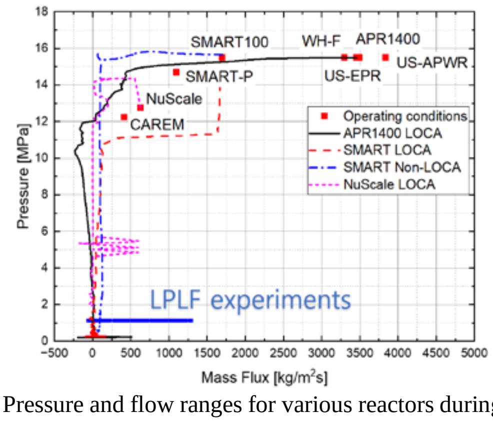


Fig. 1. Pressure and flow ranges for various reactors during the normal operation and the LOCA

However, this is not the case for SMRs. For SMRs, it is essential to employ a specific CHF correlation applicable to low-pressure low-flow conditions in order to verify their compliance with radiological acceptance criteria under DBAs (SB LOCA, SLB) and an AOO (Inappropriate Actuation of the Emergency Core Cooling System) all of which turn into the low pressure low flow core condition. Specifically, these criteria require that, in the case of the DBAs, limited fuel damage is permissible, whereas for the AOO, fuel damage must be prevented. Consequently, the use of a specific CHF correlation tailored to low pressure low flow condition for specific DBAs and AOO as well as the use of a high pressure high flow CHF correlation is mandatory in SMRs

3.2 Non-conservatism of CHF Look-Up Table under low-pressure and low-flow condition

At present, the most widely used CHF correlation worldwide is the Groeneveld Look-Up Table (LUT) 2006 [8]. The Groeneveld LUT was developed by compiling a vast database of CHF experimental measurements spanning pressures of 0.1–20 MPa, mass fluxes of 0–8,000 kg/m²·s, and qualities ranging from –0.5 to 1.0. Based on these datasets, the LUT provides interpolated CHF values for given conditions of pressure, mass flux, and quality within the specified ranges. The Groeneveld LUT is incorporated into various thermal-hydraulic system codes, including MARS-KS, and is regarded as the most general-purpose CHF correlation because it encompasses the broadest pressure–mass flux–quality domain. In cases where a reactor-specific CHF correlation covering the relevant thermal-hydraulic regime is not available, it is most frequently employed as a substitute.

To evaluate whether the Groeneveld LUT implemented in MARS-KS is applicable under low-pressure low-flow conditions, representative CHF experimental data obtained under such conditions [9] were used to assess the predictive performance of the code. The results [10] indicate that, under low-pressure low-flow conditions, the Groeneveld LUT significantly overpredicts CHF—by approximately 50% compared to experimentally measured values. Therefore, this finding highlights that CHF correlations suitable for the low-pressure low-flow regimes relevant to SMR safety analysis must be developed through independent experiments tailored to the thermal-hydraulic operating range of SMRs. Furthermore, it demonstrates that the use of the general-purpose Groeneveld LUT under such conditions may lead to non-conservative predictions of CHF values.

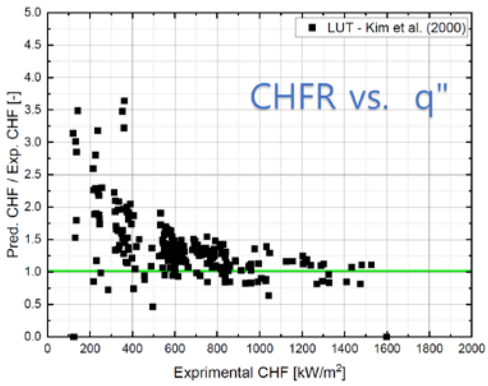


Fig. 2. CHF ratio (LUT prediction/Measurement)

3.3 Consideration on premature CHF mechanism due to two-phase instability

At low-pressure conditions, the density difference between water and steam increases, resulting in a higher slip ratio and larger bubble sizes, which in turn enhances the role of buoyancy. In addition, low-flow conditions further intensify the relative influence of buoyancy. As a result, under low-pressure and low-flow conditions, the likelihood of two-phase flow instability increases.

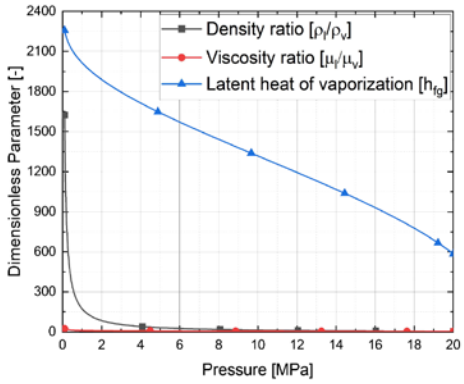


Fig. 3. Water Property Change due to Pressure

Representative two-phase flow instability mechanisms that can occur under low-pressure and low-flow conditions include (i) Ledinegg instability, (ii) flow pattern transition instability, (iii) density wave oscillation, and (iv) pressure drop oscillation [11,12]. Among these, mechanisms (i) and (ii) are known to cause flow excursion, while mechanisms (iii) and (iv) are known to induce flow oscillation. When such two-phase flow instabilities occur, the influence of unstable flow (flow excursion or flow oscillation) can lead to the premature occurrence of CHF at levels significantly lower than the CHF measured under stable conditions (Premature CHF phenomenon).

Yang et al. [6,13] argued that, under low-pressure conditions below 7 MPa, the critical heat flux is highly susceptible to two-phase flow instabilities, with the Ledinegg instability mechanism being the most dominant instability mechanism. In addition, since most SMRs adopt passive safety systems that rely on natural circulation mechanisms with two-phase flow, they are more prone to flow instabilities compared to active safety systems employing forced circulation by pumps, due to their higher nonlinearity and lower driving force [12].

Therefore, for SMRs, it is essential to evaluate the occurrence of natural circulation flow instabilities under low-pressure and low-flow conditions. If such instabilities are confirmed, the CHF correlations established under stable flow conditions for low-pressure and low-flow regimes must be modified to account for the premature CHF phenomenon induced by flow instabilities (i.e., by reducing the CHF value).

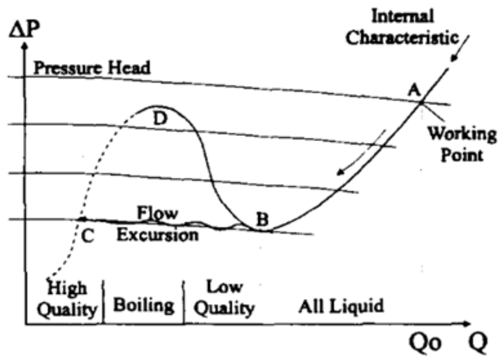


Fig.4. Ledinegg Instability Mechanism

3.4 Incorporation of the mixing vane effect on CHF under low-pressure and low-flow conditions

Under high-pressure high-flow conditions, the mixing vanes incorporated in fuel assemblies generally act to enhance the CHF. Accordingly, CHF correlations for high-pressure and high-flow conditions in large commercial reactors include a correction term that accounts for the CHF enhancement effect of mixing vanes. However, under low-pressure and low-flow conditions, mixing vanes do not always enhance the critical heat flux and, in some cases, may even reduce it [6,14]. The reduction in CHF observed with the installation of mixing vanes under such conditions has been attributed to the channel blockage effect caused by the excessive pressure drop induced by the mixing vanes [6].

Therefore, under low-pressure and low-flow conditions, the CHF correction term associated with mixing vanes should be determined separately based on CHF experimental data obtained under such conditions, and applied independently to the CHF correlations for low-pressure and low-flow regimes

3.5 Verification of the possibility of CHF occurrence due to the homogeneous nucleation mechanism

According to Liu et al. [15], under subcooled flow boiling conditions, there exist two distinct mechanisms responsible for the occurrence of CHF: one that follows the typical flow pattern and another that follows the homogeneous nucleation flow pattern. Figure 5 illustrates the CHF occurrence scenarios corresponding to these two mechanisms. In the mechanism following the typical flow pattern, bubbles generated after Net Vapor Generation (NVG) coalesce, leading to CHF occurrence. In contrast, in the homogeneous nucleation flow pattern, CHF occurs as micro-sized bubbles produced by homogeneous nucleation coalesce. In most cases, CHF is induced by the typical flow pattern, while CHF resulting from the homogeneous nucleation flow pattern is relatively rare, occurring only under special circumstances.

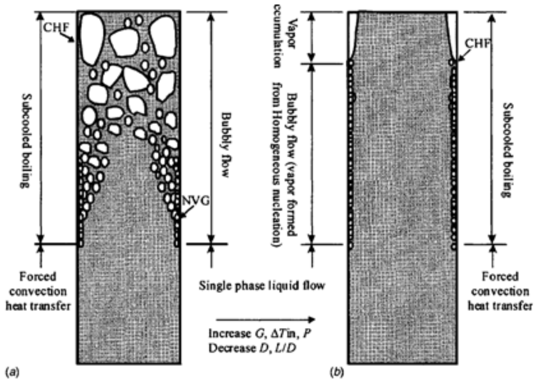


Fig. 5. CHF Occurrences by Typical Flow Pattern and Homogeneous Nucleation Flow Pattern [15]

Liu et al. [15] proposed a condition for the occurrence of CHF based on the homogeneous nucleation flow pattern as an inequality $q_{NVG} \geq q_{HN}$ (where q_{NVG} is the heat flux at which NVG phenomenon occurs at the end of the heated channel, and q_{HN} is the heat flux at which the homogeneous nucleation phenomenon occurs at the end of the heated channel.) They also identified the conditions to intensify this inequality are (i) high mass flux, (ii) high inlet subcooling, (iii) low heated length-to-diameter ratio (L/D), and (iv) small heated channel diameter — all of which increase q_{NVG}; and (v) high pressure — which decreases q_{HN}. This conclusion was derived through the determination of q_{NVG} and q_{HN} using the Levy model, the Thome correlation, and the Lienhard correlation, respectively. In addition, Liu et al. [15] predicted that the magnitude of the CHF due to the homogeneous nucleation flow pattern (CHF_{HN}) would be comparable to q_{NVG}. (That is $CHF_{HN} \cong q_{NVG}$)

Using the Groeneveld LUT(2006), the CHF values (q_{CHF}) were determined for various thermal-hydraulic conditions. Following the methodology proposed by Liu et al. [15], q_{NVG} and q_{HN} were also calculated under the same thermal-hydraulic conditions, and the results are presented together in Figure 6. It was confirmed that, across most pressure ranges, the condition $q_{HN} > q_{CHF} > q_{NVG}$ holds, indicating that CHF is predominantly governed by the typical flow pattern, with the magnitude of q_{CHF} being greater than q_{NVG}. On the other hand, in certain high-pressure ranges, the condition $q_{CHF} \cong q_{NVG} > q_{HN}$ was observed, suggesting that under high-pressure conditions, CHF occurs due to the homogeneous nucleation flow pattern, with the magnitude of q_{CHF} being comparable to q_{NVG}. Therefore, these results validate the argument of Liu et al. [15], demonstrating the plausibility of CHF occurrence due to the homogeneous nucleation flow pattern.

For SMRs, (i) due to the reduced heated length, the power peaking factor increases compared to large commercial reactors (greater than 1.5). In addition, in SMRs that adopt a soluble-boron-free design (SBFD), where boron is not used for reactivity control, a strong bottom-skewed power distribution acts at the Beginning Of Cycle (BOC), further increasing the power peaking factor (up to 1.68–2). Such a strong power peaking enhances local heat flux, thereby increasing the likelihood of CHF occurrence. (ii) The heated length (fuel rod length) in the core is about 2 m, which is relatively short, resulting in the L/D ratio being reduced to about half that of large commercial reactors (a factor contributing to the increase of q_{NVG}). (iii) At BOC, the location where strong bottom-skewed power peaking occurs is near the reactor inlet, where the subcooling is very high (also a factor contributing to the increase of q_{NVG}). Therefore, these conditions correspond to those under which CHF induced by the homogeneous nucleation flow pattern is highly likely to occur [11,13]. Consequently, in SMRs adopting SBFD, the possibility of CHF occurrence due to the homogeneous nucleation flow pattern at BOC must be carefully examined. If such a possibility is confirmed, it implies that CHF would occur near the core inlet, and thus this location-specific characteristic should be reflected in the selection of instrument locations during CHF experiments [13].

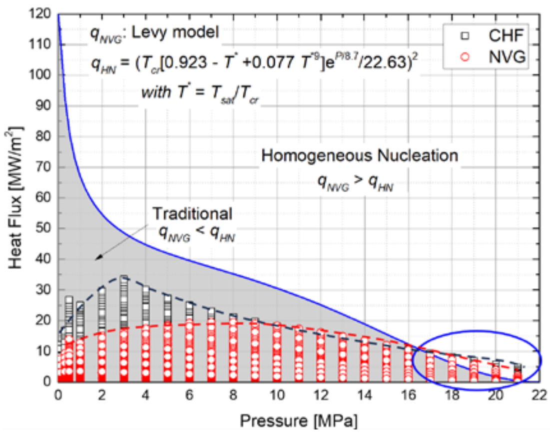


Fig. 6. Trends of q_{CHF}, q_{NVG}, q_{HN} due to Pressure Change

3.6 Avoidance of non-conservatism in CHF due to exit quenching phenomenon

As shown in Figure 7, SMRs are designed with a relatively large upper plenum compared to the core in order to secure sufficient driving force for natural circulation cooling during accidents. Reflecting this design characteristic of SMRs, CHF test facilities for SMRs are also constructed with an enlarged upper plenum section (see Figure 8a). As shown in Fig. 8a, a major issue with CHF test facilities for SMRs, which are equipped with an enlarged upper plenum, is that relatively cold fluid located in the unheated upper plenum may flow into the relatively hot upper region of the heated section during CHF testing. In other words, an exit quenching (or exit reflooding) phenomenon may occur. This phenomenon can be regarded as a problem unique to the SMR design, since it arises from the significantly enlarged upper plenum compared to the heated section—a feature that is not prominent in CHF test facilities for large commercial reactors, where the upper plenum is much smaller relative to the heated section (see Figure 8b) [13].

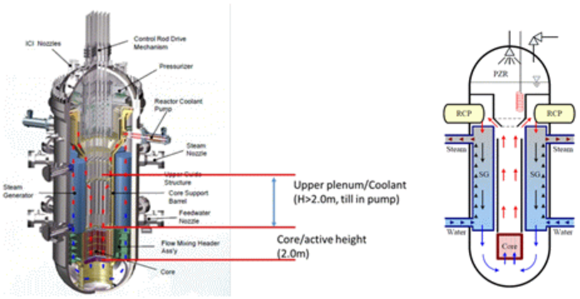
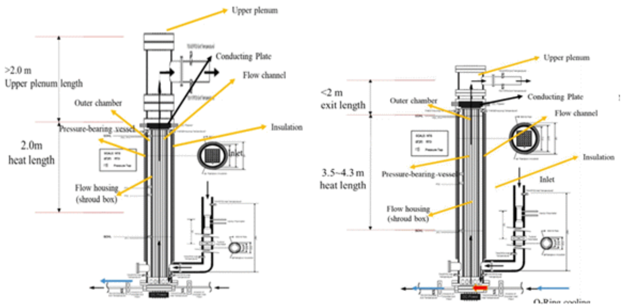


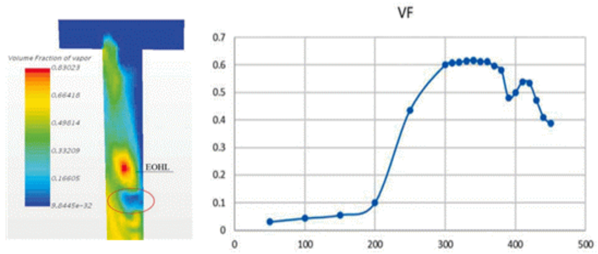
Fig. 7. Structure and Schematic of Typical SMR



(a) SMR (b) PWR

Fig. 8. Comparison of Typical CHF Test Facilities for SMR and PWR [16]

Liu et al. [17] demonstrated the occurrence of the exit cooling phenomenon through numerical simulations of CHF experiments under uniform heat flux conditions, using the system analysis code ANFR-ISS and the computational fluid dynamics code STAR-CCM+ (See Figure 9). Hoang et al. [18] also analyzed CHF occurrence by applying a non-uniform axial power distribution to the CE 5×5 test analysis model, considering cases where the upper plenum lengths were 0.483 m and 4.83 m, respectively. The MARS-KS/CTF code analysis results showed that the CHF for the condition with a very long upper plenum (4.83 m) was higher than that for the condition with a very short upper plenum (0.483 m), confirming that the exit cooling phenomenon (i.e., overprediction of CHF) indeed occurs when the upper plenum length is large. (See Figures 10, 11) In actual SMRs, the pressurizer is located in the upper plenum, where heaters and sprays are installed to control the core pressure. Therefore, this region is generally regarded as the area in which both temperature and pressure are maintained at the highest levels within the core. However, in CHF test facilities, it is difficult to replicate such a pressurizer at the upper plenum location. Consequently, if the upper plenum is simply enlarged to preserve geometric similarity, the absence of the pressurizer causes the fluid temperature in the upper plenum to become lower than that of the heated section, making it impossible to avoid the aforementioned exit cooling phenomenon during CHF testing. Furthermore, the increased heat loss through the oversized upper plenum relative to the heated section may accelerate the occurrence of the exit cooling phenomenon.



(a) Heated End (b) Section Averaged Axial

Fig. 9. Vapor Fraction Distribution of CHF Test Section [17]

The exit cooling phenomenon is an important safety issue, as it causes the CHF measured in experiments to be higher than the actual CHF occurring in an SMR core, thereby making CHF tests non-conservative. Therefore, CHF tests conducted for SMR design must: (i) be performed in test facilities designed to minimize such exit cooling phenomena, and (ii) develop CHF correlations that take into account the non-conservatism in CHF measurements caused by exit cooling.

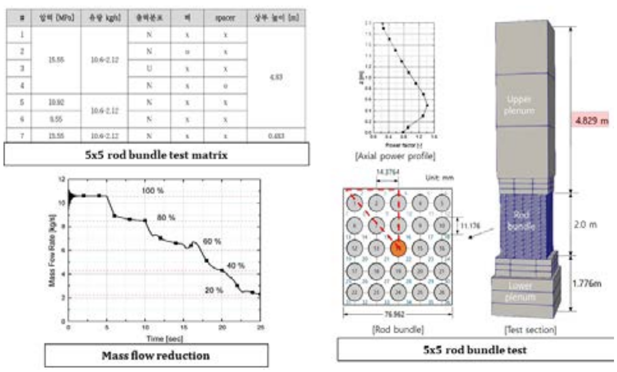


Fig. 10. Analysis Model for Upper Plenum Effect on CHF by CE 5x5 Test [18]

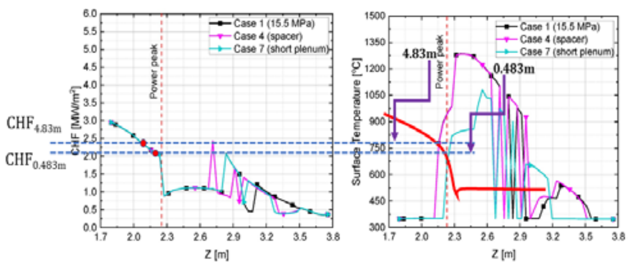


Fig. 11. Effect of Upper Plenum Height on CHF [18]

4. Difference between Active Safety System and Passive Safety System in terms of Conservative Safety Analysis

Most SMRs currently under development worldwide, including the i-SMR being developed in Korea, adopt passive safety systems. This trend is driven by the expectation that the use of passive safety systems can significantly reduce the frequency of core damage during AOOs and DBAs, compared to reactors employing active safety systems. Specifically, the i-SMR adopts passive safety systems such as PSIS, PRHRS, and PCS. SMART-100 employs PSIS, PRHRS, and CPRSS; VOYGR adopts DHRS, ECCS, and CS; SMR-160 incorporates PCCS and PCHRS; BWXT utilizes ECCS and PCCS; and the UK SMR adopts PDHR and ECCS as passive safety systems.

However, the safety analysis of reactors equipped with passive safety systems differs fundamentally from that of reactors employing active safety systems. This distinction is clearly illustrated in Figures 12 and 13. Figure 12 shows the MARS-KS thermal-hydraulic model for safety analysis of the OPR1000 reactor equipped with an active auxiliary feedwater system, while Figure 13 presents the MARS-KS thermal-hydraulic model for safety analysis of the APR+ reactor equipped with a passive auxiliary feedwater system.

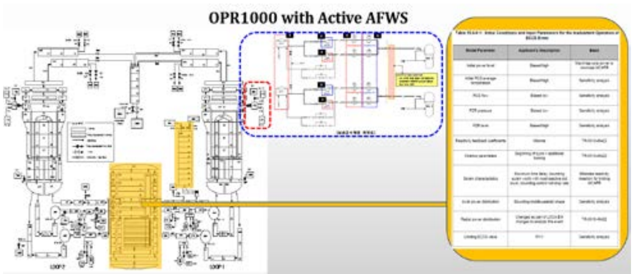


Fig. 12. OPR1000 MARS-KS Analysis Model including Active Auxiliary Feedwater System

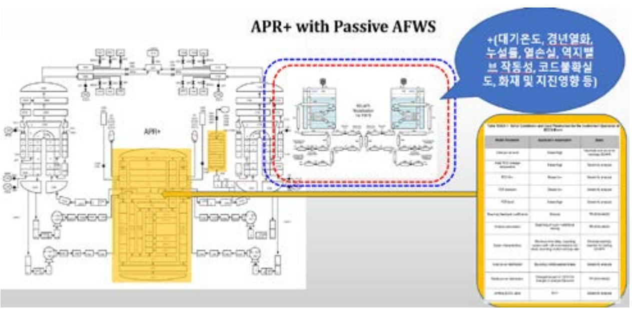


Fig. 13. APR+ MARS-KS Analysis Model including Passive Auxiliary Feedwater System

As shown in Figure 12, the active auxiliary feedwater system is modeled in the MARS-KS code in a very simple manner using components such as the Time Dependent Junction and Time Dependent Volume. (Within the small dotted red box) This simplified modeling is based on the rationale that, in the case of an active auxiliary feedwater system, once the auxiliary feedwater pump is actuated by the auxiliary feedwater signal, the injection of auxiliary feedwater into the steam generator is guaranteed to succeed according to the pump performance curve. In contrast, as shown in Figure 13, the passive auxiliary feedwater system is modeled in detail for the entire system and coupled with the reactor system. (Within the large dotted red box) This is because, although the initiation probability of the passive auxiliary feedwater system is higher than that of the active system, the amount of auxiliary feedwater delivered to the steam generator (i.e., heat removal capability — flow rate and temperature of the feedwater) is variably determined by the overall operation of the passive system.

The above case demonstrates that, compared to active safety systems, the heat removal performance of passive safety systems is more susceptible to various influencing factors. This intrinsic characteristic of passive safety systems must therefore be additionally taken into account when performing safety analyses that involve such systems. In the current conservative safety analysis methodology for reactors with active safety systems, sensitivity analyses are performed on various factors influencing the safety analysis results (e.g., reactor power, RCS temperature, RCS flow, RCS pressure, and pressurizer level; see yellow boxes in Figure 12). Based on these analyses, the plant initial conditions that yield the most limiting results for a given AOO or DBA are selected for safety analysis. When applying a conservative safety analysis methodology to reactors with passive safety systems, in addition to selecting conservative plant initial conditions based on the traditional set of factors, additional factors that affect the performance of passive safety systems must also be considered simultaneously. Only then can the analysis be regarded as a truly conservative safety analysis (see yellow boxes and blue bubbles in Figure 13).

4.1 Regulatory guidelines for a new conservative safety analysis with passive safety system

Park [19], through a review of reports from OECD/NEA [20], WENRA [21], and IAEA [22], as well as domestic regulatory guidelines on passive safety systems [23], identified the key review items for the safety assessment of passive safety system designs. Among these, the factors influencing the performance of passive safety systems were categorized as shown in Table I.

Table I lists numerous potential performance variation factors that may affect the performance of general passive safety systems. The applicability and impact of these potential performance variation factors can vary depending on the specific passive safety system of interest and AOOs or DBAs considered. Therefore, appropriate performance variation factors relevant to a given passive safety system must be carefully selected and additionally incorporated into the existing conservative safety analysis methodology.

Table I: Potential Factors Affecting Passive Safety System Performance

No.	Potential Factors Affecting PSS Performance	Note
1	Models and Correlations Uncertainties of Thermal-Hydraulic Code	Heat Transfer Model, Check Valve Model, etc.
2	Non-condensable Gas Concentration	
3	Leakage of Working Fluid	
4	Fouling Factor of Heat Exchanger	
5	Surface Effect on Condensation	Surface Contamination/Coating
6	Fluid Conditions of Heat Sink (Cooling Tank)	Temperature, Level
7	Initial System Configuration	Initial Inventory of PSS when it starts to operate
8	Aging Effect	Reduction in Pipe Diameter
9	Accident/Hazard Effect	• Containment Atmosphere Conditions* Chang (by Accident) • Temperature Distribution Along Circulatio Loop (by Fire) • Piping Slope Change (By Earthquake)
10	Boron Effect	• Flow Blockage due to Boron Precipitation • Heat Transfer Change due to Boron Coating o Heat Transfer Surface
11	Debris Effect	• Flow Blockage due to Debris • Heat Transfer Change due to Debris Coating o Heat Transfer Surface

To systematically identify the most relevant (i.e., most important) performance variation factors, Park et al. [24] considered: (i) potential performance variation factors as design or critical parameters within the REPAS (Reliability Evaluation of Passive Safety System) methodology [25]; (ii) directly applied the REPAS method to AOOs or DBAs in reactors equipped with specific passive safety systems; and (iii) derived the combinations of design or critical parameters that yield the highest failure frequency in the safety function of the passive safety system. Through this approach, they selected the most vulnerable performance variation factors.

Lee [26] identified the most vulnerable performance variation factors by (i) first evaluating the impact of potential performance variation factors on the performance of a specific passive safety system through independent assessments of each factor, and (ii) then selecting those factors that exert significant influence. Furthermore, by combining the derived performance variation factors with the REPAS method, Lee developed an advanced conservative safety analysis methodology—compared to the traditional Conservative Evaluation (CE)—in which the effects of passive safety system performance variation factors are systematically reflected. This methodology was termed the "Robustness Assessment Methodology for Performance Issues on Passive Safety Systems."

Following the methodology proposed by Lee [26], key performance variation factors were identified and, by incorporating the REPAS method, applied to a Total Loss of Flow (TLOF) accident scenario in a hypothetical v-SMART reactor. Figure 14 compares the results of the advanced conservative safety analysis methodology, which reflects key performance variation factors of passive safety systems, with those of the traditional conservative evaluation (CE) methodology, in terms of the arrival time at the safety shutdown temperature (488 K). In the case of the TLOF accident, the coolant temperature must be cooled below the safety shutdown temperature within a specified time (36 hours) by the operation of the passive residual heat removal system. The CE-based analysis (red curve) shows that the temperature decrease reaches the safety shutdown temperature much more slowly compared to the best estimate (BE) analysis (blue curve), which uses nominal values of the relevant parameters. However, when the major performance variation factors of the passive residual heat removal system are additionally considered (gray curves), the temperature decrease trend can be either slower or faster than that of the CE analysis, depending on the extent of variation. These results—particularly the slower decrease case—clearly demonstrate that only by incorporating the key performance variation factors of passive safety systems into the traditional CE methodology can give truly conservative safety analysis result.

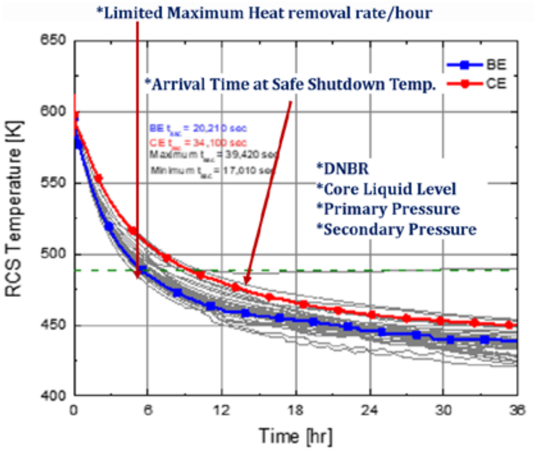


Fig. 14. Coolant Temperature Behavior during TLOF accident in v-SMART with PRHRS Performance Variation Factors Considered

In addition, when figure 14 is examined from the perspective of the limited maximum heat removal rate per hour, it shows that the maximum cooling rate can also be affected by variations in the key performance factors of passive safety systems. This indicates that, in the safety analysis of reactors employing passive safety systems, the performance variation factors of these systems must be considered in addition to the conventional conservative methodology in order to accurately evaluate whether major safety parameters—such as the arrival time at the safety shutdown temperature and the limited maximum cooling rate—are satisfied.

5. Conclusions

Based on the design characteristics of SMRs, the safety regulatory requirements and guidelines that should be applied in SMR safety analysis can be summarized as follows.

Safety Regulatory Requirement:

- For SMR SB LOCA acceptance criteria (Figures of Merit: FOM), the following must be applied:
    1. Collapsed liquid water level in the core
    2. Critical heat flux ratio
    3. Containment pressure and temperature

Safety Regulatory Guidelines:

- In addition to CHF correlations for high-pressure and high-flow conditions, separate CHF correlations for low-pressure and low-flow conditions must be developed and used in SMR safety analyses.
- CHF correlations for low-pressure and low-flow conditions required for SMR safety analysis must be developed through independent experiments that properly reflect the thermal-hydraulic range of the SMR under development. It is inappropriate to use the generic Groeneveld LUT.
- For SMRs, the occurrence of two-phase natural circulation flow instabilities under low-pressure and low-flow conditions must be evaluated. If such instabilities are confirmed, the CHF correlations for low-pressure and low-flow conditions should be modified to reflect premature CHF phenomena induced by flow instabilities (i.e., by reducing CHF values).
- Correction terms for the effect of mixing vanes on CHF under low-pressure and low-flow conditions must be separately determined based on independent CHF experimental data under such conditions and applied independently to the CHF correlations for low-pressure and low-flow conditions.
- For SMRs adopting soluble-boron-free cores, the possibility of CHF occurrence due to homogeneous nucleation flow patterns at the beginning of cycle (BOC) must be confirmed. If confirmed, this implies CHF occurrence near the core inlet, and such location-specific characteristics must be reflected in the selection of instrumentation locations during CHF experiment planning.
- For CHF experiments conducted to support SMR design:
    1. CHF test facilities must be constructed to minimize exit cooling phenomena.
    2. CHF correlations must be established considering the non-conservatism of CHF measurements caused by exit cooling phenomena.
- When applying a conservative safety analysis methodology to SMRs with passive safety systems, in addition to the conservative initial conditions used for reactors with active safety systems, the various performance variation factors of passive safety systems must be considered simultaneously.
- A systematic process must be used to identify the critical number of performance variation factor combinations that most strongly affect passive safety system performance. For this, methodologies developed from recent research or their equivalent, should be applied.
- Conservative initial conditions and systematically identified critical combinations of passive safety system performance variation factors must be combined to conduct safety analyses of SMR AOOs and DBAs.

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