Development of Multipurpose Regulatory PSA Model

Chang-Ju Lee,[a] Joon-Eon Yang,[b] Key-Yong Sung,[a] Hho-Jung Kim,[a] Jaejoo Ha [b]
a Korea Institute of Nuclear Safety, P.O. Box 114, Yuseong, Daejeon, Korea, 305-600, cjlee@kins.re.kr
b Korea Atomic Energy Research Institute, P.O. Box 105, Yuseong, Daejeon, Korea, 305-600

1. Introduction

Generally, risk information for nuclear facilities comes from the results of Probabilistic safety assessment (PSA). PSA is a systematic tool to ensure the safety of nuclear facilities, since it is based on thorough and consistent application of probability models. In particular, the PSA has been widely utilized for risk-informed regulation (RIR), including various licensee-initiated risk-informed applications (RIA).

In any regulatory decision, the main goal is to make a sound safety decision based on technically defensible information. Also, due to the increased public requests for giving a safety guarantee, the regulator should provide the visible means of safety. The use of PSA by the regulator can give the answer on this problem. Therefore, in order to study the applicability of risk information for regulatory safety management, it is a demanding task to prepare a well-established regulatory PSA model and tool.

In 2002, KINS and KAERI together made a research cooperation to form a working group to develop the regulatory PSA model - so-called MPAS model. The MPAS stands for multipurpose probabilistic analysis of safety. For instance, a role of the MPAS model is to give some risk insights in the preparation of various regulatory programs. Another role of this model is to provide an independent risk information to the regulator during regulatory decision-making, not depending on the licensee's information.

2. Overview of MPAS model

This section presents an overview of the key features of MPAS model, which mainly consists of two parts; the quality management and development process. Fig. 1 shows the overall development framework, including the quality management, the process for detailed model development, and their inter-relationships.

2.1. Review of reference quality

First of all, as a reference for quality evaluation we determined to use the "ASME PRA Standard [1]," since this document is the first substantial product in the world for this purpose. The "ASME PRA Standard" mainly consists of three parts; risk assessment technical requirements, PSA configuration control, and peer review.

The working group reviewed the quality of previous licensee-developed KSNP (Korean Standard Nuclear Power Plant) internal full-power PSA model against the technical requirements of this document.

2.2. Ranking of previous model quality

The working group found that the problems of previous KSNP PSA mainly result from the insufficiency of documentation and/or the plant-specific realistic features rather than PSA techniques generally used in a traditional PSA model [2]. They has also derived what items should be considered to upgrade the quality of PSA up to Capability Category II of Ref. [1].

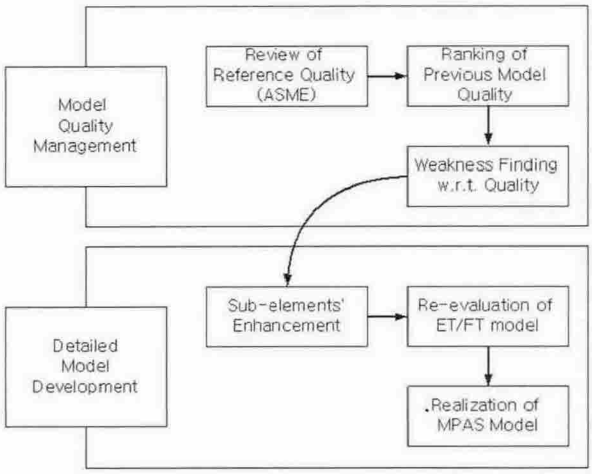

Figure 1. Overall framework for MPAS model development.

2.3. Weakness finding with respect to the quality

We have also tried to improve the quality of the PSA model based on the findings from the previous process. To improve the technical quality of KSNP PSA, following recommendations have been derived:

- The collection & incorporation of Korean reliability data of equipment,
- The analyses of important accident sequences using the best-estimate thermo-hydraulic (T/H) code, and
- The improvements of fault tree models including enhanced CCF (Common Cause Failure) analysis and HRA (Human Reliability Analysis), etc.

By way of the corresponding activities for improving our PSA quality, we thought, the accuracy and consistency of MPAS model will be enhanced a lot.

2.4. Sub-elements' characteristics

There are major five sub-elements that are needed to enhance the model from the previous KSNP PSA.

2.4.1. Initiating event analysis

Generally, risk monitoring for actual NPPs is desired to be done with the configuration control of many mitigating systems. Therefore, there is a need to separate the categories of initiating events considering on failure/rupture locations. In a MPAS model, we have divided the initiating events with 15 of loss of coolant group and 13 of transient group. In particular, each initiating event has been classified using a initiating event impact model. Right now, the potential risk of RCP seal LOCA is reviewed for the selection of new initiating events.

2.4.2. Success criteria analysis

There is no change for mission time in a MPAS model, compared with the traditional Korean PSAs. On the other hand, in T/H success criteria, there are a lot of need to re-analyze in detail. Also, when compared with the traditional Korean PSAs, the updated definition of core damage by using the "ASME PRA Standard" causes a great confusion in deciding whether the mitigating function can be successfully done for specific accident situations.

2.4.3. System analysis

In a case of system modeling, we have 3 categories; detailed, reduced, and undeveloped. Most of system modeling has been done with the detailed model. There is a need to refine the results, because some engineering system analyses were not performed in detail.

2.4.4. Human reliability analysis

Currently, an effort to make a standardized procedure in the field of human reliability analysis has been progressed in Korea. We decided to apply the output of this effort to the MPAS model.

2.4.5. Reliability data analysis

In a case of the determination of initiating event frequencies, we decided to apply the state-of-the-art methodology (both generic and plant-specific) to the MPAS model. For instance, reference values collected for initiating events, e.g. NUREG/CR-5750, can be selected based on the review of applicability to KSNP.

2.5. Re-evaluation of ET/FT model

To make the event tree and/or fault tree (ET/FT) model confidently better than now, we are performing various sensitivity studies for various items, i.e. for reliability data, human error, and major assumptions. It is noted that, in order easily to provide the diverse configuration management during risk monitoring, ET/FT model was already converted to the one-top fault tree model.

3. Current Version and Further Enhancements

The Level 1, Revision 0 MPAS model will comprise plant-specific set of PSA-based risk models that use the one-top linking logic. They will also use a standardized database. Therefore, a standardized and verified input data, especially for initiating event frequencies, equipment performance, and human performance is desirable in preparing MPAS model. Final version of MPAS model will be emerged in early 2005.

The Level 2 MPAS model to evaluate large early release frequency is considered as an item of next R&D projects. The other modeling area, such as a model for evaluating a risk during low-power and shutdown, is not currently considered.

4. Conclusions

The Level 1 MPAS model for providing regulatory risk information is being developed by the KINS-KAERI cooperation. With an effort to make a standardized approach for specific issues and elements, the MPAS model may be useful to any relevant regulatory matters, and can provide a basis in preparation of RIR in Korea.

REFERENCES

[1] ASME, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME RA-S-2002, 2002.

[2] J.-E. Yang, et al., "Evaluation and Improvement of PSA Quality in Korea,"IAEA joint Workshop on PSA Quality Decision Making, Vienna, Austria, February 16, 2004.