Characteristics Analysis of CANDU PSA for Risk-informed Application

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1. Introduction

Recently, the risk informed applications (RIA) have become a worldwide issue of the nuclear industry. In this area, the U.S.A. plays a leading role in developing the present RIA framework [1]. The other countries have adopted and/or modified the RIA framework of the U.S.A. for their own purpose. Nowadays, Korean nuclear industry is trying to introduce the RIA into Korea including the CANDU reactor.

The present RIA framework has been developed for the light water reactors such as PWR (Pressurized Water Reactor) and/or BWR (Boiling Water Reactor) in the U.S.A. So if we want to use this RIA framework for the other types of reactors such as the CANDU reactor, etc., we have to review the applicability of the present RIA framework to the other types of reactors. In this aspect, we have to consider two factors: (1) the definition of risk measures and (2) the used PSA techniques.

In this paper, we have reviewed the characteristics of the CANDU PSA, i.e. Wolsong 2/3/4 PSA [2, 3]. And we have performed the sensitivity analyses to identify the issues to be resolved for the CANDU RIA Framework

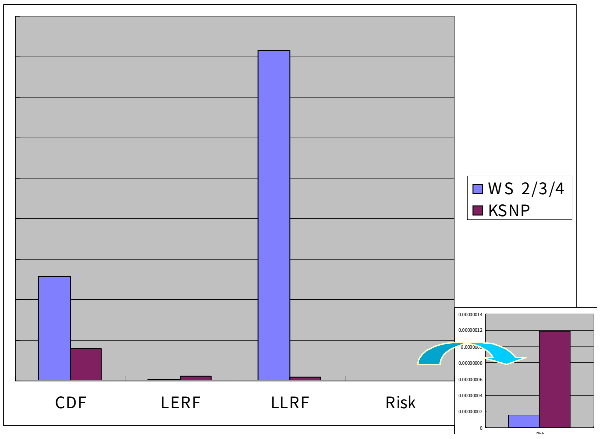
2. Comparison of The Risk Measures

In the present RIA, two risk measures are used: CDF (Core Damage Frequency) and LERF (Large Early Release Frequency) [1].

From the Wolsong 2/3/4 PSA, the concept of CDF is introduced into the CANDU PSA [3]. However, the CANDU reactor has a totally different core structure from that of the PWR. So, the core damage in the CANDU PSA is defined in a different way from that of the PWR. In the PWR PSA, the core damage is defined as the core uncovery, whereas, in the CANDU PSA, the core damage is defined as the multiple fuel channel failures. That is, the physical meaning of core damage in the CANDU PSA is totally different from that in the PWR PSA. On the other hand, the definitions of LERF, risk are the same in the both PSA.

Therefore, it is inappropriate to compare the CDF of PWR and CANDU directly. However, to identify the differences caused by the different definition of core damage, we compared the CDF and other risk measures of a PWR (Korea Standard Nuclear Power Plant: KSNP) and CANDU in Figure 1 [3-5].

As shown in Figure 1, the risk measures of both reactors show quite different patterns. This is mainly due to the different definition of core damage and plant characteristics. This aspect will be discussed in Section 4 in detail.



(LLRF: Large Late Release Frequency)

Figure 1. Comparison of Risk Measures

3. Review of CANDU PSA Characteristics

The PSA provides the basic risk information required in the RIA. A number of different PSA techniques can be used in the various PSA areas. This can result in the different risk profile even for the same reactor. So, in order to make a CANDU RIA framework consistent with the PWR RIA framework, the PSA techniques used in the PWR and CANDU PSA are to be compatible with each other.

In Korea, most PWR PSA have used the similar PSA techniques [5]. However, the Wolsong PSA followed the Canadian practice [2]. So the Wolsong PSA was updated based on the PWR PSA practice in Korea [3]. However, there are still some differences in the PWR and CANDU PSA. So, we performed some sensitivity analyses on Wolsong Level 1 PSA based on the PWR practices. The major differences between the Wolsong and the PWR PSA and the results of sensitivity analyses are summarized in the next sub-sections.

3.1. The system boundary of component

In most PWR PSA data, the instrument and control (I&C) parts are included in the component boundary. For instance, the failure of I&C part related to a pump is included in the failure data of the pump itself. Whereas, in the CANDU PSA, the I&C parts of a pump are separated from the mechanical parts of the pump. The effects of such different system boundary are estimated by the sensitivity analysis. The results showed that there are no big changes in the CDF due to the different definition on the system boundary.

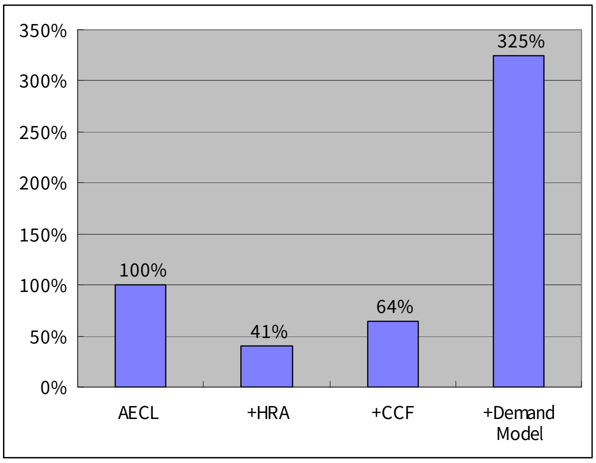


Figure 2 Changes of CDF

3.2. The estimation of human error probability (HEP)

In the Wolsong PSA, the original HEP is estimated by using the paired comparison method. These HEP are revised by using the HRA (Human Reliability Analysis) methodology used in PWR PSA such as ASEP and THERP [3]. The change of CDF due to the revised HEP is shown in Figure 2. The CDF of Wolsong PSA decreases to the 41% of the original CDF value.

3.3. The modeling of common cause failure (CCF)

When AECL performed the Wolsong PSA [2], they assumed that there is no CCF in the CANDU reactor based on the Canadian practice. However, this assumption resulted in the underestimation of the CDF in the CANDU PSA. And the CANDU reactor experiences also show some CCF such as the CCF of relays. So we included the CCF model into the Wolsong PSA by using Multi-Greek Letter (MGL) method [3]. The CDF with the CCF model and the revised HEP becomes 64% of the original CDF value.

3.4. The modeling of dormant failures

In the PWR PSA, the failure of the stand-by systems is modeled by using the failure probability on demand. Whereas, in the CANDU PSA, the same failure is modeled as the $1/2 \times \lambda \times T$ where λ represents the failure rate during the test interval, T. In general, the failure probability on demand has the value of $\lambda \times T$. So when we used the CANDU PSA approach, the failure probability of a stand-by system is underestimated by factor of 0.5 comparing to that of the PWR PSA. The change of CDF due to this difference is shown in Figure 2.

4. Discussions & Conclusions

Nowadays, Korean nuclear industry is trying to introduce the RIA into Korea. For the PWR, the present RIA framework developed by U.S.A. can be adopted with some minor modifications. However, for the CANDU reactor, we may need a new RIA framework that incorporates the characteristics of the CANDU reactor. In order to develop the RIA framework for the CANDU reactor, following two aspects are to be considered: the definition of risk measures such as the CDF and the effects of the unique PSA techniques used in the CANDU PSA.

We have analyzed the characteristics of the CANDU PSA from these points of view. As we can see in Figure 1, the risk measures of both reactors show quite different patterns. So we think that the present RIA criteria of Reference [1] based on the risk measures of PWR cannot be used for the CANDU type reactor directly. And Figure 2 shows that the different PSA techniques can cause big differences in the PSA results.

From above, we can conclude that we need to derive new numerical criteria for Δ CDF, and to define new risk measures instead of LERF for the CANDU RIA. In addition, the PSA of CANDU should be revised for the RIA.

At present stage, it is impossible to derive the appropriate risk measures and criteria for the CANDU RIA. However, we think that the numerical criteria of CDF may be higher than that of PWR. To determine appropriate risk measure and numerical criteria for the CANDU RIA, we need to perform various sensitivity analyses including the Level 2 and 3 PSA of the CANDU reactor.

Acknowledgement

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