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# A Study on the Reliability Assurance Program for Risk Informed Performance Based Regulation in Korea

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## 1. Introduction

The KINS (Korea Institute of Nuclear Safety) roadmap for risk-informed, performance-based regulation (RIPBR) for nuclear power plants [1] includes a plan to introduce a Reliability Assurance Program (RAP). Accordingly, detailed information on RAP is needed; however, sufficiently comprehensive introductory material is not yet available. Although several domestic papers on RAP have been published [2, 3], they address only parts of RAP, which is insufficient for understanding the program in its entirety. We believe that Korea should thoroughly understand RAP before its adoption so that subsequent implementation and institutionalization can proceed smoothly. Therefore, this paper discusses a broad range of RAP-related topics, including the background to RAP's development in the United States, its relationship with RTNSS (Regulatory Treatment of Non-Safety Systems), RAP for non-light-water reactors, its relationship with the Maintenance Rule, and safety classification. In particular, recent moves to legislate the Maintenance Rule [4, 5]—which can be regarded as part of O-RAP (Operational RAP)—and the growing importance of D-RAP with the emergence of non-LWRs, for which RIPBR is essential, make this paper especially timely.

## 2. Necessity of RAP

### *2.1 Background of the RAP's development*

During the 1979 Three Mile Island (TMI) accident in the United States, a pressurizer power-operated relief valve (PORV) opened due to an increase in primary-system pressure and then failed to reclose even after the pressure decreased. This "stuck-open" failure significantly aggravated the event. After the accident, the U.S. Nuclear Regulatory Commission (NRC) recognized the need for an institutional framework to improve the reliability of nuclear plant equipment and prevent recurrence of similar events. Accordingly, the NRC decided to introduce a Reliability Assurance Program (RAP), a reliability-assurance framework for systematic management of equipment important to safety throughout the entire life cycle [6].

Because RAP must be applied starting from design and construction, it is not well suited for retroactive application to operating plants; it is more appropriate for newly constructed plants. In the 1990s, evolutionary advanced light-water reactors (e.g., AP600, AP1000, ABWR), i.e., Generation III plants equipped with passive safety systems, emerged. Consistent with this trend, RAP application was requested in the licensing process for these Generation III designs. These new plants had limited operating experience and greater uncertainty associated with passive safety features; thus, the rationale for introducing a new reliability-assurance regulation—requiring assurance from the design and manufacturing stages—was well aligned. Consequently, the NRC required a design-stage reliability assurance program, D-RAP (Design Reliability Assurance Program), for such plants [7–10].

Meanwhile, in the 1990s, improving the economic competitiveness of nuclear power was an important policy issue in the United States. In addition, operating plants were already managing reliability and availability of key equipment through the Maintenance Rule [4, 5] and in-service testing (IST). Therefore, there was little need to create a separate O-RAP that would impose additional burden on licensees. The NRC decided not to establish new O-RAP requirements and instead to continue leveraging existing programs [10].

### *2.2 Purpose of the RAP*

The primary purpose of D-RAP is to demonstrate that the assumptions on reliability, availability, and performance used in probabilistic safety assessment (PSA) and safety analyses for a new plant are actually met during design, fabrication, and installation, and to establish a management system that can systematically monitor them.

The following objectives of RAP were defined [9].

- To ensure that, as evolutionary advanced light-water reactors (ALWRs) are designed, constructed, and operated, the reliability assumptions and risk insights for risk-significant SSCs do not change.

- To prevent degradation of the reliability of risk-significant SSCs during plant operation.
- To minimize the occurrence of transients that could challenge ALWR SSCs.
- To ensure that the functions of these risk-significant SSCs operate reliably when needed.

Accordingly, D-RAP requirements apply to new reactor designs rather than to existing large, mature LWRs—specifically, plants with passive safety systems, Generation IV (GEN-IV) reactors, and non-light-water reactors (Non-LWRs)—during design, construction, fabrication, and installation. They do not apply to newly constructed plants that are essentially the same as existing large LWR designs with extensive operating experience.

## 3. The Scope SSCs of RAP

### *3.1 RAP-Scope SSCs of LWR*

D-RAP is not applied uniformly to all structures, systems, and components (SSCs) in a plant; instead, it is limited to SSCs that are risk-significant and important to safety [11]. For LWRs, risk-significant SSCs are determined as follows:

- SSCs (both safety-related and non-safety) with high values of importance measures.
  - Components with PSA importance RAW (Risk Achievement Worth) ≥ 2.
  - Components with PSA importance FV (Fussell–Vesely) ≥ 0.005.
- SSCs important from a defense-in-depth (DID) perspective.
- SSCs identified by an expert panel.

In addition, the following SSCs are included in the D-RAP scope [11]:

- Regulatory Treatment of Non-Safety Systems (RTNSS) (It is explained further in “3.2 RTNSS”)

### *3.2 RTNSS*

For plants with passive safety systems (most ALWRs), passive systems have limited operating experience and relatively small driving forces; therefore, to address uncertainties under accident conditions, certain active non-safety systems are designated as Regulatory Treatment of Non-Safety Systems (RTNSS). RTNSS SSCs are not required to meet the full set of safety-grade requirements, but they are subject to appropriate regulatory controls [8–10].

For selecting RTNSS among active non-safety systems, the following five scope/criteria were proposed [9, 10].

A. SSCs needed to mitigate ATWS (Anticipated Transient Without Scram) specified in 10 CFR 50.62 or SBO (Station Blackout) specified in 10 CFR 50.63, as deterministic NRC performance requirements.
B. SSCs needed to assure long-term safety (the period from 72 hours after a DBA through the subsequent 4 days) and to cope with seismic events.
C. SSCs needed to meet CDF < 1.0E−4/year and LRF < 1.0E−6/year for operational and shutdown conditions.
D. SSCs needed to achieve containment performance goals for severe accidents.
E. SSCs needed to prevent adverse system interactions between passive safety systems and active non-safety SSCs.

Thus, for example, an “RTNSS A” SSC is an active non-safety system selected according to the RTNSS-A scope and criteria above.

The RTNSS scope in SECY-93-087 [7] focused on non-safety active systems that provide a defense-in-depth role for passive systems, whereas SECY-94-084 [9] appears to have expanded RTNSS to reinforce vulnerabilities of passive-plant designs based on the five criteria above.

### *3.3 RAP Scope SSCs of Non-LWR*

<u>Risk-significant SSC Determination</u>

For Non-LWRs, some reactor types (e.g., molten salt reactors) do not have the conventional “core damage” concept. Accordingly, new regulatory approaches (NEI 18-04 [12], RG 1.233 [13]) have been introduced, requiring licensing basis events (LBEs) to satisfy frequency–consequence (F–C) targets. If an SSC’s unavailability within an LBE would prevent the F–C target from being met, that SSC is deemed risk-significant.

<u>Safety-Related (SR) SSC Determination</u>

- Among risk-significant SSCs, a subset is designated as SR SSCs, as follows:
  - The minimum set of SR SSCs shall satisfy all required safety functions (e.g., decay heat removal, reactivity control), and using SR SSCs alone, the site boundary dose for DBAs shall not exceed 25 rem.
  - SSCs that prevent the frequency of beyond-design-basis events (BDBEs) from increasing

into the DBE region and outside the F–C targets are designated as SR.

- SR SSCs may also be identified by an expert panel based on defense-in-depth (DID).

<u>Non-Safety-Related with Special Treatment (NSRST) SSC Determination</u>

- SSCs that are risk-significant but not designated as SR.
- SSCs identified by an expert panel as needing special treatment from a DID perspective.

In Non-LWRs, NSRST (Non-Safety-Related with Special Treatment) is used in place of the LWR concept of RTNSS. That is, risk-significant SSCs for Non-LWRs consist of SR SSCs and NSRST SSCs, which together form the scope of D-RAP SSCs for Non-LWRs.

## 4. SAR for D-RAP

### *4.1 SAR for D-RAP in LWR*

With respect to RAP, an applicant should submit the RAP information as Chapter 17.4 of the Safety Analysis Report (SAR) when applying for a design certification (DC) or combined license (COL). The criteria and process used by NRC staff to review SAR Chapter 17.4 RAP are described in Standard Review Plan (SRP) Chapter 17.4 [11].

RTNSS, meanwhile, should be submitted as SAR Chapter 19.3 for DC or COL applications. SRP Chapter 19.3 describes how NRC staff review the applicant's SAR Chapter 19.3 RTNSS submittal.

In addition, the requirements for ITAAC (Inspections, Tests, Analyses, and Acceptance Criteria) are described in SRP 14.3. ITAAC provides assurance—through inspections, tests, analyses, and acceptance criteria—that a DC/COL LWR is fabricated, constructed, installed, and will be operated as designed. During construction, all ITAAC items must be completed and satisfied, and the NRC must verify them before fuel loading/operation. Although D-RAP elements for key SSCs were originally planned to be verified through ITAAC, experience from Vogtle Units 3 and 4 indicated limited effectiveness. Accordingly, the NRC clarified in 2019 via SRM-SECY-18-0093 [14] that D-RAP does not need to be included in ITAAC.

### *4.2 SAR for D-RAP in Non-LWR*

The need for and role of RAP have been reflected in the recent technology-inclusive, risk-informed, and performance-based (TI-RIPB) regulatory framework, and RAP is being applied to Non-LWRs as well. However, with the introduction of RG 1.253 [15], the overall SAR format was comprehensively revised. As a result, RAP information that had previously been included in LWR SAR Chapter 17.4 is no longer reflected in the same structure; instead, it is distributed across different chapters and sections. Therefore, the LWR SRPs cannot be directly applied to Non-LWRs, and the SAR format has been changed to align with Non-LWR licensing processes.

Figure 1 illustrates this change: the left side shows the traditional LWR SAR format, while the right side shows the Non-LWR SAR format, highlighting that RAP and RTNSS items are addressed in different chapters/sections in the Non-LWR SAR (Long Mott NPP : Xe-100 HTGR).

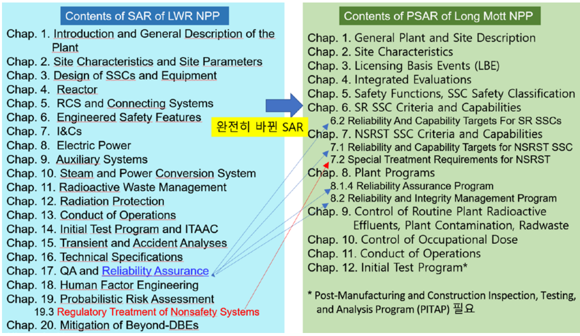


Fig. 1. Different SAR Format for D-RAP in Non-LWR

## 5. RAP Experience in Korea

### *5.1 RAP of APR1400 (NRC DC)*

To meet export conditions for the UAE, Korea applied for a U.S. design certification (DC) for the APR1400 [16]. Accordingly, RAP was described in APR1400 SAR Chapter 17.4 [17], which was reviewed by the NRC (APR1400 ultimately received DC in 2019).

For APR1400, the method for selecting D-RAP scope SSCs is the same as described in Section 3.1 of this paper (RAP-Scope SSCs).

For APR1400, 26 non-safety items were selected as RAP scope SSCs (risk-significant non-safety-related SSCs determined by design RAP), and their safety classification was upgraded to "A" [18]. Most of these 26 non-safety items are considered RTNSS.

Safety Class A was introduced in ANSI/ANS-58.14-2011 [19] as an intermediate class between safety-related and non-safety-related. It does not reach full safety-grade QA, but it provides comparable, strengthened (i.e., augmented) QA. SRP Chapter 17.5 [20] requires augmented QA for RAP or RTNSS SSCs. Therefore, RAP implementation is recommended to be performed by the existing QA organization rather than by establishing a separate dedicated team.

To confirm the intent of RAP—that SSCs designed, fabricated, and installed under D-RAP should maintain PSA-based performance such as reliability and availability during operation—we reviewed how the non-safety items selected as RAP scope in the APR1400 DC are being managed in Korean operating plants, even though Korea has not yet implemented D-RAP. The review indicates that most of the non-safety items selected as RAP scope in the APR1400 DC are managed under an "augmented" safety class in domestically constructed APR1400 plants.

### *5.2 Maintenance Rule and D-RAP*

If RAP is divided into D-RAP and O-RAP, O-RAP is very similar to the Maintenance Rule (MR), which has been implemented in the United States since 1996. Under the MR, performance criteria (e.g., reliability and availability) are established for key SSCs and monitored. By leveraging this existing function, the NRC concluded that there was no need to impose additional O-RAP regulatory requirements. D-RAP covers the period from design, fabrication, and installation through first fuel loading; after that, it transitions into the O-RAP domain.

In the United States, for COL plants, applicants are required to describe how D-RAP is linked to O-RAP.

In Korea, the Maintenance Rule has been operated as a voluntary program at domestic nuclear power plants by KHNP since 2006, and it is reportedly moving toward legal codification in the near future.

### *5.3 i-SMR and D-RAP*

If D-RAP is applied to the Korean i-SMR, the D-RAP approach used for the similar SMR design NuScale would be a useful reference.

<u>RAW criterion</u>

Because NuScale's plant CDF is very low, many components inherently yield RAW values greater than 2. Therefore, if the traditional criterion (RAW $\geq$ 2.0) is applied, an excessively large number of SSCs would be classified as risk-significant, increasing the licensee burden. To address this, NuScale did not use the traditional importance-measure thresholds; instead, it adopted separate criteria to select risk-significant SSCs and obtained NRC approval [21]. A similar approach may be needed for i-SMRs with very low CDF.

<u>RTNSS Augmented class</u>

Because the i-SMR also adopts passive safety systems, it is necessary to identify the RTNSS and decide whether their safety classification should be assigned as Grade A(augmented). If there are no RTNSS components, a justified rationale for their absence should be provided.

For design aspects not specified in Korean notices or regulations, overseas regulatory guidance and industry codes/standards applied by an applicant are used as reference documents during regulatory review. As SMART-100 and APR+ distinguished RTNSS [22–24], i-SMR designs are also expected to derive RTNSS SSCs.

## 6. Conclusions

This paper examined the background, purpose, scope, relationship with RTNSS, relevant SAR chapters, and domestic experience of D-RAP in preparation for the introduction of D-RAP to Korean reactor facilities.

RAP scope SSCs are risk-significant SSCs. Risk significance is basically evaluated using PSA importance measures, and additional SSCs may be selected by an expert panel considering defense-in-depth (DID) and deterministic insights. For plants with passive safety systems, RTNSS SSCs are included; for Non-LWRs, NSRST SSCs are included. Safety-related (SR) SSCs are, of course, included in RAP scope.

Because D-RAP is applied from design, fabrication, and installation, it is difficult to apply to operating plants. Therefore, the emergence of a new reactor type is an appropriate timing to require D-RAP, because new reactors have limited operating experience and higher uncertainty, providing a strong regulatory rationale for D-RAP. In Korea, i-SMR, APR+, and non-LWRs (e.g., SFR, HTGR) are such candidates.

If D-RAP is applied to i-SMRs and the plant CDF is very low, risk-significant SSCs may need to be selected using new criteria rather than the traditional RAW $\geq$ 2.0 threshold, similar to NuScale's approach.

In particular, because non-LWR is designed using TI-RIPB methodologies that explicitly use SSC reliability and availability [13], D-RAP is inherently needed to prevent degradation of SSC reliability/availability performance during design, fabrication, and installation. As Korea develops non-LWRs, it needs to institutionalize a RAP framework.

## REFERENCES

[1] 정구영, "원전 RIPBR 도입 로드맵 주요내용 및 향후 계획", 2024 KNS 추계학술발표회 워크숍

[2] Myung-Ki Kim, et. al., "Reliability Assurance Program for the APR1400 DC", Autumn Meeting of KNS, Oct 25, 2012

[3] Jae Gab Kim, et. al., "A Methodology for Reliability Assurance Program in Low Power Shutdown Probabilistic Safety Assessment", Spring Meeting of KNS, May 17, 2018

[4] 10 CFR Section 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.",

1991
[5] NRC, RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."
[6] Polish, TJ, "Reliability assurance program and its relationship to other regulations", Winter meeting of ANS, Nov. 1994
[7] NRC, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)", SECY-89-013, Jan. 1989
[8] NRC, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs", SECY-93-087, April 1993
[9] NRC, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs", SECY-94-084, March 1994
[10] NRC, SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)," May 22, 1995.
[11] Standard Review Plan, 17.4 Reliability Assurance Program, Rev. 1
[12] NEI, "Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development, Rev 1, NEI 18-04, Aug. 2019
[13] NRC, RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors", June 2020
[14] NRC, "Recommended Change to Verification of the Design Reliability Assurance Program, SRM-SECY-18-0093, September 2018
[15] NRC, RG 1.253, "Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors, 2024. 3. 1
[16] APR1400 Design Control Document Tier 1, APR1400-K-X-IT-14001-NP Rev 3, August 2018
[17] APR1400 Design Control Document Tier 2, Chapter 17 QA & Reliability Assurance, APR1400-K-X-FS-14002-NP Rev 3, Aug 2018
[18] APR1400 Design Control Document Tier 2, Chapter 3 design of structures, systems, components, and Equipment, APR1400-K-X-FS-14002-NP Rev 3, Aug 2018
[19] American Nuclear Society (2011). Safety and Pressure Integrity Classification Criteria for Light Water Reactor Plants, ANSI/ANS-58.14 (pp. 7~26).
[20] NRC, Standard Review Plan, 17.5 Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants
[21] NuScale Power, LLC, "Risk Significance Determination," TR-0515-13952-NP-A, Revision 0.
[22] Jin Hee Park, et. al., "Preliminary RTNSS Analysis for SMART", Trans. of KNS Spring Meeting, May 13, 2021
[23] 제145회 원자력안전위원회 회의록, 2021. 8. 13
[24] Dae Hyung Lee, et. al., "iPOWER Regulatory Treatment of Non-Safety System", Trans. of KNS Autumn Meeting, Oct 24, 2019