Nuclear Engineering and Technology 52 (2020) 69–77

Contents lists available at ScienceDirect

Nuclear Engineering and Technology

journal homepage: www.elsevier.com/locate/net

Original Article

# Development of a regulatory framework for risk-informed decision making

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ARTICLE INFO

*Article history:*
Received 27 March 2019
Received in revised form
27 June 2019
Accepted 30 June 2019
Available online 1 July 2019

*Keywords:*
Risk-informed decision making
Probabilistic safety assessment
Accident sequence precursor analysis
Significance determination process

ABSTRACT

After the Fukushima Daiichi accidents, public concerns on nuclear safety and the corresponding burden of nuclear power plant licensees are increasing. In order to secure public trust and enhance the rationality of current safety regulation, we develop a risk-informed decision making (RIDM) framework for the Korean regulatory body. By analyzing all the regulatory activities for nuclear power plants in Korea, eight action items are selected for RIDM implementation, with appropriate procedures developed for each. For two items in particular – the accident sequence precursor analysis (ASPA) and the significance determination process (SDP) – two customized risk evaluation software has been developed for field inspectors and probabilistic safety assessment experts, respectively. The effectiveness of the proposed RIDM framework is demonstrated by applying the ASPA procedure to 35 unplanned scrams and the SDP to 24 findings from periodic inspections.

## 1. Introduction

Following the Fukushima Daiichi accidents, public concern in Korea on the safety of nuclear power plants (NPPs) has increased, while the licensee (KHNP, Korea Hydro & Nuclear Power) has been struggling to enhance safety features. Previously, a risk-informed decision making (RIDM) [1] was successfully adopted in the regulatory system of the United States Nuclear Regulatory Commission (USNRC) after the TMI-2 accident to mitigate public concerns and the licensee's unnecessary burden [2,3]. The RIDM refers to the way in which risk information from probabilistic safety assessment (PSA) is used as part of an integrated process in making decisions about safety issues at NPPs [1]. The RIDM approach has to date been implemented in the United States, Finland, France, Spain, and Taiwan, with Japan planning to implement the RIDM into their regulatory inspection system by 2020 [4].

Before 2010, KHNP submitted several amendment requests for surveillance intervals, allowable outage time, and in-service inspection targets based on their risk information. The Korean regulatory body accepted [5] these amendment requests according to KINS/GT-N24 [6], Regulatory Guides 1.177 [7] and 1.178 [8]. On the other hand, attempts to introduce the RIDM into major regulatory activities such as “risk-informed periodic inspection”, “graded periodic inspection”, [9] and “monitoring effectiveness of maintenance program” [10] failed to be implemented in legislation due to a lack of confidence in PSA and insufficient consensus on regulatory resource relocations.

The purpose of this study is to develop an RIDM framework for the Korean regulatory body with high adaptability. In Section 2.1, a strategy for the RIDM framework development is derived from lessons learned from previous works to implement risk-informed activities into the Korean regulatory body. Next, regulatory activity items in which the RIDM can be applied are identified in Section 2.2, ultimately giving a total of eight action item, after which the RIDM procedures corresponding to the identified action items are introduced in Section 2.3. In Section 3, the effectiveness of the proposed RIDM framework is examined by applying it to recent safety issues in Korean NPPs.

## 2. Development of the RIDM framework

### 2.1. Development strategy

The regulatory body in Korea has been trying to implement risk-informed activities into the domestic regulatory framework [5] since the establishment of the Nuclear Safety Policy Statement [11]. However noticeable achievements have not been made mainly

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https://doi.org/10.1016/j.net.2019.06.029

because of resistance within the regulatory body, which can be attributed to the following two reasons.

(1) Lack of consensus on regulatory resource relocation: The relocation of resources for the implementation of the new regulation was met with resistance, and was also unable to be harmonized with domestic legislation. For example, the purpose of regulatory inspections for NPPs in the Nuclear Safety Act and related sub-decrees is only to ensure that the safety components of NPPs are in the licensed condition. Thus, if all components of a NPP meet the condition, there is no need to grade the safety level of the NPP or to relocate regulatory resources.
(2) Lack of confidence in PSA: Due to concerns of staffs about the uncertainty of PSA, it was hard to utilize PSA information in the regulation decision-making process. Staffs in the regulatory body may think that the PSA results are unreliable because of uncertain reliability data and error-prone complicated PSA models used by the licensee.

For reason (1), the lack of consensus in the regulatory body, it is necessary to emphasize that PSA can give an additional justification for the current regulatory activities. In particular, to mitigate the resistance in the regulatory body, we suggest applying risk information to regulatory activities in which the RIDM can be easily implemented while minimizing the resource relocation. Concerning reason (2), the lack of confidence in PSA, we believe that staffs can better understand the strengths and weaknesses of PSA by actually applying PSA information to current works, which will result in the enhancement of the PSA confidence. These three aspects of experience, understanding, and confidence compose a basic positive feedback system, as shown in Fig. 1.

Fig. 2 shows a diagram of the RIDM framework development reflecting the above implementation strategies. The main steps are (1) to identify action items that can be easily implemented with promising improvements of safety and the decision-making process through the evaluation of an expert panel, (2) to develop documents and software for supporting the action items, and (3) to examine its effectiveness through pilot applications.

## 2.2. Identification of the RIDM action items

### 2.2.1. Regulatory activity analysis

Every regulatory activity includes its own decision making. Thus, dedicated systematic analyses are carried out for all the regulatory activities. In Korea, Korea Institute of Nuclear Safety (KINS) performs nuclear safety regulatory activities such as regulatory review and inspection, except for nuclear security activities. Table 1 categorizes all the KINS activities into 4 main categories and 25 sub-categories. With this information, RIDM activity candidates are developed for each sub-category. Table 1 also shows the corresponding RIDM candidates for the categorized regulatory activities.

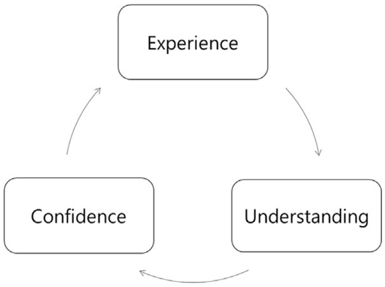


**Fig. 1.** The positive feedback system.

### 2.2.2. Assessment of candidates

In order to identify specific action items among the 49 candidates listed in Table 1, each candidate is assessed in terms of safety improvement and implementation feasibility. The USNRC [12] has applied seven guidelines to screen out regulatory activities that are not suitable for the RIDM implementation. Based on these seven USNRC guidelines, we develop a questionnaire using a 5-point scale considering the development strategy. The developed questionnaire includes two questions that evaluate the safety improvement and three that evaluate the implementation feasibility, as follows.

1. Improvement Category:
   1-1 Would a risk-informed approach help in improving the activity's safety? (Significant improvement +5 ~ No improvement +1)
   1-2 Would a risk-informed approach improve the regulatory decision making process of this candidate? (Significant improvement +5 ~ No improvement +1)
2. Feasibility Category:
   2-1 Do information (data) and analytical models exist that are of sufficient quality or could they be reasonably developed to support risk-informed regulatory activity? (Exist with sufficient quality +5/Exist without sufficient quality +4/Do not exist, but can be developed with sufficient quality +3/Do not exist, but can be developed with questionably sufficient quality +2/Do not exist, and cannot be developed with sufficient quality +1)
   2-2 At what cost level for the regulatory body and licensee can the startup and implementation of a risk-informed approach be realized? (Very low cost +5 ~ very large cost +1)
   2-3 To what extent would existing factors, e.g., legislative, judicial, or adverse stakeholder reaction, preclude changing the regulatory approach in an area, and hence limit the utility of implementing a risk-informed approach? (Could be implemented without any rule change, and no stakeholder reaction +5 ~ Implementation would be impossible +1)

The developed questionnaire was given to an expert panel comprising eleven members from the regulatory body, two from a related engineering company and ten from a related research institute [13]. All experts had more than ten years of experience in the field of nuclear safety.

The final RIDM candidates from among the total 49 candidates were selected via a screening-out process and subsequent prioritization assessments, as follows. When both of the improvement category questions (1-1 and 1-2) acquired less than 3 points on average from the expert panel, the candidate was screened out. When at least one question in the feasibility category (2-1, 2-2, and 2-3) acquired less than 3 points on average, the candidate was screened out because of its difficulty in implementation. As a result of the above screening-out assessment, 25 candidates were initially excluded from the total 49.

A prioritization analysis was then conducted on the remaining candidates for final selection. In the prioritization procedure, improvement and feasibility scores, denoted by $\boldsymbol{I}$ and $\boldsymbol{F}$, respectively, are calculated by

$$\boldsymbol{I} = \sum_{\boldsymbol{y}=1}^{2} \frac{\sum_{\boldsymbol{i}} \textbf{Score of expert } \boldsymbol{i} \textbf{ for question } \boldsymbol{y}}{\textbf{number of experts}}, \tag{1}$$

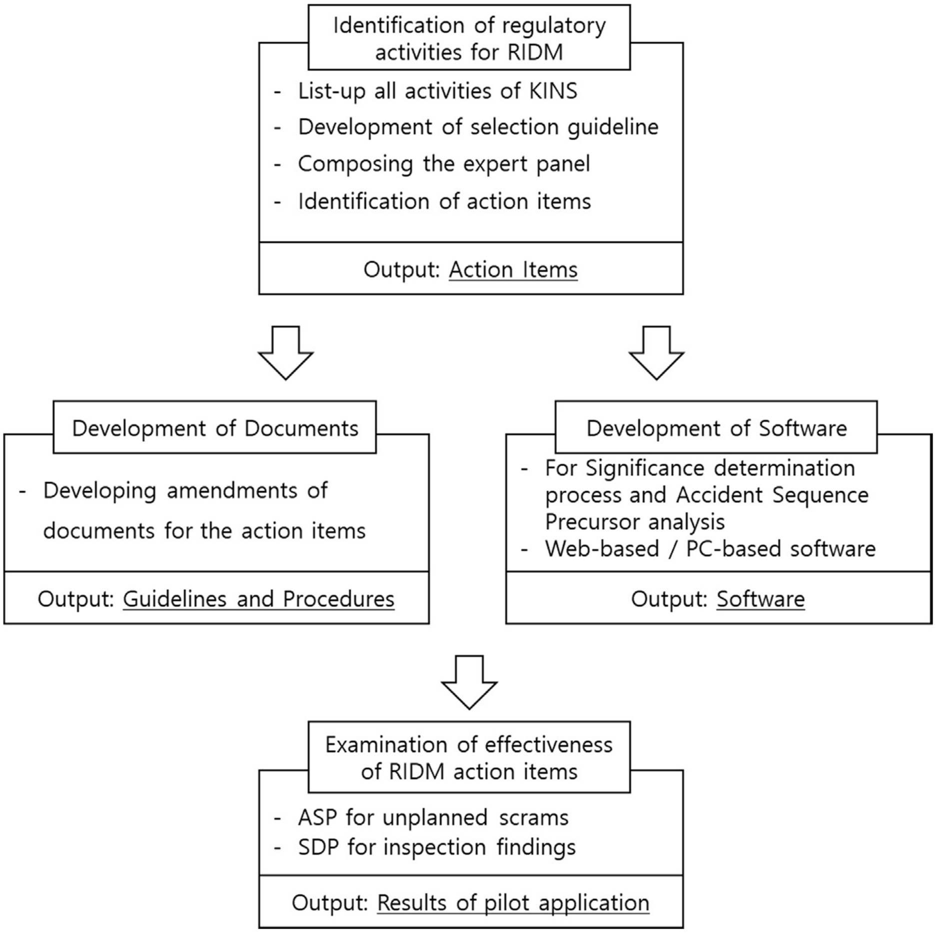


**Fig. 2.** Systematic diagram for developing the RIDM framework.

$$\boldsymbol{F} = \frac{2}{3} \times \sum_{\boldsymbol{x}=1}^{3} \frac{\sum_i \textbf{Score of expert } \boldsymbol{i} \textbf{ for question } \boldsymbol{x}}{\textbf{number of experts}}, \tag{2}$$

where $y$ and $x$ are indices for the improvement and feasibility questions, respectively, and $i$ is the expert index. The ‘2/3’ on the right side of Eq. (2) is inserted to make the maximum $\boldsymbol{F}$ value equal to the maximum of $\boldsymbol{I}$. Then, an $x$-$y$ plot is drawn with the $x$ value of $\boldsymbol{F}$ and the $y$ value of $\boldsymbol{I}$. From this plot, the priority of each candidate can be evaluated following the concept in Fig. 3, which shows four priority areas for a given item: high-priority (1st area), necessary with troubleshooting (2nd area), near-term (3rd area) and low-priority (4th area).

Fig. 4 shows the results of the prioritization analysis. The criteria for dividing the areas was determined as 7.0, which resulted in about 10 candidates in the high-priority area. From this analysis, all the candidates in the high-priority area, one candidate with the highest $\boldsymbol{I}$ value in the 2nd area, and one with the highest $\boldsymbol{F}$ value in the 3rd area were selected as the final candidates.

### 2.2.3. *Determination of action items*

Finally, the action items could be determined by merging the 14 candidates in Fig. 4 that can be implemented by the same procedure. The final list of determined action items is as follows.

#1. Accident Sequence Precursor Analysis (ASPA [14]) in incident/accident investigation and follow-up review (4.1.1)
#2. Significance Determination Process (SDP [15]) on inspection findings issued from periodic inspection and daily inspection (3.2.2, 3.4.2)
#3. Derivation of improvement items from a regulation perspective in PSAR/FSAR review (1.1.2)
#4. Independent risk impact assessment in amendment request for NPPs (1.2.1, 2.3.1, 2.4.1)
#5. Independent risk impact assessment in periodic safety review (PSR) safety improvement plans of the licensee (2.1.2, 2.2.2)
#6. Independent validation of submitted PSA compatibility in PSR, licensing and continued operation regulatory review (1.1.1, 2.1.1, 2.2.1)
#7. Impact assessment of design changes compared to preceding plants for new licensing (1.1.4)
#8. Daily Inspection observation item selection for resident inspectors (3.4.1)

Action items #1 (ASPA) and #2 (SDP), which are suggested to be newly implemented in KINS, require field inspectors as well as PSA expert activities, while action items #3–#8 can be done by PSA experts only. Since the PSA experts cannot review all the related events, field inspectors can support ASPA and SDP by performing preliminary assessments. Note that action items #3–#7 are RIDM processes to improve existing work procedures done by PSA experts, while action item #8 requires PSA experts to newly provide the on-site inspection team a priority list for daily inspection.

**Table 1**
Regulatory activities and RIDM candidates.

| Category | Sub-Category | RIDM Candidates |
|---|---|---|
| 1. Review (New NPPs) | 1.1 License Review (Standard Design Approval, Construction Permit, Operating License) | 1.1.1 Independent validation of submitted PSA |
| | | 1.1.2 Derivation of improvement items from a regulatory perspective |
| | | 1.1.3 Derivation of priority review item |
| | | 1.1.4 Risk assessment of design changes compared to existing NPP |
| | 1.2 Change Review (Construction Permit Amendment, Minor Change) | 1.2.1 Independent risk assessment of amendments of a construction permit |
| 2. Review (Operating NPPs) | 2.1 Periodic Safety Review | 2.1.1 Independent validation of the submitted PSA |
| | | 2.1.2 Risk assessment of safety improvement plan of the licensee in PSR |
| | | 2.1.3 Derivation of priority review item |
| | 2.2 Continued Operation (CO PSR, Equipment Life Assessment, Radiation Environmental Impact Assessment) | 2.2.1 Independent validation of the submitted PSA |
| | | 2.2.2 Risk assessment of safety improvement item licensee suggested for continued operation |
| | | 2.2.3 Derivation of priority review item |
| | 2.3 License Amendment Review (Operation Permit Change, Minor Matters Change) | 2.3.1 Independent risk assessment of the amendment |
| | 2.4 Risk-Informed Application Review | 2.4.1 Verification of submitted risk assessment results |
| | 2.5 Topical Report Review | 2.5.1 Independent risk assessment to submitted technical topical report (especially for risk-informed application) |
| | 2.6 Radiological Emergency Plan Review | 2.6.1 Sensitivity analysis of risk according to emergency plan (using level 3 PSA) |
| | 2.7 Decommissioning Safety Review | 2.7.1 Sensitivity analysis of SFP and LPSD risk according to decommission plan |
| 3. Regulatory Inspection | 3.1 Pre-operational Inspection | 3.1.1 Derivation of the priority inspection item |
| | | 3.1.2 Significance determination process on inspection findings |
| | 3.2 Periodic Inspection | 3.2.1 Derivation of priority inspection item |
| | | 3.2.2 Significance determination process on inspection findings |
| | 3.3 Quality Assurance Inspection | 3.3.1 Derivation of the priority inspection item |
| | | 3.3.2 Significance determination process on inspection findings |
| | 3.4 Resident Inspector Daily Inspection | 3.4.1 Daily inspection observation item selection |
| | | 3.4.2 Significance determination process on inspection findings |
| | 3.5 Suppliers Inspection (Planned/Unplanned) | 3.5.1 Derivation of priority inspection item |
| | | 3.5.2 Significance determination process on inspection findings |
| | 3.6 Special Inspection | 3.6.1 Derivation of the priority inspection item |
| | | 3.6.2 Significance determination process on inspection findings |
| | 3.7 Radiological Emergency Periodic Inspection | 3.7.1 Derivation of Priority inspection item |
| | | 3.7.2 Significance Determination Process on Inspection Findings |
| | 3.8 Decommissioning Inspection | 3.8.1 Derivation of the Priority inspection item |
| | | 3.8.2 Significance Determination Process on Inspection Findings |
| 4. Others | 4.1 Incident/Accident Investigation and Follow-up Review | 4.1.1 Accident Sequence Analysis of the events reported by regulation (notice of government) |
| | | 4.1.2 Accident Sequence Analysis of the events which is out of regulation (notice of government) |
| | 4.2 Radiation Accident Response | 4.2.1 Accident Sequence Analysis of the events reported by regulation (notice of government) |
| | 4.3 Operating Experience Analysis | 4.3.1 Importance analysis of International operating experience |
| | 4.4 Regulatory Issues Response | 4.4.1 Importance analysis of regulatory issues such as forgery case and performance test missing |
| | 4.5 Regulatory Research | 4.5.1 Prioritization of regulatory research candidates |
| | 4.6 Regulatory Policy and Standards Establishment | 4.6.1 Nuclear safety policies, regulations, and standards establishment priorities and risk impact analysis |
| | 4.7 International Cooperation | 4.7.1 Prioritization of international cooperation candidates |
| | 4.8 Miscellaneous | 4.8.1 Verification of MR Importance and performance criteria |
| | | 4.8.2 Improvement of safety performance indicators system of KINS |
| | | 4.8.3 Review and inspection of nuclear cycle facilities and research reactor |
| | | 4.8.4 Review and inspection of radioactive isotope usage permission |
| | | 4.8.5 Review and inspection of radioactive waste treatment facility |
| | | 4.8.6 Radiation in the natural environment regulation |
| | | 4.8.7 Measurement and assessment of environmental radiation |
| | | 4.8.8 Investigation on the person with abnormal dosimeter reading results |
| | | 4.8.9 Management of nuclear-related license and qualification examination |

### 2.3. *RIDM implementation for action items*

#### 2.3.1. *Amendments of regulatory documents*

Because all the regulatory activities are governed by relevant regulatory documents, RIDM implementations for the identified action items can be realized through document amendments. First, regulatory documents including the Notice of Nuclear Safety and Security Commission (NSSC), Regulatory Standard and Regulatory Guideline of KINS, and the work procedures of KINS were surveyed to find documents related to each action item. Next, new RIDM activities corresponding to the action items were suggested by inserting them into the related documents. The suggested new RIDM activities are processes that provide regulatory decision makers with risk information. Even though this may be an indirect way of using risk information, it is expected that decision makers will make better decisions than when considering only deterministic information. As more risk information is used for decision making, confidence in risk information and its impacts on the decision-making process will become stronger, as discussed in Section 2.1. Table 2 summarizes the new RIDM activities that can be implemented by amending related documents.

In Table 2, it can be seen that the new activities for ASPA and SDP

**Fig. 3.** Prioritization concept diagram.

include risk significance evaluation, which can be conducted by referring to the Risk Assessment Standardization Project (RASP) handbook [16] and Inspection Manual Chapter 0609 [15] of the USNRC describing their technical methodology.

### 2.3.2. Development of special risk assessment software for ASPA and SDP

As mentioned in Section 2.2.3, the ASPA and SDP action items requires, in addition to regulatory document amendments, special software for preliminary significance assessment. For this purpose, a web-based software named SEM (Significance Evaluation Management) [13] has been developed for field inspectors to screen-out issues with low safety significance; the characteristics of SEM include high accessibility, simplicity in use, and high maintainability. Fig. 5 shows screenshots of SEM.

In addition to SEM, a specialized software named RYAN (Risk analYsis for ASP/SDP of NPP) [17] has been developed by the Korea Atomic Energy Research Institute (KAERI) for PSA experts to efficiently conduct detailed significance evaluation. RYAN has enhanced features for safety significance assessment, such as a function for identifying and modifying basic events and comparing evaluation cases, so that integration calculation time can be saved. Basic event probabilities and exposure times can be modified using RYAN for a realistic safety significance evaluation, while they are fixed to their maximum values of 1.0 and 1 year, respectively, in SEM. Fig. 6 shows a screenshot of RYAN.

In the cases of high-significance issues determined by RYAN, a more detailed analysis including a revision of the PSA model can be followed by AIMS-PSA [18], which is a general-purpose PSA software such as NUPRA [19], RISKMAN [20], and SAPHIRE [21]. AIMS-PSA can integrate multiple event trees and fault trees to build one big fault tree called the one top for quantification. In this study, the RYAN and SEM calculations were conducted using the same one top fault tree, which was modeled by AIMS-PSA, as their input data. Note that AIMS-PSA, RYAN and SEM all use a common PSA engine, FTREX [22]. Fig. 7 shows the safety significance evaluation process.

## 3. Application results

In order to examine the effectiveness of the developed RIDM framework, we apply it to safety issues that can be covered by RIDM

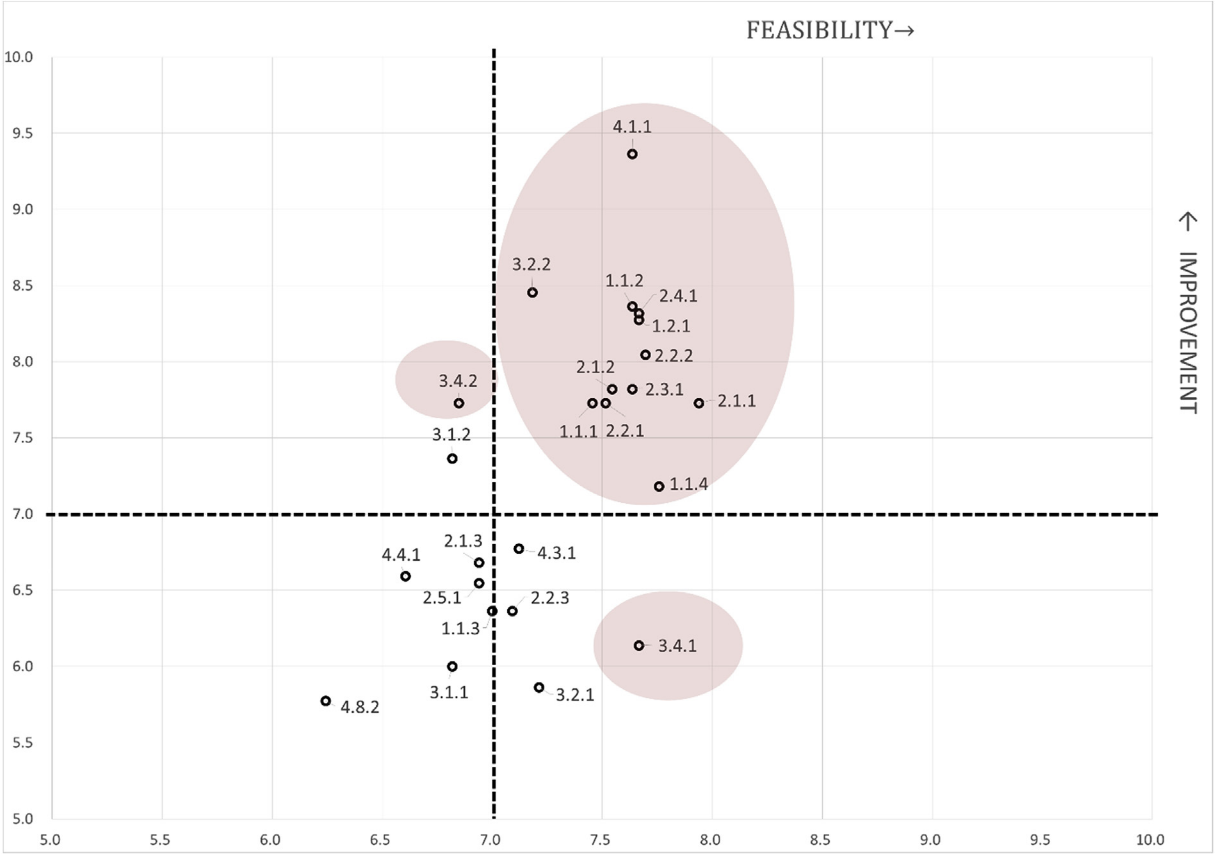


**Fig. 4.** Prioritization diagram showing the selected candidates.

**Table 2**
Developed documents for action items.

| Action Item | New activities | Draft documents developed |
|---|---|---|
| 1 | To include the risk significance evaluation result in the Event Report of the licensee and the event investigation report of the regulatory body in the incident/accident investigation process | Work Procedure of KINS |
| | To add the risk significance as a criterion for the event scale determination process of the regulatory body in the incident/accident investigation process | Notice of NSSC<br>Instruction of NSSC |
| 2 | To include risk significance of inspection findings in re-criticality meeting document of periodic inspection | Work procedure of KINS |
| | To include the risk significance of inspection findings in the report of periodic inspection of the regulatory body | Provision of handling services of KINS<br>Work procedure of KINS |
| | To include a safety significance grade in each inspection finding | Notice of NSSC<br>Instruction of NSSC |
| 3 | To suggest the improvement of design and operating procedures for new NPPs based on independent risk evaluation | (N/A)[a] |
| 4 | To perform an independent risk assessment on the amendment request using risk information | Safety Review Guide of KINS |
| 5 | To perform independent risk evaluation for safety improvement plans | Safety Review Guide of KINS |
| 6 | To perform an independent validation of submitted PSA compatibility | (N/A) |
| 7 | To perform an independent impact assessment of design changes compared to the preceding plant in new licensing | (N/A) |
| 8 | To select components for daily inspection of resident inspectors | Instruction of NSSC<br>Work procedure of KINS |

[a] There is no need to revise current regulatory documents and procedures.

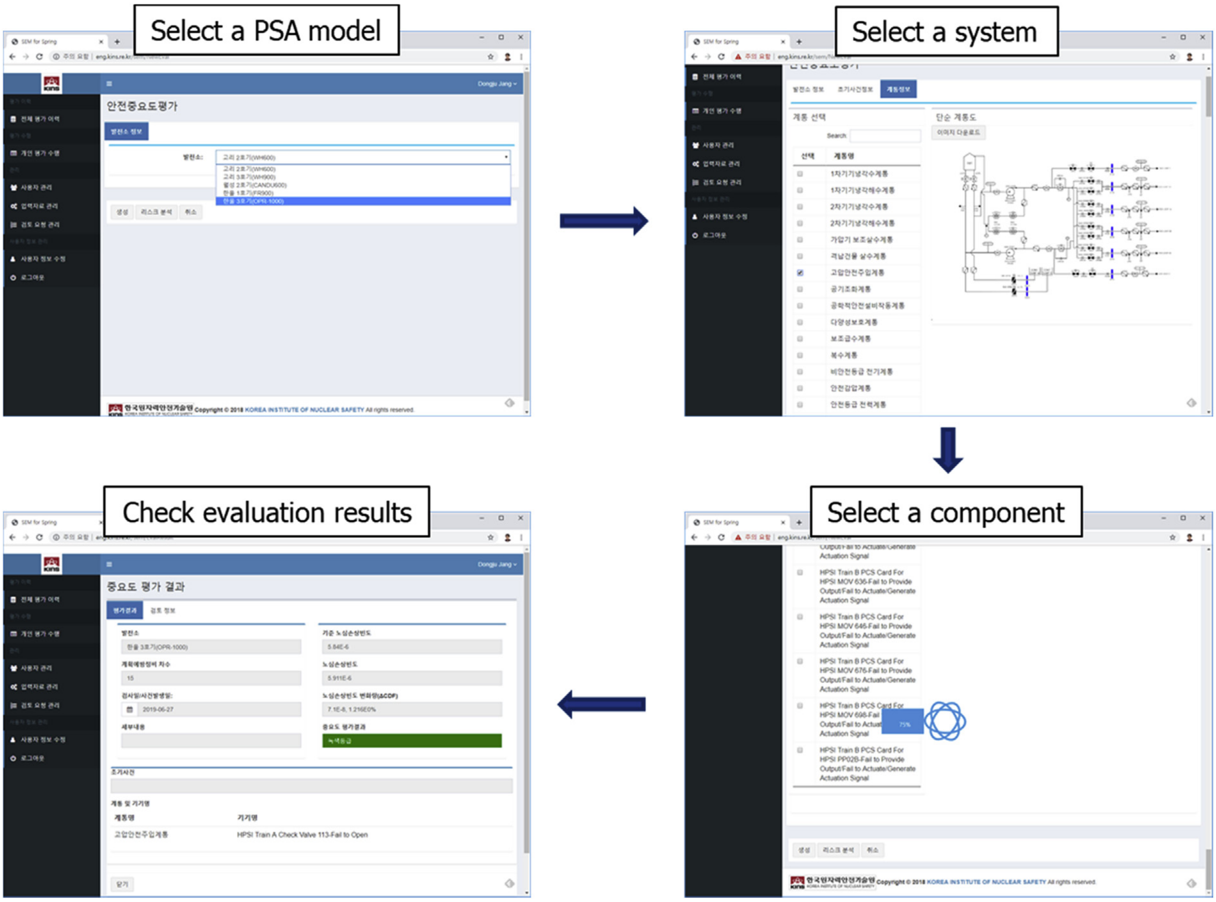
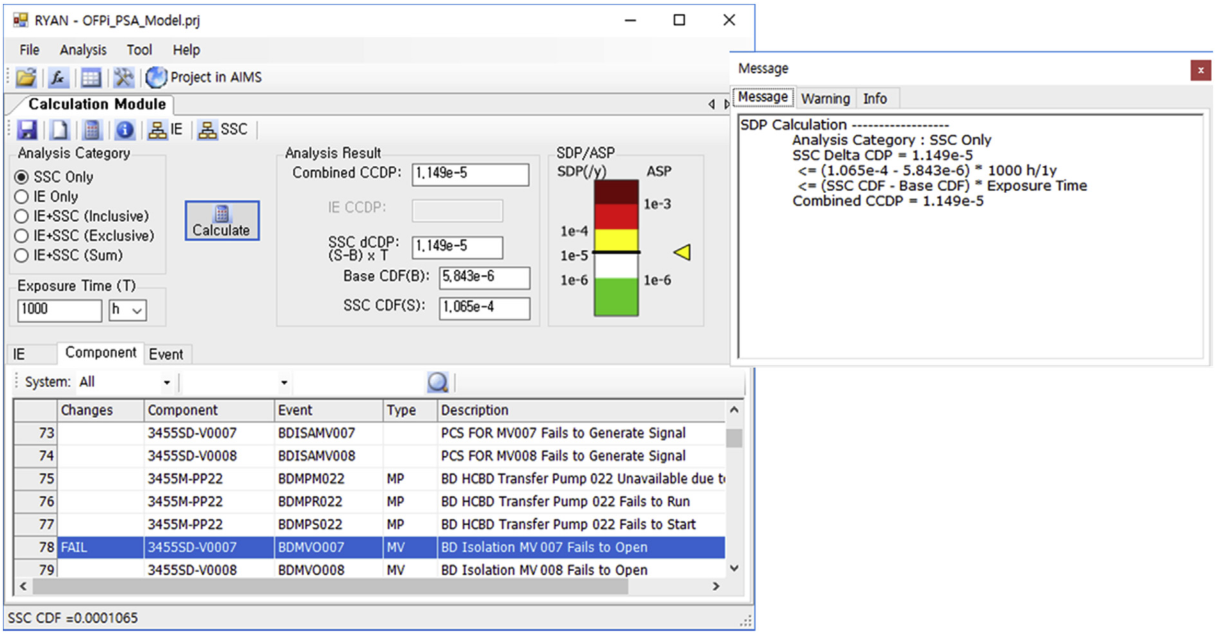


**Fig. 5.** Screenshots of SEM.

action items #1 and #2, i.e., ASPA and SDP. In Section 3.1, the developed ASPA procedure is applied to 35 unplanned scrams in OPR1000 reactors, including an unplanned scram for a corrective maintenance of a leakage from reactor coolant system (RCS). In Section 3.2, the developed SDP procedure is applied to 420 findings from periodic inspections, among which an inadequate maintenance for the trisodium phosphate (TSP) storage tank is presented. In the required SEM and RYAN calculations, a PSA model of the Multi-purpose Probabilistic Analysis of Safety (MPAS) for Hanul units 3 and 4 is used. Note that the following results can differ depending on the plant-specific PSA models applied.



Analysis Category
● SSC Only
○ IE Only
○ IE+SSC (Inclusive)
○ IE+SSC (Exclusive)
○ IE+SSC (Sum)

Exposure Time (T): 1000 h

Analysis Result
Combined CCDP: 1.149e-5
IE CCDP:
SSC dCDP: (S-B) x T: 1.149e-5
Base CDF(B): 5.843e-6
SSC CDF(S): 1.065e-4

System: All

| | Changes | Component | Event | Type | Description |
|---|---|---|---|---|---|
| 73 | | 3455SD-V0007 | BDISAMV007 | | PCS FOR MV007 Fails to Generate Signal |
| 74 | | 3455SD-V0008 | BDISAMV008 | | PCS FOR MV008 Fails to Generate Signal |
| 75 | | 3455M-PP22 | BDMPM022 | MP | BD HCBD Transfer Pump 022 Unavailable due t |
| 76 | | 3455M-PP22 | BDMPR022 | MP | BD HCBD Transfer Pump 022 Fails to Run |
| 77 | | 3455M-PP22 | BDMPS022 | MP | BD HCBD Transfer Pump 022 Fails to Start |
| 78 | FAIL | 3455SD-V0007 | BDMVO007 | MV | BD Isolation MV 007 Fails to Open |
| 79 | | 3455SD-V0008 | BDMVO008 | MV | BD Isolation MV 008 Fails to Open |

SSC CDF =0.0001065

```
SDP Calculation ------------------
        Analysis Category : SSC Only
        SSC Delta CDP = 1.149e-5
         <= (1.065e-4 - 5.843e-6) * 1000 h/1y
         <= (SSC CDF - Base CDF) * Exposure Time
        Combined CCDP = 1.149e-5
```

**Fig. 6.** Screenshot of RYAN.

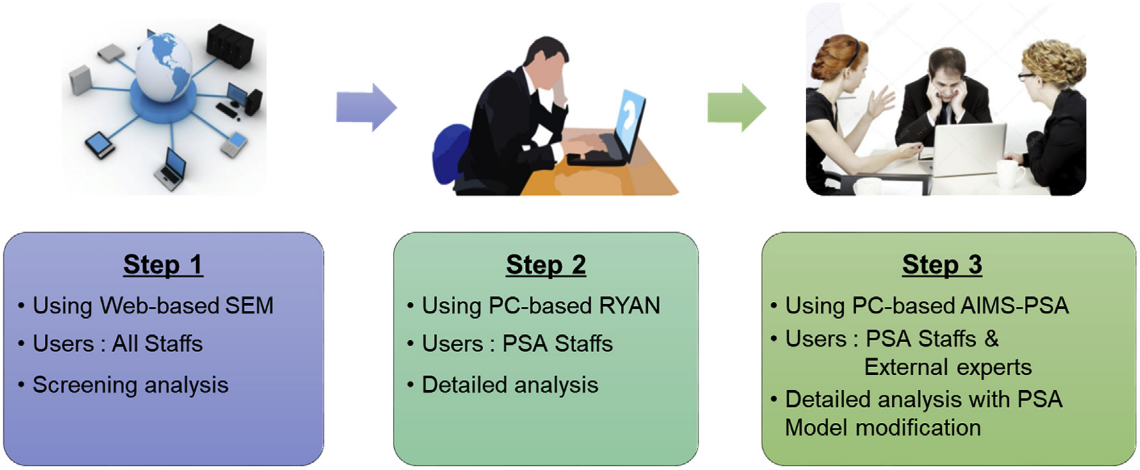


**Fig. 7.** Safety significance evaluation system.

### 3.1. *ASPA on unplanned scrams*

#### 3.1.1. *ASPA application for unplanned scram to mitigate RCS leakage*

The developed ASPA procedure is applied to an unplanned scram at the Hanul site. In this case, the NPP was manually scrammed for a corrective maintenance of a leakage from the RCS. The amount of leakage from a vent hole for a heated junction thermocouple of the reactor vessel was less than 0.1 gpm. This leakage was caused by corrosion of the vent ball acting as a seal. It was revealed that there was a nonconformity in the material of the vent ball. In the ASPA analyses, the initiating event of a small break loss of coolant accident (SBLOCA) in the MPAS model is selected as a surrogate for the degradation of the vent ball.

The ASPA is conducted following the methodology in the RASP handbook [16]. Therefore the significance grades of the unplanned scrams are categorized into 'Significant precursor', 'Precursor' and 'No precursor' by the criteria in Table 3 [16]. In this table, CDP and CCDP denote the core damage probability and the conditional core damage probability, respectively.

**Table 3**
ASPA criteria.

| ASPA grade | Criteria |
|---|---|
| No Precursor | $\triangle$CDP (or CCDP) $< 1.0 \times 10^{-6}$ |
| Precursor | $1.0 \times 10^{-6} < \triangle$CDP (or CCDP) $< 1.0 \times 10^{-3}$ |
| Significant Precursor | $1.0 \times 10^{-3} < \triangle$CDP (or CCDP) |

In the SEM calculation for the preliminary significance assessment that is supposed to be done by a field inspector, the frequency of the initiating event and the exposure time [16] are set to 1.0 and one year, respectively, for conservative results. From SEM, the CCDP value is calculated to be $2.347 \times 10^{-4}$ which indicate that the event is a Precursor in Step 1 of Fig. 7 by Table 3. Next, the detailed PSA analysis in Step 2 is conducted by RYAN. In the RYAN calculations, the following assumptions are applied for a realistic analysis.

- An increased likelihood of SBLOCA can be modeled by multiplying its frequency by ten.
- The exposure time of the degradation is assumed to be three months, which is still a conservative assumption because the operability of the leak detection system may be maintained.

The ΔCDP value calculated by RYAN is $2.607 \times 10^{-7}$, indicating that the event is not a precursor. From this application, it is demonstrated that the SEM calculation, taken together with RYAN, can be successfully employed as a preliminary screening tool.

### 3.1.2. ASPA results for recent unplanned scrams in OPR1000 units

From 2010 to 2017, there were 42 unplanned scrams in OPR1000-type reactors in Korea, i.e., Hanul units 3-6, Hanbit units 3-6, Shin-Kori units 1&2, and Shin-Wolsong units 1&2. Among these events, the developed ASPA procedure is applied in the same manner to the previous section for 35 unplanned scrams that occurred in the full-power operation mode and were induced by internal causes, because the OPR1000 MPAS model can cover only those cases.

Figs. 8 and 9 show the results of ASPA conducted by SEM and RYAN, respectively. From Fig. 8, one can see that there are no significant precursors, and 10 events (or 29%) among the 35 events are estimated to be precursors by the preliminary analyses of SEM. Fig. 9 indicates that only 7 cases among the 10 latent precursors are determined to be precursors by detailed analyses using RYAN with exposure time and basic event probability values. From these results, one can see that SEM provides a conservative enough significance grade to not screen-out any precursors from the field inspection.

## 3.2. SDP on findings from periodic inspections

### 3.2.1. SDP application for inadequate maintenance of a TSP storage tank

The developed SDP procedure is applied to a periodic inspection finding issued at the Hanul site: inadequate maintenance of a TSP storage tank. In this case, it was found that several bolts and nuts of the TSP storage box inside the reactor containment building were inadequately tightened during a periodic inspection. This kind of performance deficiency can impair the integrity of the suction line of the containment recirculation sump in an accident condition. In the SDP analyses, therefore, failures of valves between the containment recirculation sump and the suction of the recirculation pump are selected as surrogates for the performance deficiency of the NPP. The significance results in the SDP analyses are categorized into 'Red', 'Yellow', 'White', and 'Green' following the RASP handbook [16], as shown in Table 4. In the table, CDF denotes the core damage frequency.

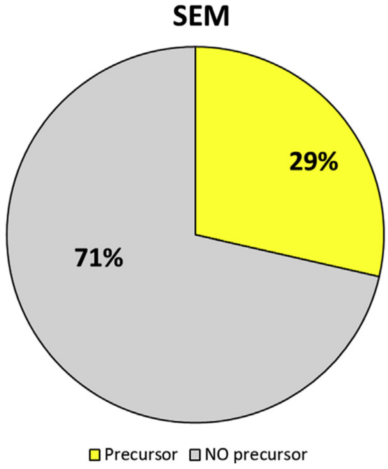


**Fig. 8.** Result of ASPA by SEM.

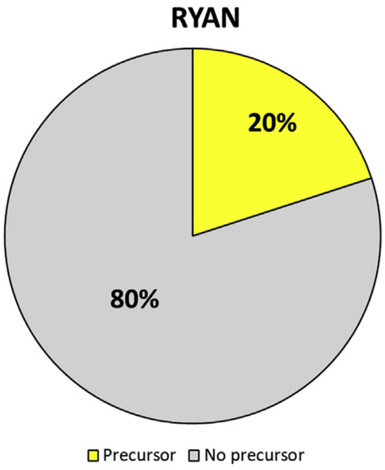


**Fig. 9.** Result of ASPA by RYAN.

In the SEM calculations, the probability values of fail-to-open events of the motor operated valves (MOV) in the high pressure safety injection trains A and B are set to 'True.' ΔCDF estimated by SEM is $5.439 \times 10^{-4}$ per year, and thus the safety significance level of this event is determined to be 'Red' in Step 1 of Fig. 7. For subsequent RYAN calculations, the following assumptions are used.

- The exposure time is assumed to be 9 months, which is half of the surveillance interval of the MOVs.
- The failure probability of the MOVs is multiplied by 100 to model performance deficiency.

The ΔCDF value calculated by RYAN is $2.515 \times 10^{-6}$ per year, corresponding to 'White' by the significance criteria of Table 4.

### 3.2.2. SDP results for recent findings from periodic inspection of OPR1000 units

From 2010 to 2017, 420 findings were issued during the periodic inspections of all the NPPs in Korea. Among these findings, 307 are screened out because they are not related to any components or initiation events of the PSA model. Among the remaining 113 findings which can be regarded as more-than-minor findings, we apply the developed SDP procedure to 24 findings that can be analyzed by the OPR1000 MPAS model with the level 1 full-power condition and internal causes.

Figs. 10 and 11 show the results of the SDP conducted by SEM and RYAN, respectively. From these figures, one can see that 11 cases (or 46%) among the 24 findings are calculated to have significance levels over than 'Very low safety significance', which are reduced to 5 (or 21%) by the RYAN calculations.

## 4. Conclusions

In order to improve the current safety regulation system in Korea, a RIDM framework with high adaptability has been developed by analyzing all the regulatory activities of KINS and identifying eight action items through an expert panel. Amendments of related documents for each action item have been suggested and new PSA tools have been developed. The effectiveness of the

**Table 4**
SDP criteria.

| SDP Colors | Meaning | Criteria |
|---|---|---|
| Green | Very low safety significance | $\Delta CDF < 1\times10^{-6}$ |
| White | Low to moderate safety significance | $1\times10^{-6} < \Delta CDF < 1\times10^{-5}$ |
| Yellow | Substantial safety significance | $1\times10^{-5} < \Delta CDF < 1\times10^{-4}$ |
| Red | High safety significance | $1\times10^{-4} < \Delta CDF$ |

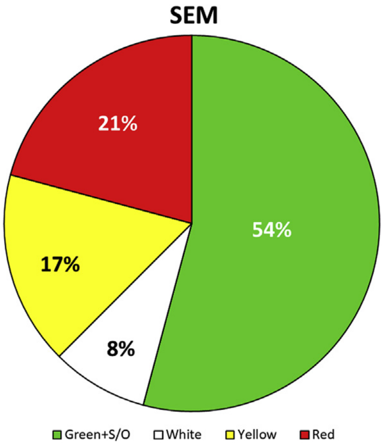


**Fig. 10.** Result of SDP by SEM.

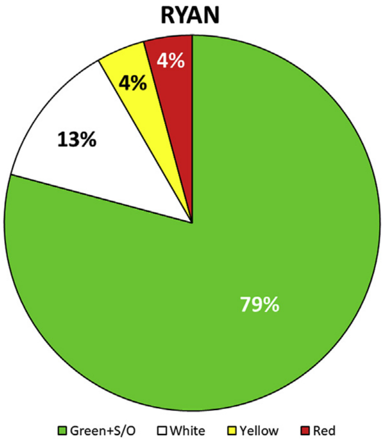


**Fig. 11.** Result of SDP by RYAN.

proposed RIDM framework is successfully demonstrated via applications of the ASPA and SDP procedures to recent safety issues in Korean NPPs. It is expected that the developed RIDM framework will be implemented in the Korean regulatory system.

## Acknowledgment

This research was supported by a Nuclear Research & Development Program of the National Research Foundation (NRF) grant, funded by the Korean government, Ministry of Science and ICT (NRF-2015M2A8A4047841).

## Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.net.2019.06.029.

## References

[1] IAEA, Risk Informed Regulation of Nuclear Facilities: Overview of the Current Status, 2005. TECDOC-1436.
[2] US Nuclear Regulatory Commission, Achieving Modern Risk-Informed Regulation, 2018. SECY-18-0060.
[3] G. Saji, Safety goals in risk-informed, performance-based regulation, Reliab. Eng. Syst. Saf. 80 (2003) 163–172.
[4] M. Yamamoto, Preparation status for the pilot program of the new nuclear regulatory inspection, in: OECD/NEA Workshop for the Introduction of ROP, 2018. June 21.
[5] Korea Institute of Nuclear Safety, Policy Report for Risk-Informed Regulation, September 2010. KINS/AR-911.
[6] Korea Institute of Nuclear Safety, Technical Guidelines on General Requirements for Risk-Informed Application for Amendment on Reactor Facilities, 2007. KINS/GT-N24.
[7] US Nuclear Regulatory Commission, An approach for plant-specific, risk-informed decisionmaking: technical specifications, Regul. Guide 1 (177) (1998).
[8] US Nuclear Regulatory Commission, An approach for plant-specific risk-informed decisionmaking for inservice inspection of piping, Regul. Guide 1 (178) (1998).
[9] Korea Institute of Nuclear Safety, Establishment of Implementation Program for Graded Regulation Using Risk and Performance Information, 2011, 2011-0002049.
[10] Y. J. Lee, H. S. Kim and T. W. Kim, "PSA lessons learned from maintenance rule," in Transactions of the Korean Nuclear Society Autumn Meeting, October 2005.
[11] Ministry of Science and Technology, Nuclear Safety Policy Statement, September 1994.
[12] US Nuclear Regulatory Commission, Risk-Informed Regulation Implementation Plan, 2001. SECY-01-0218.
[13] Korea Institute of Nuclear Safety, Development of Regulatory PSA Model Utilization Framework, 2018. KINS/GR-628.
[14] US Nuclear Regulatory Commission, Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission, 1978. NUREG/CR-0400.
[15] US Nuclear Regulatory Commission, NRC INSPECTION MANUAL Chapter 0609, SIGNIFICANCE DETERMINATION PROCESS, 2015.
[16] US Nuclear Regulatory Commission, Risk Assessment of Operational Events Handbook - Volume 1 – Internal Events Revision 2.0, 2013.
[17] Korea Atomic Energy Research Institute, Development of Basic Regulatory Framework for APR1400 Level 1 PSA, 2018. KAERI/RR-4383.
[18] S.H. Han, H.G. Lim, S.C. Jang, J.E. Yang, AIMS-psa: a software for integrated PSA, in: 13th International Conference on Probabilistic Safety Assessment and Management (PSAM 13), October 2016.
[19] B.O.Y. Lydell, Retrospective review of experience with using the NUPRA software, Reliab. Eng. Syst. Saf. 30 (1990) 339–353.
[20] D. Wakefield, S. Epstein, Y. Xiong, K. Nouri, RISKMAN®, celebrating 20+ years of excellence!, in: International Conference on Probabilistic Safety Assessment and Management (PSAM2010), April 2010.
[21] US Nuclear Regulatory Commission, Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Vol. 1 Summary Manual, vol. 1, 2008. NUREG/CR-6952.
[22] W.S. Jung, S.H. Han, J.J. Ha, Advanced features of the fault tree solver FTREX, in: Korean Nuclear Society Conference, 2005.