Improvement Strategy of Probabilistic Safety Assessment Quality for Domestic CANDU Nuclear Power Plants

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1. Introduction

The purpose of this study is to develop the strategy for quality improvement on the PSA for Wolsong NPPs. To achieve this purpose, any related regulations and requirements to the PSA of Canada has been reviewed. The comparative analysis for the status of all the essential technical elements of CANDU NPP PSA for both Wolsong and Canada has been performed as well. The essential technical elements include the initial event analysis, event tree analysis, system analysis, component reliability database, common cause failures analysis, human reliability analysis, thermal hydraulic analysis, external event analysis (earthquake and flooding), and Level 2 analysis. Based on the established strategy, the revision to the PSA model on full power for Wolsong Unit #1 is currently in progress. By applying this strategy for quality improvement on the PSA to Wolsong NPP, it is expected that the PSA quality for domestic CANDU NPP will be upgraded to the level of corresponding PSA quality for the current domestic PWR NPPs.

2. Methods and Results

This section describes the contents and the results performed to establish the strategy for quality improvement on the PSA for domestic CANDU NPP.

2.1 Regulations and Requirements for PSA in Canada

PSA regulations and requirements for nuclear power plants in Canada described in S-294 [1] and issued by CNSC are as follows:

- Perform a facility specific level 2 PSA for each CANDU NPP in question
- Ensure that the PSA models reflect the plant as built and operated, as closely as achievable and are developed using assumptions and data that are realistic and practical
- Ensure that the level of detail of the PSA is consistent with the NPP testing and configuration management programs
- Ensure that the methodologies and computer codes used in the PSA are consistent with best industry practices
- Include both internal and external events in the PSA
- Include both at power and shutdown states in the PSA
- Include sensitivity analysis, uncertainty analysis and importance measures in the PSA

It is identified that the PSA for domestic CANDU NPP meets all of the above requirements except for the shutdown state. The PSA models for both low power and shutdown states will be developed through the project which is currently in progress by KHNP.

2.2 The Strategy for Quality Improvement on CANDU NPP PSA by Essential Technical Elements

2.2.1 Initial Event Analysis

The IEs have been selected through the following reviews:

- Generic initial events for which safety analysis is required
- Systematic design using master logic diagram for front-line system and related auxiliary system including radioactive substances
- The antecedent PSA results for CANDU NPPs
- Shutdown histories of CANDU NPPs

These methods are identically applied to the initial event analysis for the PWR NPP PSA.

Also, in order to improve the quality of analysis related to the initial event in this study, the experiences from the PSA quality improvement for Kori 3 and 4 [2] and Shin-Kori 1 and 2 [3] were reflected in carrying out Failure Mode Effect Analysis for essential components of the front-line system and the related auxiliary system. In addition, a systematic documentation in grouping processes from the preliminary initial events to the final initial events helped to select the final initial events for the full power PSA for Wolsong 1.

2.2.2 Event Tree Analysis

The event tree for CANDU NPP PSA has been developed using small event tree method similar to the one used in the PWR. Unlike PWR PSA, CANDU PSA classify the status of core damages into few groups and apply separately; largely severe core damage due to the loss of structural integrity and locally limited core damage resulted from some channel failure. Whereas Wolsong NPP’s PSA considers the severe core damage as a core damage and applies accordingly.

In order to improve the quality of event tree analysis for Wolsong 1 in this study, first, the reactor manual shutdown for the initial events was not considered according to the K-HRA methodology in such case that the operator action time for the manual shutdown is too short. Second, in event of where the secondary heat removal treatment by using the main steam safety valve and condenser steam dump valve is not available, the event tree for this was constructed by changing it to the main steam line break from external containment. In addition, in the event that the liquid relief valve fails to close after opening due to the overpressure from the primary heat transport system coinciding with the degasser condenser isolation, the event tree for this was constructed by changing it to small LOCA from D_2O storage tank. Based on these, more realistic and practical event sequences were analyzed. Unlike Wolsong 2, 3 and 4, the conditional signal for the emergency core coolant system of Wolsong 1 has no variables due to the continuous coolant low pressure. Like as the liquid relief valve failure event, the reactor manual shutdown by the operator was considered in the emergency core coolant system injection due to the coolant low pressure. In the events of the pressure tube and calrandria tube break, the event tree for this was constructed by determining the availability of heat removal using the moderator system.

2.2.3 System Analysis

In Canada, the reliability analysis traditionally has been used for verifying the design from the designing stage, so the fault trees for the front-line system and auxiliary system have been developed using the large fault tree method. So the large fault tree method was applied to the PSA for Wolsong 2, 3, and 4 performed by AECL on 1996 [4]. In comparison with the small fault tree of U.S.A., the large fault tree developed in Canada had the differences in making their large fault trees in more detail by specifying the component boundary. Since the PSA for Wolsong 1 has been performed by KEPCO E&C on 2003 [5], the small fault tree method in the manner of PWR has been developed and then the small fault tree method has been applied to the PSA for Wolsong NPPs. Besides, the small fault tree method can provide the useful information in terms of understanding the weak point and producing the improvements on the equipment against the severe events. For this reason, it is considered as a proper method for Wolsong NPPs PSA in carrying out the government policy against severe events. Therefore, the small fault tree method was used in revising the PSA model on full power for Wolsong 1 in this study.

2.2.4 Component Reliability Database

For Canada, they have developed and applied the component reliability database by specifying the component boundary resulting from applying the large fault tree methods. The PSA for Wolsong 2, 3 and 4 performed by AECL was also developed with the large fault tree method. Therefore, DARA database by OPG, Canada was used for Wolsong 2, 3 and 4 [6]. Whereas EPRI-URD database [7] has been used as a general database after KEPCO E&C performed the PSA for Wolsong 1 by applying the small fault tree method that had been used in the PSA for domestic PWR NPPs.

In order to improve the quality on the database related to the component reliability in this study, NUREG/CR-6928 [8] reflecting operational experience of NPPs of U.S.A. was used as a general database. The plant generic failure probability computed by PRinS system of KHNP reflecting operational experience of Wolsong NPPs was used as a component reliability database for Wolsong 1.

2.2.5 Common Cause Failures Analysis

At first the common cause failures were not considered in analyzing the reliability in Canada, however, later on they were introduced by referring the PSA techniques developed overseas. For this reason, the common cause failures were not considered in the PSA for Wolsong 2, 3 and 4 performed by AECL. Afterwards AECL, a design and engineering company from Canada, had applied Unified Partial Method (UPM) to analyze the common cause failures. Since there was no data analysis for the common cause failures for CANDU NPPs was performed, the common cause failures data from U.S.A. was used. MGL (Multiple Greek Letter) method has been applied to the PSAs for domestic CANDU NPPs and the common cause failures data from EPRI-URD database of U.S.A. has been used for domestic CANDU NPPs PSA.

Generally, allowing for the symmetry of the probability for the common cause failures event, failure probability Q_t for a random one component in *m* line is as follows:

$$Q_t = Q_1 + {}_{m-1}C_1Q_2 + {}_{m-1}C_2Q_3 \cdots$$
$$= \sum_{k=1}^{m} {}_{m-1}C_{k-1}Q_k$$

Here Q_k is the probability of a basic event involving *k* specific components in a common cause component group of size *m*

In the above function, the representative common cause failure assessment models used in computing Q_k are the α-factor model and the MGL (Multi Greek Letter) model. In this study, in order to improve the quality of the related common cause failure analysis, the α-factor model was used due to the relatively high advantages in uncertainty analysis when revising the full power PSA model for Wolsong 1. The probability for the common cause failure will be estimated depending on the test method of the components which are involving in the common cause failures. There are non-staggered test and staggered test depending on the test period and method. The probabilities for these are expressed as follows:

Non-Staggered Test :

$$Q_k^{NS} = (k / {}_{m-1}C_{k-1})(\alpha_k / \alpha_t) * Q_t = (CCF_k) * Q_t$$

Staggered Test:

$$Q_k^{S} = (\alpha_k / {}_{m-1}C_{k-1}) * Q_t = (CCF_k) * Q_t$$

$\alpha_k = n_k/(\sum_{j=1}^{m} n_j)$

$\alpha_t = \sum_{k=1}^{m}[k * \alpha_k]$

For CCF_k values from the above function, the most recent data from NUREG/CR-5497 [9] published in 2007 was used as a basic database for the common cause failure event analysis.

2.2.6 Human Reliability Analysis

THERP methodology [10] and ASEP methodology [11] have been used for human reliability analysis in performing CANDU NPPs PSA for both domestic and Canada. The existing analysis methods did not specify the detailed procedures or standards and assessment rules and guidelines. There has even been the uncertainty in human reliability resulting from the subjectivity of the involved analysts. In order to meet the standard of ASME PRA Standard Category II in regards to the quality standard of human reliability analysis, to minimize the uncertainty due to the difference between analysts, to improve the consistency in the analysis, and to distinguish the differences between error cases, K-HRA methodology [12] was developed by KAERI in 2005. Based on THERP and ASEP methodologies, K-HRA method systematized the analysis procedures, standard rules, and guidelines.

In this study, in order to improve the quality of the human reliability analysis, when revising the PSA model on full power for Wolsong 1, the new K-HRA methodology was applied to provide the unified analysis methods and to minimize the subjectivity of the involving analyst. The standardized assessment rules and guidelines were used for defining the human error event, screening analysis, task analysis, quantitative analysis, and documentation.

2.2.7 Thermal Hydraulic Analysis

The thermal hydraulic analysis in the PSA is carried out to establish the successful criteria of each header of event sequence used in the initial event; to verify the models and assumptions for the PSA analysis, and to estimate the allowable time for operator actions which is used in the human reliability analysis. Especially, the required allowable time for the operator action is the essential input data for the human reliability analysis. Thermal hydraulic behavior analysis is required to obtain the changes in progress variables and allowable time for operator actions after the initial event. The thermal hydraulic analysis is also required to clarify the basis and background of the allowable time for operator actions which were presented in the existing PSA report on full power and on continuous operation for Wolsong Unit #1. By performing the thermal hydraulic analysis for all the possible initial events, the existing values were verified and re-assessed. CATHENA code [13], which is the safety analysis code for Wolsong 1, was used to perform the thermal hydraulic analysis for level 1 internal events.

2.2.8 External Event Analysis (Earthquake and Flooding)

AECL has been applying the Seismic Margin Analysis (SMA) method [14] of EPRI for seismic event analysis. CNSC, the regulatory agency in Canada, had clarified that the preferred method for seismic event analysis is a PSA-based SMA method. Accordingly, a PSA-based SMA was used to analyze the seismic event for PSA for Wolsong 1.

The procedures for analyzing the flooding for both domestic and Canada consist of; understanding the source of the flooding, defining the flooding zone, assessing the flooding occurrence frequency, understanding the target components by the flooding zone for the PSA, assessing the flooding spreading, developing the scenarios accounting for the flooding protection facilities and operator actions, and quantitative progress. AECL has been using the old data like WASH-1400 [15] as data to assess the flooding occurrence frequency.

In this study, in order to improve the quality of external event analysis for Wolsong 1, models for the flooding and the earthquake were revised by reflecting the results of PSA on full power.

The seismic event analysis for Wolsong 1 used a PSA-based SMA method to perform the seismic fragility assessment on each of the structures, systems, and components, and then the review for the initial event occurrence for the components which are vulnerable to the earthquake and the analysis of event sequences were accordingly performed. Thus, the lists of components which are vulnerable to the earthquake and event sequences were produced. The results showed that HCLPF history is more than 0.3g.

For the flooding analysis for Wolsong 1, the piping break frequency analysis, which had been performed within the target area for the detail analysis, was applied to be performing in the extended target area for the quantitative screening analysis as the screening analysis process. The calculation for the piping break frequency presented in 59 SDM-5 [16] is expressed as follows:

$$FC = (L/D) * 8.8E\text{-}08/YEAR$$

Where, the piping break frequency (FC) is expressed as pipe length (L), pipe diameter (D), it is the empirical equation produced by the accumulated data from the practical operational experience.

2.2.9 Level 2 Analysis

Level 2 analyses for both domestic and Canada have been performed through; understanding the features of the containment function, analyzing the PDS event sequences, analyzing the containment event tree including the containment bypass event, and analyzing the radioactive source terms. For analyzing the severe event sequences, MAAP-CANDU code is used in Canada and ISAAC code in Korea.

In this study, in order to improve the quality on Level 2 analysis, containment filtering and ventilation system (CFVS), as an item recently added to design change, was reflected to prevent the containment from damaging due to the overpressure when severe event occurs. CFVS is equipped with the depressurization function and filtering function

3. Conclusions

As a result of this study, the strategy for quality improvement on the PSA for domestic CANDU NPP was established by reviewing the regulations and the requirements related to the PSA for Canada and the comparative analysis for the status of all the essential technical elements (initial event analysis, event tree analysis, system analysis, component reliability database, common cause failures analysis, human reliability analysis, thermal hydraulic analysis, external event analysis {earthquake and flooding}, and Level 2 analysis) of CANDU NPP PSA for both domestic and Canada. In addition, during the course of this study, the list of the technical elements available to apply to CANDU NPP was produced by the review of the experiences by the technical elements from the quality improvement on domestic PSA (Kori 3, 4 and Shin-Kori 1, 2) and then it was applied to establish the strategy for quality improvement on the PSA. Based on the established strategy, the revision to the PSA model on full power for Wolsong 1 is currently in progress. By applying this strategy for quality improvement on the PSA to domestic CANDU NPP, it is expected that the PSA quality for domestic CANDU NPP will be upgraded to the level of corresponding PSA quality for the current domestic PWR.

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