

A Review of NuScale Licensing Case Focusing on CHF under LOCA Conditions for Water - Cooled SMR

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1. Introduction

Critical heat flux (CHF) is the state in which contact between the coolant and the fuel-rod surface is interrupted, a vapor film forms, and the resulting sharp degradation in heat-transfer performance can damage the fuel cladding. Accordingly, water-cooled nuclear power plants are designed so that CHF does not occur during normal operation and anticipated operational occurrences (AOOs), to ensure that the specified acceptable fuel design limits (SAFDLs) are not exceeded [1]. Water-cooled small modular reactor (SMR) development is active worldwide, and Korea is likewise developing the i-SMR (innovative small modular reactor) design [2]. The i-SMR must also demonstrate, under Article 17 of the Regulations on Technical Standards for Nuclear Reactor Facilities, that fuel damage does not occur during normal operation and AOOs [3]. However, because of the i-SMR's innovative design features such as passive safety systems, there are inherent limitations to directly applying regulatory requirements and legacy CHF correlations established for large light-water reactors (LWRs) [4].

The NuScale Power Module (NPM), with its integral configuration and reliance on natural circulation, could not be evaluated under the LWR regulatory framework without modification. Accordingly, the U.S. NRC established CHF-based LOCA acceptance criteria and NuScale developed CHF correlations reflecting the low-flow natural-circulation regime characteristic of normal operation [5].

By reviewing NuScale's CHF and LOCA-related licensing case, this paper identifies a potential issue that may arise when directly applying LWR requirements to i-SMR. Based on this point, it proposes regulatory response measures for the additional CHF evaluations under LOCA conditions needed to support i-SMR licensing.

2. CHF correlations – NPM

● Licensing Requirements

For licensing in the United States, 10 CFR 52 section 47 and 79 require that the final safety analysis report (FSAR) describe the facility and provide safety analyses of structures, systems, and components (SSCs) [6]. The safety evaluation supporting the FSAR

includes accident and transient analyses that demonstrate compliance with the General Design Criteria (GDC) in 10 CFR Part 50, Appendix A—particularly GDC 10 on reactor design requiring adequate margin so that SAFDLs are not exceeded during normal operation and AOOs. In practice, consistent with NRC regulatory guide (SRP Section 4.4) and licensing precedents, CHF/DNBR limits are established as one-sided lower tolerance limits with 95% probability at 95% confidence (95/95) [7].

● Characteristics of NPM

Compared with conventional pressurized water reactors (PWRs), the NPM differs in its operational concept. The NPM relies on natural circulation to drive the reactor coolant system; consequently, its core mass flow rate is lower than that of pump-driven PWRs, and it operates at a lower system pressure. Because these flow and pressure conditions fall outside the applicability domain of legacy CHF correlations developed for large PWRs, NuScale could not directly adopt those correlations. Instead, NuScale developed specific CHF correlations based on data from dedicated CHF test programs [8].

● CHF test

The objective of NuScale's CHF testing was to obtain data suitable for developing CHF correlations applicable to fuel-bundle design and safety-analysis codes. The tests were conducted at two facilities; Table 1 summarizes the test campaigns, and Table 2 summarizes the resulting correlations.

Table 1. CHF experiments facilities

Facility	Stern Lab [9]	AREVA KATHY [1]
Fuel/bundle & grids	• Preliminary prototypical bundles	• NuFuel-HTP2 TM with HMP TM /HTP TM grids
Test setup	• Full-length • Full-power • Uniform & cosine axial shapes • Range of NPM flow & pressure conditions	• Same as left
Data use	• Basis for NSP1 correlation	• Validate NSP2; develop NSP4

Table 2. Summary of CHF correlations

Correlation		NSP1 [1,9]	NSP2 [1,9]	NSP4 [1]
Base dataset		Stern CHF data	NSP1+ KATHY CHF	NuFuel-HTP2™ CHF DB
Local variables		• mass flux • quality • heat flux	• mass flux • quality • heat flux	• mass flux • quality • heat flux
Spacer-grid treatment		-	NSPX coefficient from KATHY K8500 HMP™ tests	In-dataset
Applicable range	Pressure (psia)	300~2300	300~2300	500~2300
	Local mass flux (lbm/hr·ft ²)	0.11~0.7 x10 ⁶	0.11~0.7 x10 ⁶	0.11~0.63 x10 ⁶
	Local quality (%)	≤ 90	≤ 90	≤ 95

● *NRC technical review, RAIs, and Audits*

Table 3 summarizes the exchanges between the NRC and NuScale concerning Request for Additional Information (RAIs) and audit activities.

Table 3. RAI and audit activities summary

Date	Sender	Category	Content
2016 [10]	NRC	Audit planning	• Ready about NSP-1 TR submission • Focus on new phenomena under low-flow conditions
2016 [11]	NuScale	Submission of CHF TR	• Request approval to use the NSP2 CHF correlation and its limit
2016 [12]	NRC	RAI	• Local conditions for Stern tests at CHF • Local conditions for AREVA test at CHF
2017 [6]	NRC	RAI	• Request that NuScale provide a subject-matter expert about data reduction and analysis
2017 [13]	NuScale	RAI response	• Submitted supplemental local-condition data • Tong Factor • Subregion margin • Data density • Inlet subcooling etc.
2017 [1]	NuScale	Submission of CHF TR Revision 1	• To get approval for using NSP2 and NSP4 correlation
2017 [7]	NRC	Safety evaluation of TR	• NSP2 correlation usable with a limit of 1.17; NSP4 correlation usable with a limit of 1.21.

Subsequently, incorporating the outcomes of the NRC's RAIs and audits, NuScale submitted TR-0116-21012 Revision 1 in November 2017 [1]. This revision added the NSP4 CHF correlation, corrected errors in the local-condition data, and reflected the NRC's additional information requests. The NRC approved NuScale's

CHF correlation because it was logically developed and sufficiently validated based on appropriate experimental data, ensuring a 95/95 confidence level that fuel rods would not experience critical heat flux during normal reactor operation and anticipated operational occurrences (AOOs), thereby meeting regulatory requirements and providing adequate safety margins.

3. LOCA acceptance criteria – NPM

● *Design-Specific LOCA Evaluation for the NPM*

In the NPM design, a loss-of-coolant accident (LOCA) differs from that of a traditional PWR. This is because the reactor coolant system (RCS) piping is of small diameter and the RCS inventory lost during a LOCA is preserved within containment and, depending on break size, recirculates back to the core at some point after event initiation. The methodology employs the deterministic approach of 10 CFR Part 50, Appendix K, and the NPM is designed to eliminate or reduce many design-basis LOCA consequences compared to a typical large PWR [14]. In large PWRs, important LOCA consequences include peak cladding temperature (PCT) resulting from core uncover, core refilling, core reflooding, fuel-cladding swelling and rupture, and the fuel metal–water reaction. Accordingly, in Part 7 of the design certification application (DCA) NuScale requested exemptions from Appendix K, Sections I.A.5, I.B, I.C.5, and I.C.7, because phenomena associated with these Appendix K criteria are essentially avoided by the design of the NPM emergency core cooling system (ECCS) [14]. Details of the staff's review of this exemption request are provided in Section 15.0.2 of this safety evaluation report (SER). The NPM LOCA indicate substantial margins with respect to the PCT limit of 1,204 °C (2,200 °F) required by 10 CFR 50.46(b)(1), as well as the other criteria contained in 10 CFR 50.46(b)(2) through (b)(4). For this design, the relevant figures of merit are not PCT but rather (1) the collapsed liquid water level (CLL) above the core, (2) the critical heat flux ratio (CHFR), and (3) containment pressure and temperature [14]. Therefore, NuScale presented the following as the primary LOCA acceptance criteria instead of the traditional PCT and obtained NRC approval:

● *MCHFR safety limit*

In NuScale's LOCA safety analysis, the safety limit for MCHFR (Minimum Critical Heat Flux Ratio) is set at 1.29, and NuScale's analysis results significantly exceeded this limit, demonstrating sufficient margin [14].

● *Maintenance of the CLL*

To preclude core uncover or overheating during a LOCA, the NuScale design uses as a key acceptance criterion that the CLL in the RPV is always maintained above the top of the active fuel. This is also connected

to compliance with the long-term core cooling requirements of 10 CFR 50.46(b)(4) and (b)(5) [14].

To meet these regulatory requirements, NuScale demonstrated safety through development of NuScale-specific CHF correlations, a rigorous test program (conducted at Stern Laboratories and the KATHY test facility), and a detailed subchannel analysis methodology (NSAM/SSAM) using approved thermal-hydraulic codes such as VIPRE-01 (VIPRE-01 is a reactor core thermal-hydraulics code, and NuScale plans to use it for the NuScale design certification application and safety analyses) [14].

4. Characteristics of i-SMR LOCA

Because i-SMR adopts an integral configuration that eliminates large-diameter piping at the design stage, large-break loss-of-coolant accident (LBLOCA) is excluded from the design-basis accident (DBA) set, and only small-break LOCA (SBLOCA) is treated as LOCA DBA [15]. Unlike LBLOCA, SMR SBLOCA maintains the core submerged in coolant; accordingly, as illustrated by the NuScale example, the direct application of LBLOCA based acceptance criteria to SMR SBLOCA may be inappropriate.

According to Chapter 15.6.5 of the PWR Safety Review Guidelines (KINS/GE-N001), for advanced light-water reactors in which core uncover is not expected during the entire LOCA, a review shall be performed to confirm that a significant number of fuel rods are not damaged by local dryout. This can be demonstrated by showing that, after depressurization, at a given pressure the limiting fuel-rod heat flux remains below the critical heat flux (CHF) [16].

Following LOCA, the primary-system pressure drops sharply, and the passive emergency core cooling system removes heat from the core via natural circulation. Consequently, under SBLOCA conditions the i-SMR operating conditions lie in a low-pressure, low-flow regime. These conditions can fall outside the applicability range of CHF correlations developed for large plants (e.g., KCE-1 [17]). In this regime, the dominant CHF mechanism may transition from departure from nucleate boiling (DNB) to dryout or instability-induced CHF [18]. Therefore, for i-SMR accident scenarios similar to the NuScale case, a new CHF correlation that covers the expanded range of flow conditions is required.

5. Conclusions

NuScale adopts an integral reactor configuration that eliminates large-diameter primary piping at the design stage; consequently, within the LOCA only SBLOCA is treated as a DBA and LBLOCA is excluded. Under SBLOCA the core remains submerged, so the regulatory focus is to demonstrate that CHF does not occur in the fuel—an approach that likewise applies to i-SMRs, which also eliminate large primary piping by design. The key difference is operating regime: NuScale

relies on natural circulation even during normal operation and therefore resides in a low-flow regime, for which it developed and validated its CHF correlations. By contrast, the i-SMR operates at high flow during normal operation due to pump drive; however, when an SBLOCA occurs, depressurization and actuation of the passive emergency core cooling system shifts the plant to natural-circulation cooling, placing it in a low-flow (and low-pressure) regime. Accordingly, as NuScale did for normal operation and AOOs, the i-SMR must develop CHF correlations that are applicable to these post-SBLOCA low-pressure/low-flow conditions.

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