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Proposal for Amending Regulatory Requirements for SDA Applications

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1. Introduction

The global shift toward advanced nuclear reactor designs, particularly small modular reactors (SMRs) and non-light-water reactors, necessitates a fundamental restructuring of regulatory requirements. Unlike traditional light-water reactor (LWR) designs, these advanced systems often employ simplified, inherent, and passive means to achieve safety functions [1, 2]. However, the innovative features of these systems pose a challenge: their performance cannot always be predicted using legacy computer codes validated against traditional LWR data [3]. In the Korean nuclear safety context, a critical debate has arisen over the adequacy of submitting only a validation test plan versus actual performance results for Standard Design Approval (SDA)—the English translation of the Korean term “표준설계인가” [4], as reflected in the official English version of the Nuclear Safety Act (NSA) [5].

Considering the U.S. Nuclear Regulatory Commission's (USNRC) definitions of both Standard Design Certification (SDC) and SDA [6], along with the fact that a "certificate" is issued for repeated use over a specified validity period while an "approval" or "permit" is for one-time use, a more accurate English equivalent for “표준설계인가” appears to be SDC rather than SDA. Therefore, the NSA should be amended in the near future to officially define the appropriate English terminology, preventing misunderstandings among stakeholders regarding the substance and intent of “표준설계인가.”

Drawing on international best practices and technical standards from the USNRC, OECD/NEA, and IAEA [7-10], this article outlines the background, necessity, and methodology for establishing rigorous thermal-hydraulic (TH) validation requirements. It argues that while a test plan is essential, demonstrating safety features through actual testing is indispensable to align with global safety principles.

2. Regulatory background

2.1 Limitation of legacy data

The necessity for new testing requirements stems from the divergence of advanced designs from the operating history of the current fleet. Current system codes (e.g., RELAP, TRACE) and their associated correlations were developed and validated using data from large-scale test facilities representing traditional PWRs and BWRs [3, 8]. Advanced reactors utilizing passive safety systems—such as gravity-driven injection or natural circulation cooling—operate under different driving forces and thermal-hydraulic conditions [3]. Consequently, legacy data and correlations may not be applicable, necessitating new experimental data to quantify safety margins [9].

2.2 International regulatory standards

The gold standard for licensing advanced designs is codified in the U.S. regulation 10 CFR 50.43(e) [1]. This rule mandates that for designs differing significantly from pre-1997 LWRs, approval is contingent upon the following:

1. The performance of each safety feature has been demonstrated through analysis, appropriate test programs, experience, or a combination thereof.
2. Interdependent effects among safety features are acceptable.
3. Sufficient data exist to assess the analytical tools (computer codes) used for safety analyses.

Critically, the regulation implies that if these conditions are not met through existing data, a prototype plant or extensive testing is required to protect the public [1, 2]. The regulatory emphasis is on the existence of sufficient data to validate codes, not merely the intent to collect it.

3. Methodology for establishing test requirements

To ensure a reactor design meets safety goals, regulators and applicants must follow a systematic process to identify gaps in knowledge and define necessary experiments.

3.1 PIRT

The foundation of any validation methodology is the PIRT process. Experts identify phenomena of interest (POI) and rank them based on their importance to safety figures of merit (FOM) (e.g., peak cladding temperature) and the current state of knowledge (SOK) [8]. If the SOK for a high-importance phenomenon is low, specific testing is required. This process determines whether separate effects tests (SETs) or integral effects tests (IETs) are necessary [8].

The PIRT process involves distinct steps: defining the issue and objectives, compiling a database of relevant knowledge, specifying hardware and scenarios, selecting

FOM, identifying all plausible phenomena, ranking their importance, assessing the current SOK, and documenting the rationale [8]. This process is the starting point to identify important operational modes and rank phenomena based on their importance to plant safety.

Determining testing needs follows a hierarchical process. It begins with component-level performance, moves to system-level reliability (particularly for passive systems), and finally addresses system interactions and interdependencies. If safety claims cannot be substantiated through component, separate effects, or partial-scale testing, a full-scale prototype test is required. A prototype is defined as a facility similar to the standard design in all features and size, but potentially including additional safety features to protect the public during the testing period.

3.2 Hierarchy of validation tests

Once data gaps are identified, a hierarchy of testing is employed to validate the thermal-hydraulic performance of the design:

- SETs: These investigate individual phenomena (e.g., critical heat flux, condensation in containment) or specific components in isolation [3, 8]. They provide localized data to develop and validate physical models and empirical correlations [3].
- IETs: IETs simulate the global system behavior and interactions between components during accident scenarios (e.g., LOCA, Station Blackout) [3, 8]. They are crucial for verifying that the system codes correctly predict the coupling between different phenomena [2].
- Prototype Testing: If component and integral testing cannot sufficiently reduce uncertainties, a full-scale prototype plant may be required to demonstrate safety features [1, 2].

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3.3 Scaling analysis [8-10]

Since full-scale testing is often impractical, reduced-scale facilities are used. However, scaling distortions—deviations between the reference reactor and the test facility—are inevitable due to the inability to preserve all non-dimensional parameters simultaneously. Common scaling methodologies include power-to-volume scaling (preserving time and heat flux), hierarchical two-tiered scaling (H2TS), and three-level scaling. Rigorous analysis is required to ensure that dominant phenomena are preserved and that distortions do not invalidate the code assessment

4. Infrastructure for validation

Korea has established a robust methodology and infrastructure for validating advanced reactor designs, utilizing methodologies aligned with international standards.

4.1 Integral test facilities

Korea utilizes sophisticated IET facilities to support licensing:

- ATLAS (Advanced Thermal-hydraulic Test Loop for Accident Simulation) [3, 8-11]: Designed for the APR1400 and advanced PWRs, this facility uses Ishii's three-level scaling methodology. It has been instrumental in generating data for code validation (MARS, RELAP5) regarding complex scenarios like direct vessel injection (DVI) line breaks.
- SMART-ITL (VISTA-ITL) [8, 11]: This facility was specifically built to validate the performance of the SMART reactor (an integral SMR). It conducted tests on passive residual heat removal systems (PRHRS) and SBLOCA to support the standard design approval.

4.2 Code assessment

The data generated from these facilities is used to validate system codes such as MARS-KS and TASS/SMR-S. The validation process ensures that the codes can accurately predict safety margins and that the uncertainty of the predictions is quantified.

The objective of collecting experimental data is to validate system thermal-hydraulic codes, demonstrating their capability to predict global parameters and local phenomena across different scales. This process, known as the evaluation model development and assessment process (EMDAP), ensures that codes can facilitate the 'up-scaling' from IET data to the full-scale reactor, provided the code has been validated against counterpart tests (CTs) at different scales [9, 11].

5. Regulatory requirements for SDA application

The Korean regulatory requirements of document submittal for applications of SDA are compared to those of the USNRC to identify any deficiencies in the current rules and regulations pertaining the SDA application in Korea. As mentioned in Introduction previously, The USNRC issues SDAs and DCs under 10 CFR Part 52 to pre-approve nuclear reactor designs, enhancing safety and streamlining licensing. SDAs represent NRC staff approval for final designs or major portions that can be reconsidered late, while DCs are Commission-level, legally binding rules covering the entire design, which is valid for 40 years that fix design issues for future site-specific licensing [6].

Table 1. Comparisons in the regulatory requirements of document submittal for DC or SDA application between Korea and USA

	Document submittal for DC or SDA application
KR	**• NSA 12 (SDA) and ED of NSA 22 (Application for SDA)** **− ER 9 of NSA (Application for SDA)*** [12] 1) Standard Design Specification (design criteria, design details, and design, construction, and performance validation plans for the reactor facility) 2) Description of the reactor's intended use 3) Description of the reactor's design technology capabilities 4) SAR (including initial test plan) 5) Plan for developing an accident management plan in accordance with guidelines notified by the NSSC ※ ER 10 of NSA (Implementation of Validation Plan)** [12]: Requires implementation of a validation plan when applying for a construction and operation permit for a power reactor.
US	**• 10 CFR 52 Subparts B (DC), E (SDA)** **− 10 CFR 52.47 (DC), 52.137 (SDA) [6]** 1) FSAR 2) Proposed ITAAC (inspections, tests, analyses, and acceptance criteria) 3) An environmental report as required by 10 CFR 51.55 4) Special requirements according to its provisions, to particular applications (1) Certification applications for evolutionary LWR designs from pre-April 1989 licensed plants must provide a nearly complete plant design, excluding site-specific elements like service water intake and ultimate heat sink. (2) Applications for significantly different or innovative LWR designs must supply a complete design (excluding site-specific elements) and meet 10 CFR 50.43(e). (3) Modular reactor design applications must describe configurations, common systems, interfaces, and interactions; the FSAR must cover differences and restrictions for safe operation during construction and startup.

[Note]: *, ** Authors' translations that differ from the English version of the ERs of the NSA.

While, the NSSC issues only “표준설계인가” (translated into “SDA”) under the NSA article 12 to pre-approve standard designs of new nuclear reactors, which is also commission-level, legally binding rules valid for 10 years in Korea. The comparison is summarized in Table 1.

6. Gap between "Planning" and "Demonstration"

The current Korean regulation, which allows SDA based on a plan for validation tests rather than the results, presents a significant regulatory risk when viewed through the lens of international safety principles.

The major gap identified is that Korean regulation requires submission of performance validation plans for the applied reactor design whereas the US regulation provides submission of both performance validation results and ITAAC proposal for the applied reactor design.

6.1 Requirement for "sufficient data"

According to the EMDAP and 10 CFR 50.43(e), the approval of an advanced design is predicated on the availability of "sufficient data" to assess the analytical tools [1, 8]. If an applicant only submits a plan, the regulatory body cannot verify:

1. Whether the computer codes used in the safety analysis are valid for the specific design features.
2. Whether "unknown" phenomena or synergistic effects (which often appear during IETs) have been missed by the analysis [2].
3. Whether scaling distortions in the planned tests will render the data unusable for the full-scale plant [9].

If sufficient data does not exist to validate analytical tools, the USNRC may determine that the facility must be licensed as a prototype plant. This classification allows the USNRC to impose additional requirements on siting, safety features, or operational conditions to protect the public during the testing period [1, 2]. Licensing a design as a 'Standard Design' without this data bypasses these critical protective measures.

6.2 Counterpart tests

Validating system codes often requires CTs, where similar transients are simulated in test facilities of different scales (e.g., LOBI, SPES, LSTF) to assess the code's scale-up capability [9]. A plan alone cannot demonstrate that the code correctly predicts scale-dependent phenomena; only the comparative analysis of actual data from these facilities can confirm the code's reliability.

6.3 Role of testing in reducing uncertainty

Testing is not merely a formality; it is a tool to reduce licensing basis analysis uncertainties [2]. Without completed tests, these uncertainties remain high. In the US, if data is insufficient, the reactor might be licensed as a prototype with strict operational restrictions until testing demonstrates safety [2]. Licensing a standard design for commercial deployment without this validation data bypasses the fundamental check that ensures the design performs as predicted.

7. Conclusions

While the Korean infrastructure (ATLAS, SMART-ITL) demonstrates the capability to perform rigorous validation, the regulatory framework must ensure that the fruits of this capability—the actual test data—are a prerequisite for design approval. As noted in the USNRC guidelines, if testing results fail to substantiate safety claims, the design must be revised or the testing objectives redefined [2]. Proceeding with design approval based solely on a validation test plan removes the regulatory hold point necessary to enforce design changes if testing reveals safety deficiencies. To align with international nuclear safety requirements, the validation of thermal-hydraulic performance must be a completed milestone, not a future promise, prior to the issuance of SDA.

Therefore, the current Article 9, Paragraph 2 of the Enforcement Regulations of the NSA needs to be amended to require submission of the documents reporting performance confirmation validation results, rather than just the performance validation test plan and proposing the ITTAC, for SDA applications for proposed new reactor designs. In addition, the current Article 10, Paragraph 1 of the Enforcement Regulations of the NSA needs to be amended to require submission of the document committing the implementation of the proposed ITTAC.

NOMENCLATURE

Acronyms

BWR: Boiling Water Reactor
CFR: Code of Federal Regulations
DC: Design Certificate
ED: Enforcement Decree
ER: Enforcement Regulations
FSAR: Final Safety Analysis Report
IAEA: International Atomic Energy Agency
ITAAC: Inspections, Tests, Analyses, and Acceptance Criteria
KR: Korea
NEA: Nuclear Energy Agency
NSA: Nuclear Safety Act
NSSC: Nuclear Safety and Security Commission
OECD: Organization for Economic Co-operation and Development
PIRT: Phenomena Identification and Ranking Table
PWR: Pressurized Water Reactor
SAR: Safety Analysis Report
SDA: Standard Design Approval
SRP: Standard Review Plan
USNRC: United States Nuclear Regulatory Commission

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