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PSA Technical Issues in Gen-VI Reactors

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1. Introduction

Most world-wide operating commercial nuclear reactors are classified in Generation-II category. The Gen-III reactors have just started to be deployed, and Gen-III+ reactors are at the advanced stage of commercialization. Since the safety and reliability of these reactors have had a good grade, it is widely recognized that the nuclear energy has a crucial role to play in mitigating the ever-increasing world energy needs. In 2000, the U.S. Department of Energy (DOE) launched a new program, called Gen-IV Initiative, to broaden the opportunity of nuclear energy utilization by making further advances in nuclear energy systems design [1].

Recently, Very High Temperature Reactor (VHTR) and Sodium-Cooled Fast Reactor (SFR) among the Gen-IV reactors are being considered in domestic companies. The Probabilistic Safety Assessment (PSA) is one of the key technologies for the safety evaluation and licensing of the VHTR and the SFR. In addition, PSA technology takes charge of the important role for risk-informed design and licensing of Gen-IV reactors, so it has been recognized more importantly [2-3].

In this paper, technical issues of Modular High-Temperature Gas-Cooled Reactor (MHTGR) and Power Reactor Innovative Small Module (PRISM) PSA are identified, and major considerations for the VHTR and SFR PSA review are suggested.

2. Results of the MHTGR PSA Review

The MHTGR conceptual design was submitted by the U.S. DOE, and the staffs of Nuclear Regulatory Committee (NRC) had reviewed this design with emphasis on the unique provisions in accomplishing the key safety functions [4]. After consulting the review results, the technical issues of the MHTGR PSA are summarized as follows:

• The methodology used in its PSA is a traditional event-tree/fault-tree approach, and the uncertainties usually associated with the risk quantification were further exacerbated by the paucity of design details.

• The accident initiators developed for the MHTGR were derived by using a logic diagram. The set of initiators was appropriate but incomplete to describe the potential risk associated with the reactor.

• The success criteria and normal configurations of service water system and circulating water system were not defined.

• The AC power bus loads were not developed.

• Failure modes and data analyses for valves were uncertain because specific types were not given.

• Common-mode and common-cause events were not present explicitly in the models.

• Human error events were too vaguely described to determine whether they were assumed to occur before the event initiation or after.

• There was no list of basic events, so tracing the results of the PSA was restricted.

• Release categories were assigned only to those sequences with frequencies greater than 10^{-8}/year. As core-damage sequences were not developed, only non-core-damage releases were given, because very low unavailability assigned to the Reactor Cavity Cooling System (RCCS) primarily caused potential fuel-failure sequences to be truncated by the 10^{-8}/year cutoff criterion.

3. Results of the PRISM PSA Review

The PRISM conceptual design was submitted by DOE in 1986, and NRC published the Pre-application Safety Evaluation Report (PSER) for the PRISM liquid-metal reactor [5]. Among the contents of PSER, the technical issues of PRISM PSA are summarized:

• Depending on the initiating event, a mean time to recover (MTTR) was estimated. But, the licensee did not document how the MTTRs were estimated.

• The impact of support-system-level failures and system interactions among safety systems, support systems, and other modules had not been assessed in detail because of lack of design detail.

• The PSA missed the detail of data required to substantiate occasional optimistic estimates of system reliability.

• Essential support system failures, system interactions, and human errors were not modeled.

• Common-cause beta factors were assumed small.

• The original PSA did not provide specific sources of data used in the fault trees. It is also believed that some of the initiating event frequencies were underestimated.

• External events other than seismic had not been quantified, nor contributed to the final risk estimates. Seismic analysis was limited to the hazard curve assumed for the GESSAR II site. Fragilities were based on engineering judgment.

• Source-term estimates might be low for some scenarios as a result of extrapolating from oxide fuel to metal fuel.

• Retention of fission products in the metal fuel, sodium pool, cover gas region, and containment dome might be optimistic so needed to be substantiated.

• A mechanistic analysis of the accident sequences had not been performed. Generic assumptions made in the PSA might not accurately represent some of more important accident sequences.

• The safety goal policy statement specifies that mean values should be used when demonstrating compliance with the quantitative health objectives and large release criteria, whereas the PRISM PSA specifies "best estimate." Uncertainties had not been quantified, nor well understood at the conceptual design stage.

• The role of operators was not apparent in the PSA. Credit in the form of operator recovery had been taken, although it is not clear what actions would be taken or if the operators would even be available to perform such actions.

• In order to substantiate the very low risk estimates reported in the PRISM PSA, a greater effort will be needed to achieve reasonable completeness at the lower end of the probability frequency spectrum.

4. Common Technical Issues Derived in the MHTGR and the PRISM

Even though the reactor types are different, there are some technical issues in common as follows:

• The methodology used in the MHTGR and PRISM PSA consisted of the traditional event-tree/fault-tree (ET/FT) approach. However, even though the approach is traditional, discussions for the other adequate methodology for the VHTR and SFR PSA are needed.

• Common-cause failures and human error events were not explicitly presented or poorly assumed in the models. It should be checked that the assumptions are under the proper rationale.

• For tracing and analyzing the results, specific sources of data and list of basic events should be submitted.

• Uncertainty analysis of components having non-specified features is needed.

These common issues should be analyzed and considered in detail when the VHTR and the SFR PSA are reviewed or the standards for reviewing their PSAs are made.

Also, even though it is not described commonly in the MHTGR and the PRISM PSA, a few remarkable issues must be checked:

• There are a few passive systems in the MHTGR and the PRISM. Also, it seems that there are a few passive systems in the VHTR and the SFR. But, the reliability analysis methodology of passive systems is not fully developed yet. So, further consideration on the analysis methodology for the passive system reliability is required.

• Although data used in the PSA is so uncertain that some of the basic initiating event frequencies have been underestimated, and some event sequences are not evaluated because of truncation cutoff, whether their consequences are serious or not. Therefore, a truncation cutoff should be applied carefully.

• Finally, a PSA in the construction stage has a lot of issues which require additional review and questions that unsolved. Therefore, a limited utilization of the PSA should be made at this stage.

5. Conclusions

The SFR and the VHTR among the Gen-VI reactors are selectively developed for future viable energy resources in Korea. Concurrent with this situation, it seems that the review on the PSA results will be done for licensing of those reactors in the near future. In order to support the review, PSA technical issues that are derived from the previous MHTGR and the PRISM PSA review are identified in this study.

We believe that the common technical issues derived from the MHTGR and the PRISM PSA review cases can be utilized to enhance the current Gen-VI PSA technology.

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