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Preliminary Lessons-learned for Risk Assessment of SMRs from NuScale PSA Models

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1. Introduction

Numerous challenges related to SMRs(Small Modeler Reactors) are being pursued globally, and currently, NuScale Power, LLC has received the SDA(Standard Design Approval) from the U.S. NRC(Nuclear Regulatory Commission), which is based on an LWR(Light Water Reactor) technology.

Meanwhile, an i-SMR (innovative-SMR) is under development in South Korea. Among the top-tier requirements, i-SMR aims to achieve a CDF(Core Damage Frequency) of 1.0e-9/ry per single module, boron-free operation, fully passive safety systems, and load-following(flexible power) operation, which is basically based on the LWR technology[1]. The agency for developing i-SMR plans to apply for a SDA, with corresponding regulatory research also being conducted concurrently.

According to the Korean Nuclear Safety Act, there are legal requirements regarding risk assessment when applying for the SDA. Comparing with large-scale NPPs(Nuclear Power Plants), the SMRs have higher likelihood of affecting other modules in the event of an accident. Moreover, due to the fully passive safety systems, non-safety grade electrical systems, free-boron operation and flexible operation, a different approach is required for risk assessment conducted so far.

This paper presents a preliminary study related to the risk assessment methodology for light water SMRs by analyzing the differences between the NuScale PSA(Probabilistic Safety Assessment) model and large LWRs(Light Water Reactors). The characteristics of the NuScale US600 PSA model have not been publicly discussed in South Korea, so, based on the publicly available FSAR(Final Safety Analysis Report) Chapter 19 submitted by NuScale to the U.S. NRC and the NRC's evaluation report, this paper attempted to draw lessons-learned comparing with the characteristics with existing LWRs.

2. NuScale Design Characteristics Related to PSA

In contrast large-scale NPPs, all primary-side equipment exists within a single reactor vessel, excluding large LOCAs(Loss of Coolant Accidents). Additionally, the use of non-safety grade AC power eliminates Station Blackout as an initiating event[2].

2.1 Plant Familiarization

Table 1 lists the key systems required when performing a NuScale PSA.

Table 1. Systems Modeled in the PSA

System	Abbreviation
Chemical and volume control system	CVCS
Containment flooding and drain system	CFDS
Control rod drive system	CRDS
Decay heat removal system	DHRS
Demineralized water system	DWS
Electrical power systems	EHVS, EMVS, ELVS, EDSS, BPSS
Emergency core cooling system	ECCS
Module protection system	MPS
Reactor coolant system	RCS
Ultimate heat sink	UHS
Containment system	CNTS

Three systems, which show distinct differences from conventional large-scale NPPs, are selected and briefly explained below:

- CVCS: Most LOCA initiating events are significantly influenced by the CVCS loop rupture. As modeled in the PSA, the CVCS consists of a single loop with the DWS as the suction source for two parallel makeup pumps. The system provides the primary coolant makeup capability to remove core heat in the event of a LOCA.
- DHRS: While serving the role of the secondary-side heat removal system in traditional large-scale NPPs, the DHRS is designed as a passive safety system, ensuring higher safety. As modeled in the PSA, the DHRS consists of two redundant trains, one feeding each SG. Each train of the DHRS is equipped with a passive isolation condenser-type heat exchanger located in the reactor pool and two actuation valves. The system functions to remove core heat from the RCS(Reactor Coolant System).
- ECCS: As a fully passive safety system, it does not include electrically driven pumps and is composed of RVVs(Reactor Vent Valves) and RRVs(Reactor Recirculation Valves). As modeled in the PSA, the ECCS consists of three independent RVVs and two independent RRVs, which open to allow recirculation of

reactor coolant water between the reactor vessel and the CNV(Containment Vessel) to remove core heat during a transient state.

2.2 Initiating Events and Event Trees

NuScale proposed 11 initiating events/5 groups, as listed in Table 2.

Table 2. Initiating Event List of NuScale PSA

Category	Initiating Event
Loss-of Coolant Accident and Decrease in Reactor Coolant Inventory Events	CVCS Pipe Break Outside Containment - Charging Line (IE-CVCS--ALOCA-COC)
	CVCS Pipe Break Outside Containment - Letdown Line (IE-CVCS--ALOCA-COC)
	CVCS LOCA Inside Containment - Charging Line (IE-CVCS--ALOCA-CIC)
	RCS LOCA Inside Containment (IE-RCS---ALOCA-IC)
	Spurious Opening of an ECCS Valve (IE-ECCS--ALOCA-RV1)
Steam Generator Tube Failure	SGTFab (IE-MSS---ALOCA-SG-)
Secondary Side Line Break	Secondary Side Line Break (IE-TGS---FMSLB-UD-)
Loss of Electric Power	Loss of Offsite Power (Loss of Normal AC Power) (IE-EHVS--LOOP---)
	Loss of DC Power (IE-EDSS--LODC-----)
Transients	General reactor trip (IE-TGS---TRAN—NPC)
	Loss of support systems (IE-TGS---TRAN—NSS)

The initiating events of NuScale differ from those of large LWRs. For large LWRs, Station Blackout must be considered, but NuScale, mostly composed of passive safety systems, excludes it as an initiating event. Furthermore, while traditional NPPs classify LOCAs based on the break size, NuScale classifies initiating events based on the break location since the operable safety systems differ depending on the location of the break.

Due to differences in design concepts, there are discrepancies in the identification of initiating events between large-scale NPPs and NuScale. The full power internal event CDF for a single module is 2.7e-10/mcyr. The event with the highest CDF is "RCS LOCA Inside Containment," for which a representative accident sequence is introduced below:

This is LOCA due to the failure of the RPV steam or feedwater line, malfunctioning of RSV, or failure of the pressurizer heater penetration. Even if RCS inventory is lost, it is contained within the CNV, ensuring that the coolant inventory inside the CNV is maintained.

To explain the third sequence shown in Figure 4: (1) A LOCA occurs inside the CNV (%IE-RCS-LOCA-IC), (2) Rapid pressurization of the CNV leads to a Reactor Trip (RP_Success), (3) Reactor cooling fails due to the failure of RRV or RVV (ECCS_Fail), and (4) Core damage occurs due to the failure to inject RCS using CVCS (CVCS_Fail). This initiating event follows a similar accident progression to MLOCA in large LWRs. In the event tree, while an ATWS event would occur in large-scale NPPs if the reactor does not trip, NuScale, due to its inherent characteristics, provides sufficient reactivity control even if the reactor does not trip, handling the initiating event within the same scenario.

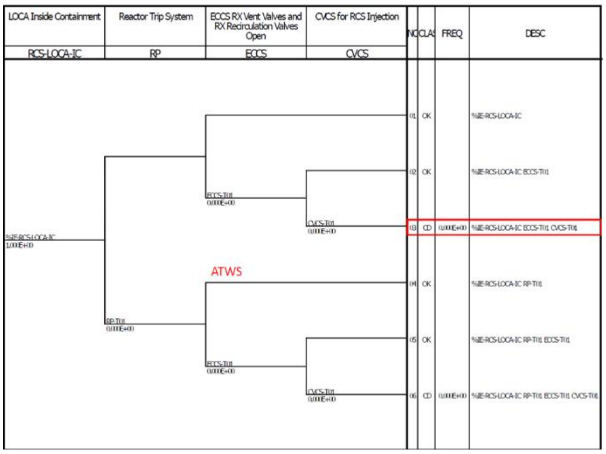

Figure 4. LOCA inside CNV Event tree

3. Regulatory Review for NuScale PSA

This section summarizes the content of Chapter 19 of the FSER(Final Safety Evaluation Report)[3], evaluated by the U.S. NRC when NuScale received its SDA in 2020:

- Level 1 Internal Event PSA at Full Power: Eleven initiating events were identified, with safety ensured beyond that of conventional PWRs through fail-safe features, passive safety systems, and heat removal systems. The review focused on the passive failures of the ECCS and DHRS, but sufficient margins to the CDF and LRF(Large Release Frequency) targets were maintained. In the sensitivity analysis for human reliability analysis, setting all values to 1 resulted in a 2 orders increase in CDF and LRF, yet they remained within acceptable target limits.
- Level 1 Internal Event PSA at Low-Power and Shutdown: During the refueling process, reliance on the PDHR(Passive DHR) removes the dependency on active systems common in large-scale NPPs. Although there is inherent uncertainty due to the lack of refueling experience, it was deemed sufficiently identified and defined. Since passive core cooling and heat removal are

achieved, so HRA(Human Reliability Analysis) is not required.

- External Event PSA at Full Power: (1) NuScale identified 41 external hazards, with analyses conducted on earthquakes, internal fires, internal flooding, external flooding, and high winds. The screening criteria used were consistent with proper criteria. (2) The screening results were considered acceptable because they were similar to those of previously certified reactor designs by the NRC.
- Evaluation of Multi-Module Risk: The parametric approach is deemed reasonable because it uses a systematic method to assess the risk of multi-modules. Although this approach heavily relies on engineering judgment-based assumptions (MMAF(Multi-Module Adjustment Factor), MMPSF(Multi-module Performance Shaping Factor)) and thus includes significant uncertainty, the NRC found it acceptable at the design certification stage.

4. Discussion and Conclusion

Since NuScale's design lacks operational history, most data used are based on the general failure probabilities of commercial PWRs, and unique components such as ECCS valves are calculated considering generic data, licensee event reports, operating experience, and design-specific information. HEP(Human Error Probability) values are calculated based on thermal-hydraulic analysis. The resulting CDF is based on limited information due to design and construction deficiencies, undeveloped operating procedures, and a lack of operational experience. However, the NRC noted that the SDA could be granted due to the high safety margin (CDF < 1.0e-7/mcry) outweighing uncertainties. These considerations should also be taken into account when SMR applies for an SDA.

Risk assessment methods not considered by NuScale also need to be developed.

Multi-modules have to consider that accident frequency at least two modules occurred and the potential for impact of accident propagation. To address this, a simplified method was used to derive multi-module risk based on a single module[2].

The method step:

1. Single module risk
2. Design capability evaluation & human effect event analysis
3. Cutset-level parametric adjustments (MMAF, MMPSF for IE(initiating Event) and BE(Basic Event))
4. PSA re-quantification and cutset generation

Expanding from a single module to a multi-module framework is a method recognized by the IAEA[4].

Analyzing the mathematical implications of this adjustment can be expressed as follows. The MMIE(Multi-module Initiating Event) frequency is expressed as $P(MMIE|IE) \times IE$ and $P(MM\ Fail)$ is expressed as $P(MM\ Fail\ |SM\ Fail) \times P(SM\ Fail)$. Thus, the equation for MMCDF is:

$$\boldsymbol{MMCDF = MMIE \times P(MM\ Fail)}$$
$$\boldsymbol{= \{P(MMIE|IE) \times IE\} \times \{P(MM\ Fail|SM\ Fail) \times P(SM\ Fail)}$$

For calculations of specific cutsets,

$\boldsymbol{SMCDF = IE \times P(SM\ CD) = IE \times A \times B \times C}$, then

$$\boldsymbol{MMCDF = (IE \times F_i) \times (A \times F_a) \times (B \times F_b) \times (C \times F_c) = (IE \times A \times B \times C) \times (F_i \times F_a \times F_b \times F_c) = SMCDF \times Adjustments}.$$

This expression reflects the conditional probability of multi-module failures given that a single module failure has already occurred. This is only valid under the assumption $\boldsymbol{P(MM\ Fail\ |SM\ Fail) = F_a \times F_b \times F_c}$.

In the area of multi-module risk, there are still unresolved issues, such as the analysis of event propagation between modules, the analysis when different initiating events occur simultaneously, and the suitability of MMPSF in these scenarios. One approach to addressing these uncertainties could be a simulation method[5] or a calculation method that creates an integrated fault tree for all modules without using MMAF, which could become a new multi-module risk assessment method. Similarly, new risk assessments for flexible operation and boron-free operation particularly for i-SMR will require convincing its regulatory satisfaction.

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