Development of PSA Audit Guideline and Regulatory PSA Model for SMART

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Abstract

SMART is under development for dual purposes of power generation and seawater desalination in Korea. It has an integral reactor type and employs advanced design features such as a passive system. For the purpose of regulatory verification to the risk level of SMART, the insights and key issues on the PSA are identified with referring some worldwide safety guides as well as its design characteristics. Regulatory PSA model under the development for the design confirmation and its preliminary result are also described.

Keywords SMART, Regulatory PSA Model, Passive System, Reactor Risk

1. Introduction

The SMART, which stands for System-Integrated Modular Advance Reactor, with a rated thermal power of 330 MW is under development by the Korea Atomic Energy Research Institute (KAERI) for dual purposes of power generation and seawater desalination. It has advanced unique design features such as integral reactor where major components of the primary system (e.g., reactor core, pressurizer, reactor coolant pumps and steam generators) are all enclosed inside of the reactor pressure vessel, and the Passive Residual Heat Removal System (PRHRS) provides a major passive means to remove decay heat by the phenomena of natural circulation. It is noted that the safety advantage of adopting such advanced features should be confirmed through analysis or test. Therefore, it is essential to develop new probabilistic safety assessment (PSA) validation guidance to review the KAERI's PSA results, and needed to get an independent regulatory PSA model to confirm the risk outcomes provided by the KAERI's PSA model.

This paper presents the insights and key issues identified through the development of audit guideline to validate the adequacy of level 1 PSA result for SMART and preliminary evaluation results of regulatory PSA model.

2. Development of PSA Audit Guideline for SMART: Insights and Key Issues

As mentioned above, it is essential to assure technical adequacy of SMART PSA during the process of design certification since SMART employs unique design concepts. For this, the key issues due to the design characteristics of SMART are identified with referring the worldwide PSA standards and requirements for the current Pressurized Water Reactor (PWR). Finally, the audit guideline (draft) for SMART will be developed.

This section presents the insights and key issues identified through the development of audit guideline to validate the adequacy of level 1 PSA result for SMART.

2.1 Initiating Events

1) Since the maximum size of piping for the reactor coolant system in SMART is 50mm, large and medium loss of coolant accident (LOCA) by a break in the piping of the reactor coolant system can be eliminated. However, it should be confirmed whether traditional categorizing approach for LOCA size is applicable to the SMART reactor, or not.

2) SMART is a first-of-the-kind reactor and has unique design features, therefore, it is possible that some unique initiating events may occur at SMART due to its unique design features and failure of support system. And also, it is necessary to identify the new initiating events against SMART unique design features.

3) In general, initiating events identified by the use of logic structure such as master logic diagram and adopting classical list of initiating events for operational PWR are grouped considering safety functions or combinations of system response. Since safety function or system response for SMART can be different with the operational PWR, it should be reviewed that initiating events for SMART can be included in the classical initiating events group in operational PWR.

2.2 System Analysis

1) In particular, the passive system design (especially the PRHRS) poses a considerable challenge to the system reliability analysis because of its uniqueness and lack of the relevant operational experience. The worldwide PSA requirements or guideline for passive system reliability are as follows[1-2];

▪ Innovative ways to structure the search for unexpected conditions that can challenge design assumptions and passive system performance need to be developed and applied to advanced reactor.

▪ It should be checked that passive system behaviour is correctly modelled.

▪ In principle, treatment of the passive safety system in PSA is the same as that of the passive systems, such as accumulators, and of inherent passive safety features, such as natural circulation of reactor coolant when the pumps are not available. However, the reviewers should pay attention that they must, as with active systems, should identify the effectiveness by thermo-hydraulic analysis and by extensive tests.

▪ Deterministic demonstration of effectiveness needs to cover the full range of accident conditions for which they are claimed.

▪ Success performance of passive systems should be demonstrated within a set of boundary conditions which can be ensured by correct system set-up, including the correct configuration of the relevant valves.

▪ Reviewers need to check that the potential for human error is fully accounted in leaving the system in a proper condition, as well as the configuration of all necessary valves which are required to act and any active initiation signals.

▪ Given correct boundary condition and satisfactory demonstration of effectiveness, it may be regarded the system as workable.

According to above requirements, the safety advantage of adopting such advanced features should be confirmed through analysis or test. It should be also confirmed that exact fault tree analysis is performed for the PRHRS in consideration of the worldwide state-of-the-art in this area.

2) In a traditional level-1 PSA, 24 hours has typically been used as a mission time for safety systems under the assumption that once core damage is prevented for 24 hours, extensive core damage will not occur afterwards because the plant will be stabilized in a safe state. However the PRHRS in the SMART design needs to be successfully operated for 36 hours before the reactor coolant system reaches the temperature where the shutdown cooling system can be started for further removal of decay heat. Therefore, reviewers should check the impact of the assumptions (e.g., requirement for operability of PRHRS for 36 hours with subsequent decay heat removal by normal active system).

3. Development of the Regulatory PSA Model for SMART Reactor

The PSA technique has matured to the extent that can provide useful risk information to the regulatory decision-making process. Since 2009, 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 to review the design certification for SMART. In this section, the approach used in developing the regulatory PSA model for SMART, PRHRS fault tree analysis, and the preliminary result are addressed, respectively.

3.1 Approach to Develop Regulatory PSA Model for SMART

This section briefly describes the general considerations to develop regulatory PSA model for SMART.

1) Identification of Initiating Events

Initiating events that will cause a reactor trip while the SMART reactor is at power was deductively identified by use of a logic structure such as master logic diagram. The list of initiating events for Pressurized Water Reactors (PWRs) was also reviewed to identify additional initiators that are applicable to SMART. As a result, some classical initiating events such as large LOCA are eliminated considering the design characteristics of SMART and total 12 initiating events are identified.

2) Frequencies of Initiating Events

The frequencies for identified initiating events were primarily taken from NUREG/CR-6928. Since this data is based on the operating experience at PWRs, it is necessary to be adjusted for SMART reactor. For example, the frequency for a steam generator tube rupture (SGTR) at SMART was 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]. Some of initiating events and those frequencies for SMART is shown in Table 1.

3) Analysis of Common Cause Failures (CCFs)

Because the Alpha-factor model is event-based and as a result more straightforward in evaluating CCF events, and further, simpler in statistical treatment as compared to the Multiple Greek Letter (MGL) model, it is used in this study to quantify the potential CCFs in SMART after identifying CCF events based on the experience data in operating light-water reactors that has been recently established by the Idaho National Laboratory (INL) [4].

Table 1. Consideration on the Initiating event frequencies

IE	Frequency(/ry)	Remarks
General Transient	9.01E-01	1.2 times of NUREG/CR-6928 (Smart is a first-of-the-kind reactor)
SGTR	1.06E-03	Excluding the frequency for SCC in NUREG/CR-6928 (SMART design characteristics)
Small LOCA	5.00E-03	Including the frequency for very small LOCA and SORV

4) Analysis of Human Reliability

For identifying human error events in SMART reactor, the applicability of set of human errors that has been typically applied in PSAs for existing PWRs was reviewed. 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.

3.2 PRHRS Fault Tree Analysis

As mentioned above, SMART reactor employs the PRHRS to remove decay heat. In this section, we discuss our preliminary evaluation of reliability for the PRHRS using fault tree technique.

Figure 1 shows a fault tree developed in this study for unavailability of the PRHRS which consists of four independent trains with 50% of the heat removal capacity for each train. The sub-gate for components failure (i.e., PRHR1-C) models the operation of various valves needed to establish the natural circulation path along with plugging of heat exchangers and pipe rupture, while the sub-gate for natural circulation (i.e., PRHR1-N) models degraded heat transfer, loss of boundary integrity, and high concentration of non-condensable gas. Quantification of the PRHRS fault tree based on the operating data of light water reactors yields a total unavailability of 7.6 E-07, and a couple of representative minimal cutsets (i.e., the first and next dominant cutsets in terms of probability ranking) are:

- PRH-AOV-FO-1O (Outlet AOV of PRHRS Train 1 Fail to Open) * ACP-BAC-LP-480V2 (480V AC Bus 2 Fails)

- PRH-AOV-FO-2I (Inlet AOV of PRHRS Train 2 Fail to Open) * PRH-AOV-FO-3O (Outlet AOV of PRHRS Train 3 Fail to Open) * PRH-AOV-FO-4I (Inlet AOV of PRHRS Train 4 Fail to Open)

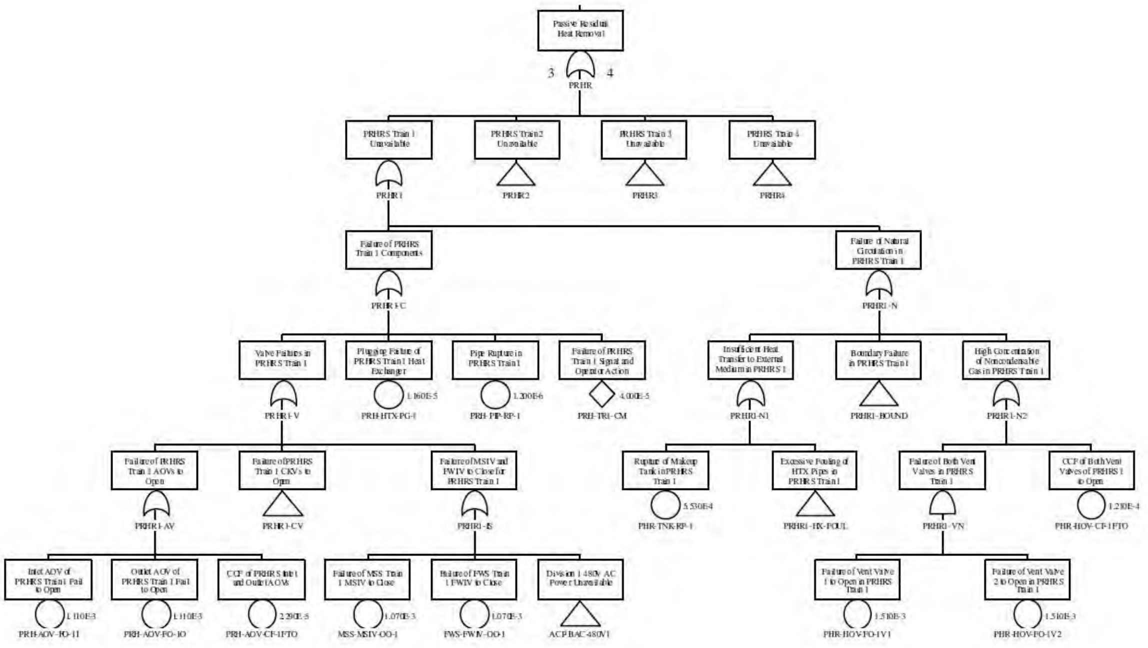


Figure 1. Fault tree for the PRHRS in the regulatory SMART PSA model

The first cutset with a probability of 1.2E-08 includes only two basic events because both train 2 and train 4 are lost by the failure of 480V AC bus 2 power which is needed to close the main steam isolation valve and the feedwater isolation valve. In this analysis, recovery actions to close these valves manually are not given credit. The second cutset with a probability of 1.4 E-09 shows that failure of each of the three PRHRS trains (i.e., train 2, 3 and 4) is caused by an active component, namely, an air-operated valve (AOV) because the flow path for the passive thermal-hydraulic function cannot be established if the AOV fails to open.

The reliability evaluation of the PRHRS yields a relatively low unavailability of 7.6E-07 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, which the PRHRS will continue to operate successfully, once initiated, under all design basis conditions.

3.3 Preliminary Results 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. In this preliminary study, core damage accident scenarios identified from the event trees and fault trees of preliminary PSA model for the SMART reactor were quantified using the SAPHIRE code resulting in a total core damage frequency (CDF) of 4.88E-05 per reactor year (ry). The two most dominant initiating events were found to be loss of feedwater (LOFW) and loss of offsite power (LOOP), contributing approximately 77.9% (CDF = 3.80E-05) and 15.4% (CDF = 7.52E-06) to the total CDF, respectively (see Figure 2). In addition, the general transients scenario also make significant contribution, i.e., 3.7% of the total CDF.

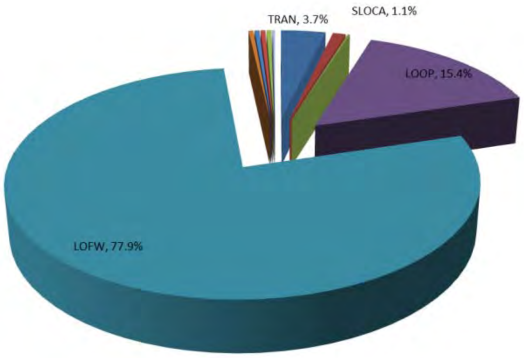


Figure 2. Preliminary importance of initiating events by the regulatory SMART PSA model

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.

Table 2. Significant accident sequences from the preliminary regulatory SMART PSA model

Dominant Scenario	Description	CDF (/ry)	%
LOFW-04	IE-LOFW*/RT*/PRHR*/PSV*/RCPSL*SDC*FAB	2.73E-05	56%
LOFW-03	IE-LOFW*/RT*/PRHR*/PSV*/RCPSL*SDC*/FAB*IRWST	1.07E-05	22%
LOOP-04	IE-LOOP*/RT*/EPS*/PRHR*/PSV*/RCPSL*SDC*FAB	4.94E-06	10%
LOOP-03	IE-LOOP*/RT*/EPS*/PRHR*/PSV*/RCPSL*SDC*/FAB*IRWST	1.94E-06	4%
TRAN-11	IE-TRAN*SGC*/PSV*/RCPSL*/PRHR*SDC*FAB	1.21E-06	2%
Total Contribution			94%

Table 2 shows the main risk significant accident sequences leading to core damage. As the event tree is shown (see Figure 3), the first dominant sequence is that the reactor successfully trips following a loss of feedwater, and the PRHRS removes decay heat for 36 hours. However, core damage occurs because the failure of shutdown cooling and also the feed and bleed operation. The second dominant sequence comes from the combination of successful reactor trip following a loss of feedwater and success of PRHRS function which removes decay heat for 36 hours. In this sequence, the plant was stable with feed and bleed cooling, but long-term cooling failed because the cooling of in-containment refueling water storage tank (IRWST) could not be properly established. The third dominant sequence is that the reactor successfully trips following a loss of offsite power, and the PRHRS removes decay heat for 36 hours. However, core damage occurs due to the same reason of the first sequence. It is note that over 94% of the total CDF is caused by failure of long-term cooling following successful operation of the PRHRS for 36 hours, namely, failure of shutdown cooling system or feed and bleeding.

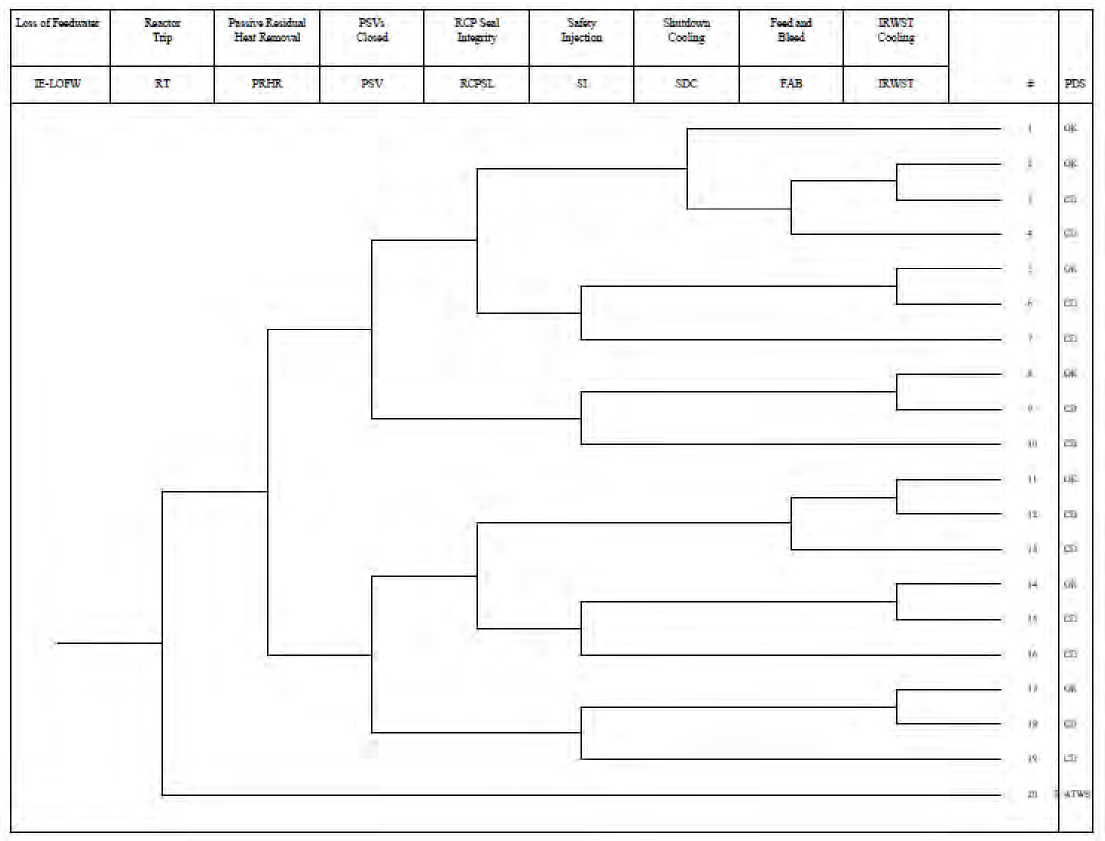


Figure 3. Event tree for the initiating event of loss of feedwater in the preliminary regulatory SMART PSA model

4. Conclusion

With referring available recent safety guides or requirements for assuring new design reactors, a lot of insights and key issues were identified for SMART reactor. Identified and currently-unresolved key issues due to SMART design characteristics are as follows:

- Applicability of classical approach for LOCA size,
- New initiating events according to the SMART unique design features,
- Grouping of initiating events considering unique design features,
- Unavailability evaluation of passive system,
- Extended mission time for the PRHRS.

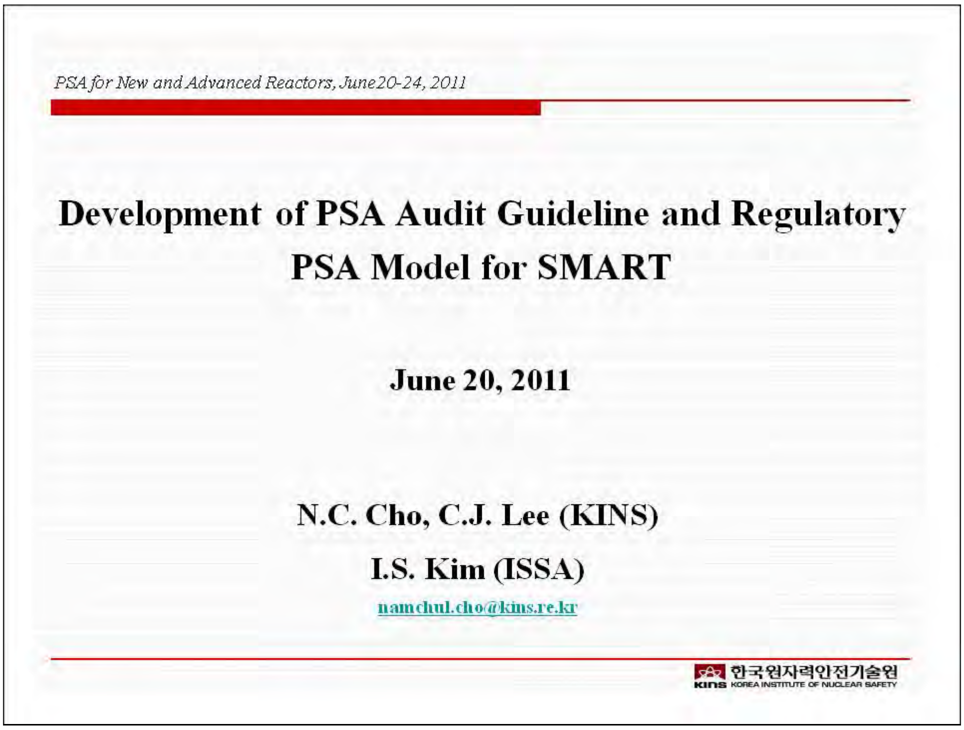
It is noted that, in the design certification by the nuclear regulatory organization, special treatment or documentation may be needed in order to assure the technical adequacy for SMART PSA.

In parallel with this concern, a regulatory PSA model for internal events at power is under development and shows a first preliminary version. This paper presents key approaches to develop regulatory PSA model, to do PRHRS fault tree analysis, and to get the preliminary evaluation result. This PSA model can be utilized not only to review the level 1 PSA results obtained by licensee, but also to support various regulatory applications. We hope that it may ultimately contribute to confirm the quantitative safety level, and find – so correct some vulnerable points of current SMART design and operation concept.

In near future, these key issues and regulatory model will be managed, refined and modified by reflecting any kind of design changes for the SMART reactor since the design is being updated.

5. References

[1] U.S. Nuclear Regulatory Commission, Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, NUREG-1860, Vol. 2, December 2007
[2] Review of Probabilistic Safety Assessments by Regulatory Bodies, IAEA Safety Reports Series No. 25, IAEA, Vienna, 2002
[3] J.C. Jo and M.J. Jhung, Flow-Induced Vibration and Fretting-Wear Predictions of Steam Generator Helical Tubes, Nuclear Engineering and Design, Vol. 238, p. 890, 2008
[4] U.S. Nuclear Regulatory Commission, CCF Parameter Estimations, 2005 update, http://nrcoe.inl.gov/results/CCF/ ParamEst2005/ccfparamest.htm, January 2007



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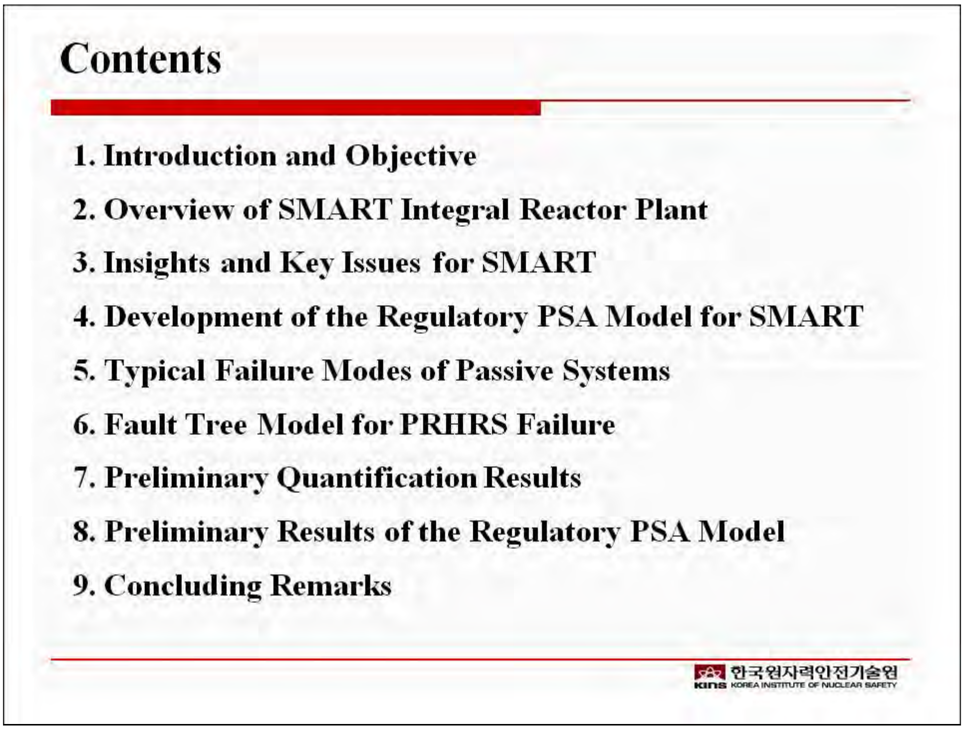

Contents

1. Introduction and Objective
2. Overview of SMART Integral Reactor Plant
3. Insights and Key Issues for SMART
4. Development of the Regulatory PSA Model for SMART
5. Typical Failure Modes of Passive Systems
6. Fault Tree Model for PRHRS Failure
7. Preliminary Quantification Results
8. Preliminary Results of the Regulatory PSA Model
9. Concluding Remarks

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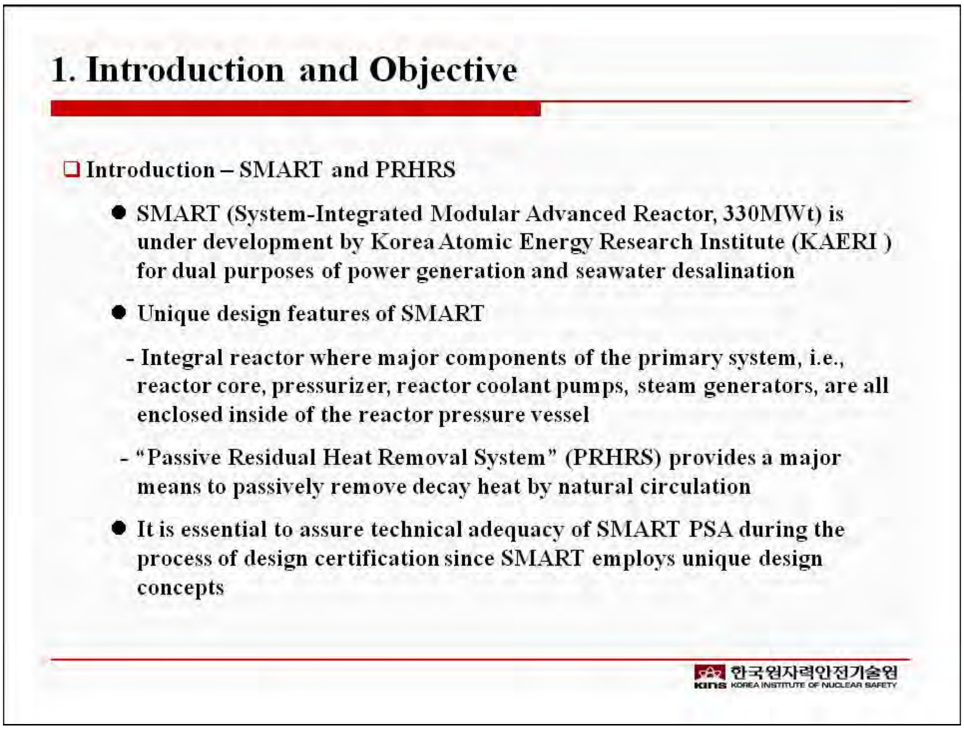

1. Introduction and Objective

❑ Introduction – SMART and PRHRS

- SMART (System-Integrated Modular Advanced Reactor, 330MWt) is under development by Korea Atomic Energy Research Institute (KAERI) for dual purposes of power generation and seawater desalination
- Unique design features of SMART
  - Integral reactor where major components of the primary system, i.e., reactor core, pressurizer, reactor coolant pumps, steam generators, are all enclosed inside of the reactor pressure vessel
  - "Passive Residual Heat Removal System" (PRHRS) provides a major means to passively remove decay heat by natural circulation
- It is essential to assure technical adequacy of SMART PSA during the process of design certification since SMART employs unique design concepts

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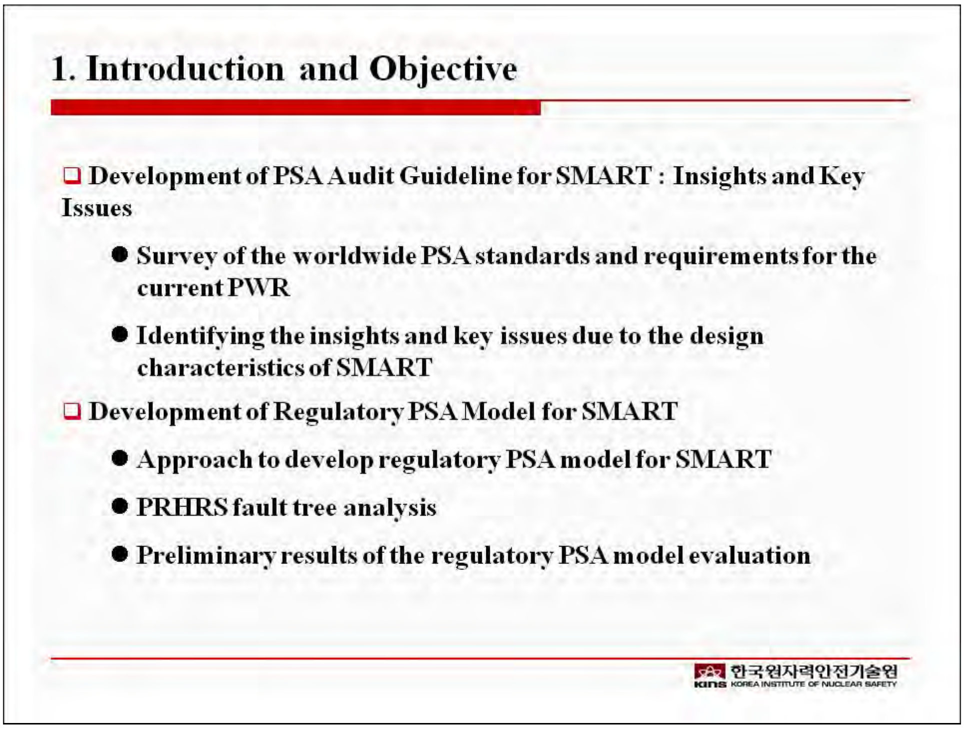

1. Introduction and Objective

❑ Development of PSA Audit Guideline for SMART : Insights and Key Issues

- Survey of the worldwide PSA standards and requirements for the current PWR
- Identifying the insights and key issues due to the design characteristics of SMART

❑ Development of Regulatory PSA Model for SMART

- Approach to develop regulatory PSA model for SMART
- PRHRS fault tree analysis
- Preliminary results of the regulatory PSA model evaluation

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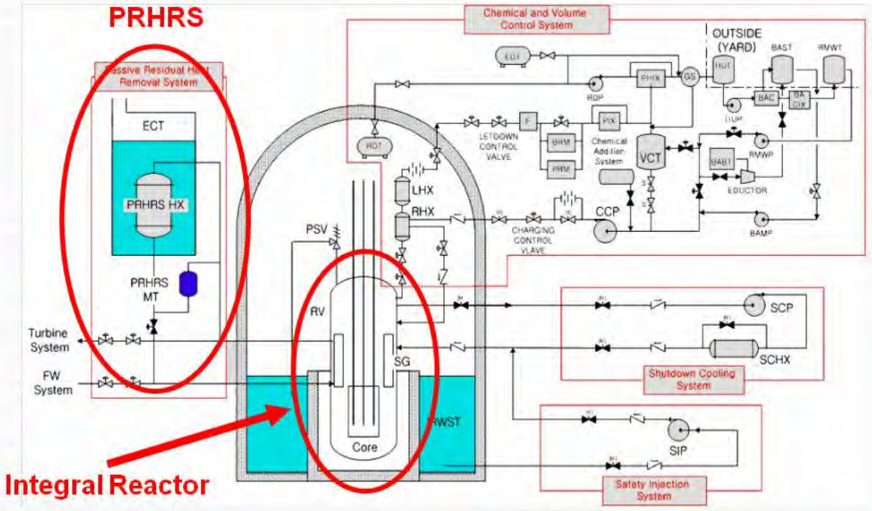
2. Overview of SMART Integral Reactor Plant

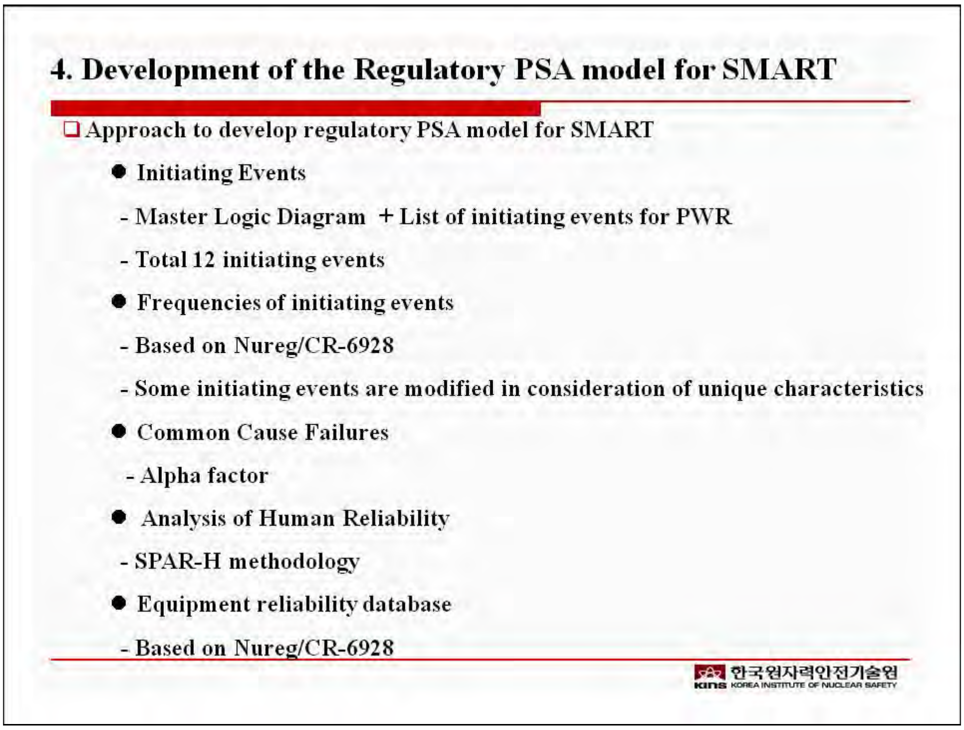


3. Insights and Key Issues for SMART

- ❑ Initiating Events
  - Traditional categorizing approach for LOCA size
  - New initiating events against SMART unique design features
  - Initiating events group
- ❑ System Analysis
  - Treatment of the passive safety system (PRHRS)

    - Demonstrate the effectiveness by thermo-hydraulic analysis and by extensive tests

    - Check the correct system set-up including the correct configuration of the relevant valves
  - Mission time



4. Development of the Regulatory PSA model for SMART

❑ Approach to develop regulatory PSA model for SMART

- Initiating Events
  - Master Logic Diagram + List of initiating events for PWR
  - Total 12 initiating events
- Frequencies of initiating events
  - Based on Nureg/CR-6928
  - Some initiating events are modified in consideration of unique characteristics
- Common Cause Failures
  - Alpha factor
- Analysis of Human Reliability
  - SPAR-H methodology
- Equipment reliability database
  - Based on Nureg/CR-6928

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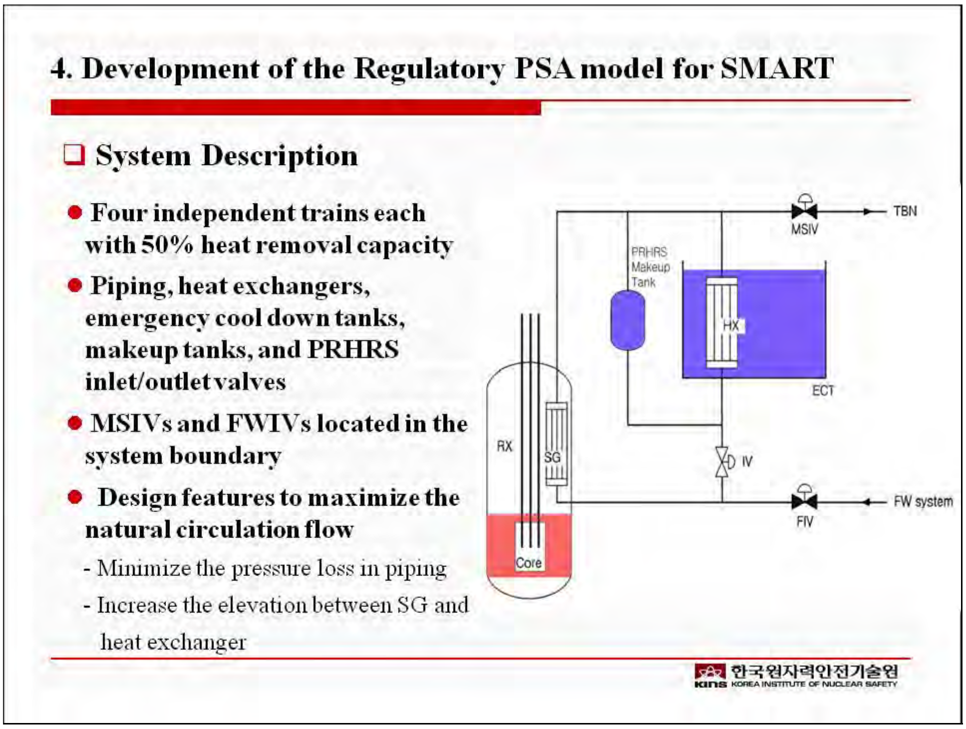

4. Development of the Regulatory PSA model for SMART

❑ **System Description**

- **Four independent trains each with 50% heat removal capacity**
- **Piping, heat exchangers, emergency cool down tanks, makeup tanks, and PRHRS inlet/outlet valves**
- **MSIVs and FWIVs located in the system boundary**
- **Design features to maximize the natural circulation flow**
  - Minimize the pressure loss in piping
  - Increase the elevation between SG and heat exchanger

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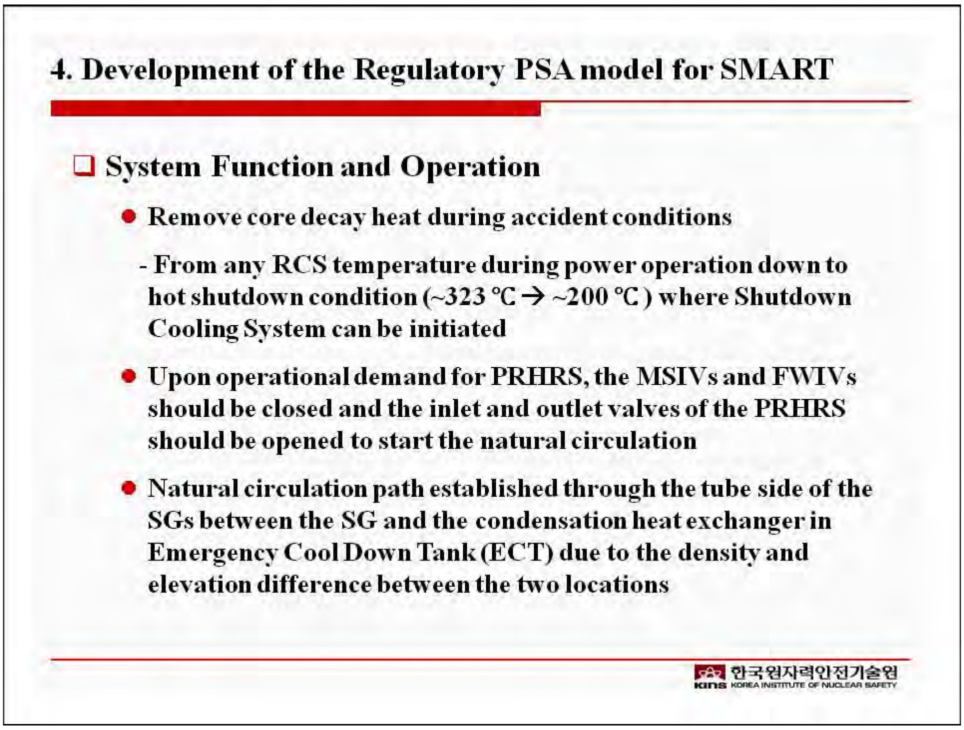

4. Development of the Regulatory PSA model for SMART

❑ **System Function and Operation**

- **Remove core decay heat during accident conditions**

  - From any RCS temperature during power operation down to hot shutdown condition (~323 ℃ → ~200 ℃) where Shutdown Cooling System can be initiated

- **Upon operational demand for PRHRS, the MSIVs and FWIVs should be closed and the inlet and outlet valves of the PRHRS should be opened to start the natural circulation**
- **Natural circulation path established through the tube side of the SGs between the SG and the condensation heat exchanger in Emergency Cool Down Tank (ECT) due to the density and elevation difference between the two locations**

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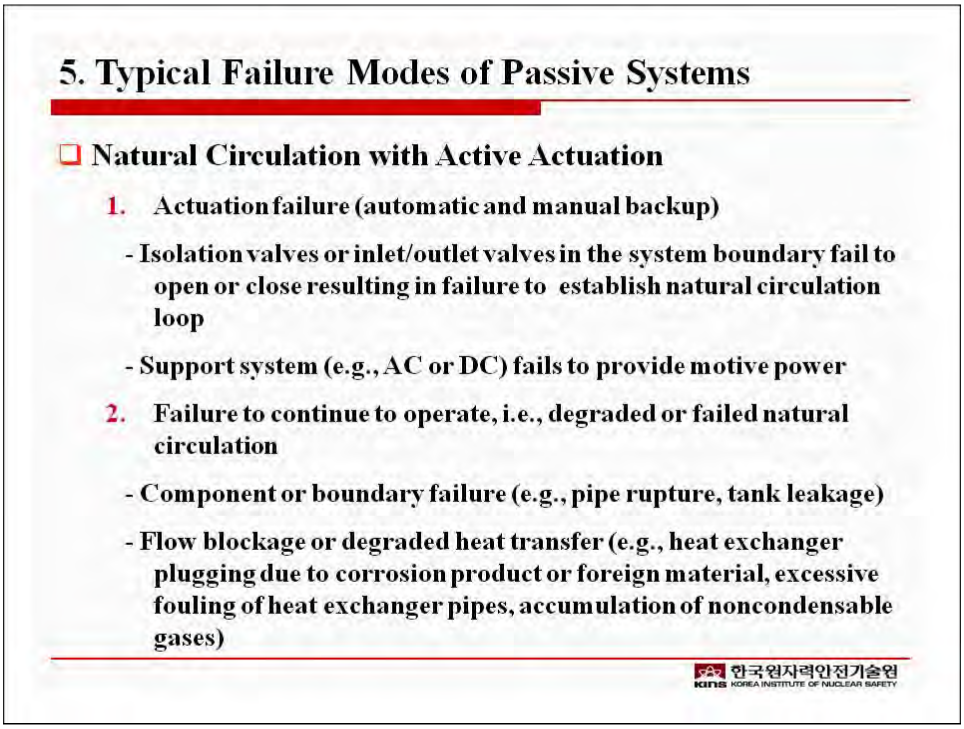

5. Typical Failure Modes of Passive Systems

❑ **Natural Circulation with Active Actuation**

1. **Actuation failure (automatic and manual backup)**

   - Isolation valves or inlet/outlet valves in the system boundary fail to open or close resulting in failure to establish natural circulation loop

   - Support system (e.g., AC or DC) fails to provide motive power

2. **Failure to continue to operate, i.e., degraded or failed natural circulation**

   - Component or boundary failure (e.g., pipe rupture, tank leakage)

   - Flow blockage or degraded heat transfer (e.g., heat exchanger plugging due to corrosion product or foreign material, excessive fouling of heat exchanger pipes, accumulation of noncondensable gases)

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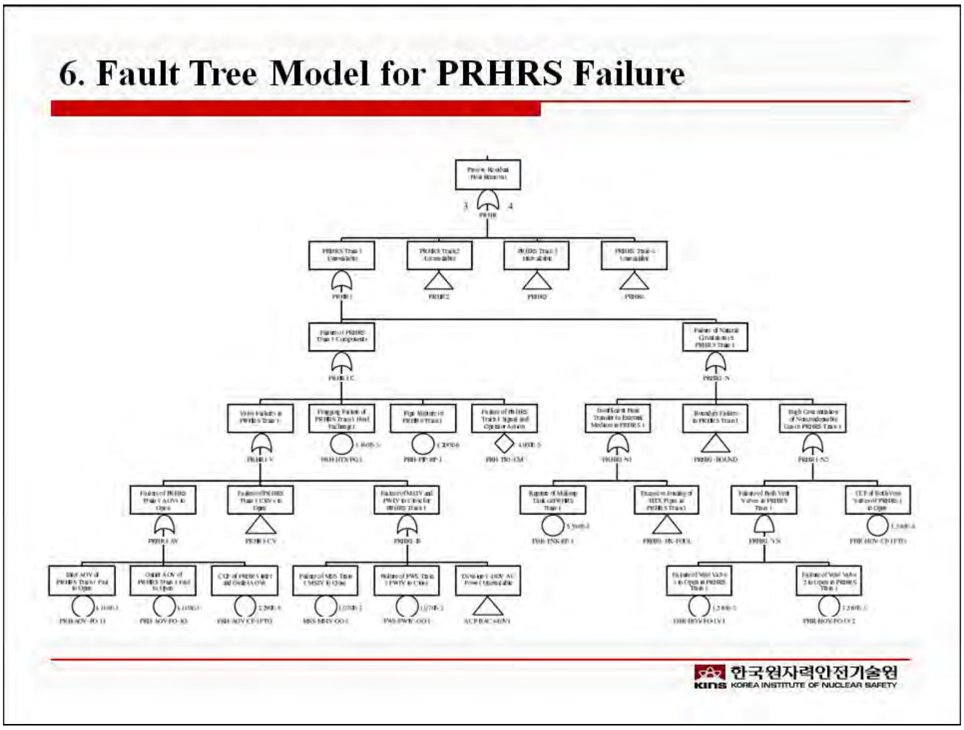

6. Fault Tree Model for PRHRS Failure

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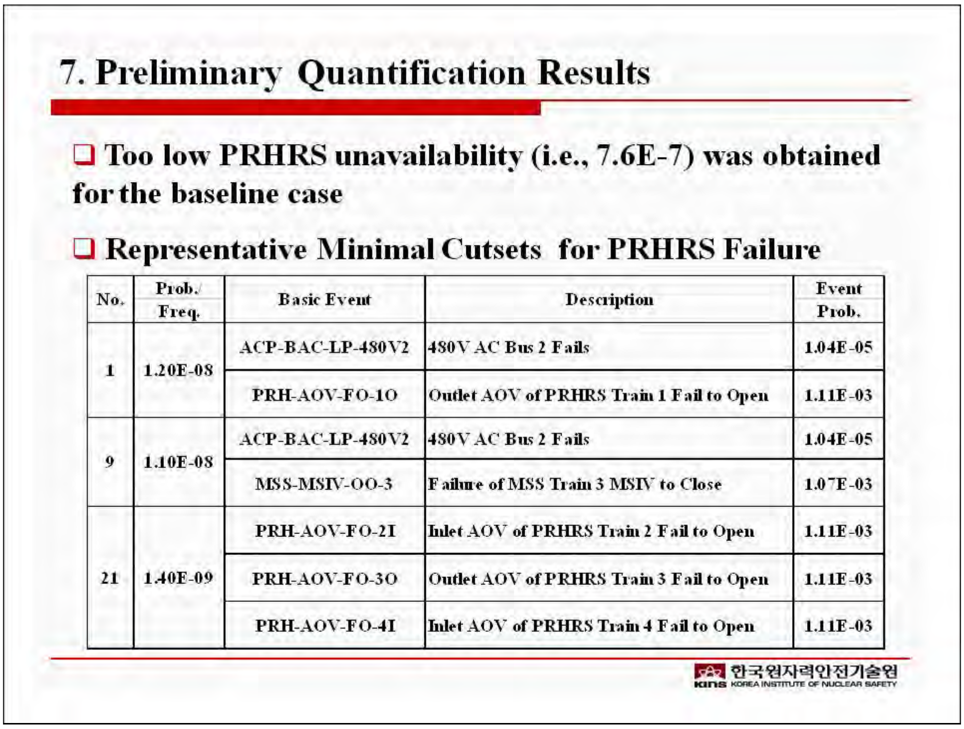

7. Preliminary Quantification Results

- ❑ Too low PRHRS unavailability (i.e., 7.6E-7) was obtained for the baseline case
- ❑ Representative Minimal Cutsets for PRHRS Failure

No.	Prob./ Freq.	Basic Event	Description	Event Prob.
1	1.20E-08	ACP-BAC-LP-480V2	480V AC Bus 2 Fails	1.04E-05
		PRH-AOV-FO-1O	Outlet AOV of PRHRS Train 1 Fail to Open	1.11E-03
9	1.10E-08	ACP-BAC-LP-480V2	480V AC Bus 2 Fails	1.04E-05
		MSS-MSIV-OO-3	Failure of MSS Train 3 MSIV to Close	1.07E-03
21	1.40E-09	PRH-AOV-FO-2I	Inlet AOV of PRHRS Train 2 Fail to Open	1.11E-03
		PRH-AOV-FO-3O	Outlet AOV of PRHRS Train 3 Fail to Open	1.11E-03
		PRH-AOV-FO-4I	Inlet AOV of PRHRS Train 4 Fail to Open	1.11E-03

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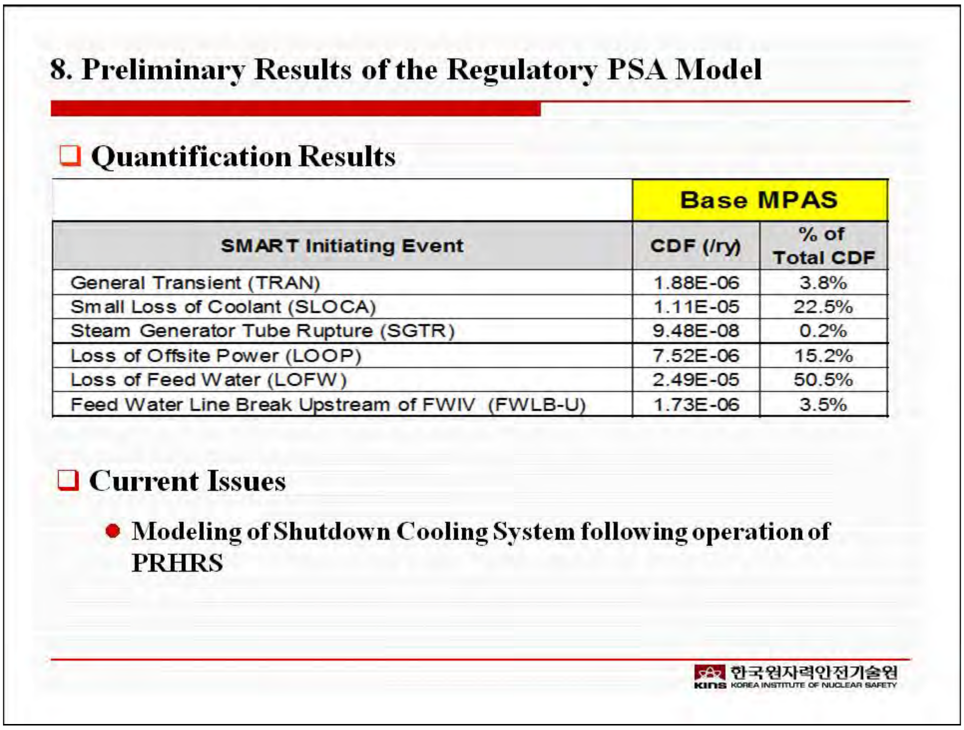

8. Preliminary Results of the Regulatory PSA Model

❑ **Quantification Results**

	Base MPAS	
SMART Initiating Event	**CDF (/ry)**	**% of Total CDF**
General Transient (TRAN)	1.88E-06	3.8%
Small Loss of Coolant (SLOCA)	1.11E-05	22.5%
Steam Generator Tube Rupture (SGTR)	9.48E-08	0.2%
Loss of Offsite Power (LOOP)	7.52E-06	15.2%
Loss of Feed Water (LOFW)	2.49E-05	50.5%
Feed Water Line Break Upstream of FWIV (FWLB-U)	1.73E-06	3.5%

❑ **Current Issues**

- Modeling of Shutdown Cooling System following operation of PRHRS

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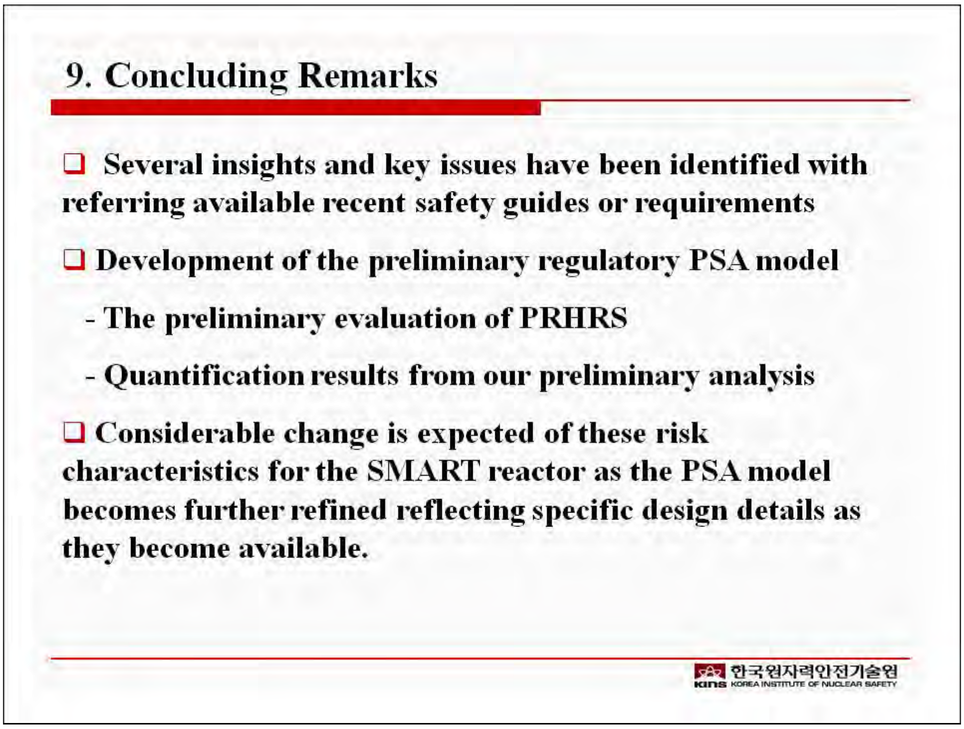

9. Concluding Remarks

❑ Several insights and key issues have been identified with referring available recent safety guides or requirements

❑ Development of the preliminary regulatory PSA model

- The preliminary evaluation of PRHRS
- Quantification results from our preliminary analysis

❑ 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.

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Thank You.