Assuring SMART Design Safety through Regulatory PSA Model

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1. Introduction

The technique of Probabilistic Safety Assessment (PSA) has matured to the extent that can provide useful risk information to the regulatory decision-making process, as evidenced by an active pursuit of the U.S. Nuclear Regulatory Commission (NRC) toward Risk-Informed Performance-Based Regulation (RIPBR) in almost every facet of regulatory activities requiring evaluation of safety implications. To take advantage of the PSA technique in the design-certification process of SMART integral reactor, a regulatory PSA model is under development at the Korea Institute of Nuclear Safety (KINS) focusing on internal events at power that may lead to core damage. The objective of this paper is double-fold: 1) discuss the approach used in constructing the level-1 at-power PSA model for internal events; and then 2) briefly present the results from the preliminary PSA model developed in this study [1].

2. Approach to Develop Regulatory PSA Model

SMART (System-Integrated Modular Advanced Reactor) is a 330 MWt reactor under active development by Korea Atomic Energy Research Institute (KAERI) for power generation and seawater desalination [2]. This section briefly describes our methods selected to construct the regulatory PSA model.

1) Identification of Initiating Events – Initiating events that will potentially cause a reactor trip while the SMART reactor is at power will be deductively identified by use of a logic structure such as master logic diagram. The classical list of initiating events for Pressurized Water Reactors (PWRs) will also be utilized to identify any additional initiators that are potentially applicable to SMART. However, since SMART is a first-of-the-kind reactor, further evaluation will be made of whether a reactor scram can be induced by some events in connection with such novel features as the Passive Residual Heat Removal System (PRHRS), or enclosure of the pressurizer, helical steam generators, and reactor coolant pumps inside the reactor pressure vessel.

2) Frequencies of Initiating Events – The frequency data for those initiating events identified above for SMART are primarily taken from the initiator frequency database of NUREG/CR-6928 which is an updated version of NUREG/CR-5750. Since this data is based on the operating experience at PWRs, it is adjusted for SMART reactor where necessary. For instance, the frequency for a steam generator tube rupture (SGTR) at SMART will be adjusted in consideration of the following unique characteristics as compared to the typical steam generators associated with the database: a) helical-coil tube bundle design of the SMART steam generators as opposed to the typical straight-tube designs, b) significant differences in thickness and length of the tubes, and in the differential pressure between the primary and secondary systems; and c) compressive forces as opposed to tensile forces resulting in a larger potential for stress corrosion cracking [3]. Most design characteristics of the SMART steam generator tend to reduce the likelihood of tube rupture as compared to the steam generators of typical large-scale PWRs currently in operation. Further elaboration on this subject will be provided elsewhere.

3) Analysis of Common-Cause Failures – Although Multiple Greek Letter (MGL) model was often used in analyzing common-cause failures (CCFs), there is an increasing tendency of replacing it with Alpha-Factor model because the latter is event-based and as a result more straightforward in evaluating CCF events, and further, simpler in statistical treatment as compared to the former. The Alpha-Factor model is used in this study to quantify the potential common-cause failures in the SMART plant after identifying CCF events based on the CCF experience data in operating light-water reactors that has been recently established by the Idaho National Laboratory (INL) [4].

4) Analysis of Human Reliability – The human error events are identified by first evaluating the applicability of the set of human errors that has been typically applied in PSAs for existing PWRs. Additional human errors are also identified especially in connection with the unique SMART design and operational features. The SPAR-H methodology developed by INL under the auspices of U.S. NRC (see NUREG/CR-6883) is used for estimating human error probabilities because of several advantages such as focus on key performance shaping factors (PSFs), evaluation of the influence of each PSF on the human act with discrete scales, facilitated evaluation of dependency between multiple human error events, etc.

5) Equipment Reliability Database – The aforementioned database of NUREG/CR-6928 also contains unreliability parameters estimated for various types of equipment, and therefore, this industry-average performance data is used as a primary component database in developing the regulatory PSA model.

6) PSA Software Package – The SAPHIRE software package, maintained and continually updated by INL,

has been selected as a primary tool by which the PSA level-1 analysis for SMART plant will be performed, because it represents a proven technology with enhanced user friendliness. Initially released in the name of IRRAS in 1987, SAPHIRE with a companion software tool called GEM provides various features:

- PC-based fault tree and event tree graphical and text editors
- Cutset generation and quantification
- Importance measures and uncertainty modules
- Relational database with cross-referencing features
- External events analysis (e.g., seismic, location transformation)
- Rule-based recovery and end-state analysis
- Basic events templates and process flags
- Changes sets and sensitivity analysis
- Initiating event assessments and condition assessments

3. Preliminary Result of the PSA Evaluation

A preliminary PSA model has been developed in this study using the most widely used 'small event tree-large fault tree' method and the approach discussed above. A special emphasis was placed on the reliability assessment of the PRHRS [5] because of its unique design feature of passive operation. The reliability evaluation of the PRHRS yields a relatively low unavailability of 7.6×10^{-7} primarily because of the redundancy built into the system (i.e., 2 out of 4 success criteria). However, the system unavailability may increase to some extent if the failure mechanisms for the operating passive system (e.g., breakage of natural circulation as a result of stratification, foreign material obstructions, etc.) with latent human errors potentially causing system failure or degradation are more fully accounted for. Although these failure mechanisms are not expected to cause the system unavailability markedly increased, it would have to be made sure, among others, that the PRHRS will continue to operate successfully, once initiated, under all design basis conditions.

In this study the core-damage accident scenarios identified from the event trees and fault trees of the preliminary PSA model were quantified using the SAPHIRE code resulting in a total core-damage frequency (CDF) of 8.4×10^{-7} per year. The two most dominant initiating events were found to be small-break loss of coolant accident (LOCA) and SGTR, contributing approximately 46% and 32% to the total CDF, respectively. In addition, the anticipated transients without scram (ATWS) scenarios also make significant contribution, i.e., 16% of the total CDF. However, considerable change is expected of these risk characteristics for the SMART reactor as the PSA model becomes further refined reflecting specific design details as they become available.

4. Conclusions

Once the regulatory PSA model is more fully developed for the SMART integral reactor, the risk information from the model can be used in evaluating its design safety. Examples include:

- Assessment of the overall risk associated with at-power operation of the SMART reactor focusing on the potential damage to the reactor core
- Investigation of dominant accident scenarios accounting for multiple failures, common-cause failures, human errors, systems interaction, etc
- Identification of vulnerability of the reactor design to specific initiating events, accident sequences, component failures, or human errors

Furthermore, we also envision that the regulatory PSA model could serve as a tool in implementing Risk Informed Regulation (RIR) for various purposes such as: 1) evaluation of technical specification requirements as proposed by the applicant for design certification, 2) determination of the safety significance of operational events occurring at the SMART plant, or 3) assessment of the risk impact associated with inspection findings.

ACKNOWLEDGMENT

This work has been carried out under the Nuclear R&D program supported by the Ministry of Education, Science and Technology, Republic of Korea.

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