Development and Preliminary Application of the Methodologies for Evaluating Severe Accident Management Program

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Abstract

Nuclear power plant, generally, has been designed under defense in depth concept and safety evaluation is concentrated on those concepts in design. The most important aspect of defense in depth concept in view of severe accident is accident management and the prevention and mitigation of nuclear power plant are generally reflected on traditional Level 2 PSA. In this study, the methodology frame work is suggested using combining accident management, Level 2 PSA and severe accident researches. This methodology frame work can be an integrated evaluation technique for the assessment of existing and potential severe accident management (SAM) measures.

I. Introduction

Severe accident management (SAM) strategies to terminate or mitigate severe accidents has been currently developed and implemented at nuclear power plants (NPPs) world-wide. A basic understanding of the plant capabilities and limitations during severe accidents is normally achieved through a plant-specific probabilistic safety assessment (PSA). Invariably the PSA is further extended to examine engineering options to further enhance the plant capabilities in the mitigation of severe accidents.

This study is concerned with the further development of integrated models for the assessment of existing and potential severe accident management measures including operator actions. The brief summary of the methodology for evaluating the severe accident management program based upon probabilistic safety assessment (PSA) are provided and sample application was performed.

II. Theoretical Background

II.1 Korean Severe Accident Legislation

After the TMI (Three Miles Island) nuclear accident in the United States, severe accidents have emerged as an important problem that can arise in non-virtual reality. The follow-up action of the TMI nuclear power plant accident, i.e., the requirements for the US severe accidents, was reflected in the domestic nuclear power plant by reviewing the necessity and improvement of application to the domestic nuclear power plant. However, the follow-up actions seems not to be inconsistent and systematic approaches because they had been pursued on a case-by-case basis without a consistent domestic regulatory principle for severe accidents.

In 2001, a severe accident policy was announced, and regulation principles for severe accidents were established in Korea. Therefore, it was required to carry out probabilistic safety assessment (PSA), development of risk monitoring system (RIMS) and severe accident management plan.

Immediately after the Fukushima nuclear power plant accident, the Nuclear Safety and Security Commission (NSSC) decided to carry out a systematic safety inspection on domestic nuclear power plants. During a month, those inspection processes had been conducted for large-scale natural disasters in nuclear power plants, research reactors and nuclear cycle facilities that were operating. The major area in the inspection were earthquake and tsunami, offsite/onsite electrical power and cooling system, severe accidents, emergency response system and nuclear power plant in long-term operation after severe accident. As a results, 50 safety improvement measures are suggested, in order to ensure safe operation of nuclear power plants even if the worst natural disasters occur through safety inspections.

Severe accident safety regulations that were imposed by administrative orders had weak legal basis, such as the declaration of a severe accident policy statement and the request for utility implementation request. Therefore, on June 23, 2016, the Nuclear Safety Act was amended to clearly define the accident management regulatory requirements including the management of severe accidents.

II.2 Previous studies related accident management program evaluation

Even if severe accidents occur, accident management program should have been made to minimize the release of radioactive material into the environment and restore the plant to a safe state. During severe accidents, accident management program generally plays important role to prevention and mitigation of accidents. On the other hand, the efforts to evaluate or investigate the effectiveness of AMP has been not extensively studied because AMP is closely related to severe accident phenomena and Level 2 PSA, and the combination methods of those three parts were difficult.

First study was performed by J.H.Scobel, et al (1998), and they suggested new methodologies to perform severe accident management of AP600 which was designed by Westinghouse. They introduced risk oriented accident analysis methodology (ROAAM) to evaluate severe accident management and this application was illustrative only, because having been conducted in the closing phases of certification review, and prior to the completion of the PRA, it does not constitute a formal submittal of the AP600 project to the USNRC. This study would be a demonstrative analysis and investigate the applicability of the methodology, named Integrated ROAAM.

Another study was performed by M.L. Ang (2001), and they performed the development and demonstration of integrated models for the evaluation of severe accident management strategies, named SAMEM. This study is concerned with the further development of integrated models for the assessment of existing and potential severe accident management (SAM) measures. This paper provides a brief summary of these models, based on Probabilistic Safety Assessment (PSA) methods and the Risk Oriented Accident Analysis Methodology (ROAAM) approach, and their application to a number of case studies spanning both preventive and mitigative accident management regimes. In the course of this study it became evident that the starting point to guide the selection of methodology and any further improvement is the intended application. The application of an integrated ROAAM approach led to the implementation, at the Loviisa NPP, of a hydrogen mitigation strategy, which requires substantial plant modifications. A revised level 2 PSA model was applied to the Sizewell B NPP to assess the feasibility of the in-vessel retention strategy. Similarly the application of PSA based models was extended to the Barseback and Ringhals 2 NPPs to improve the emergency operating procedures, notably actions related to manual operations. A human reliability analysis based on the Human Cognitive Reliability (HCR) and Technique for Human Error Rate (THERP) models was applied to a case study addressing secondary and primary bleed and feed procedures. Some aspects pertinent to the quantification of severe accident phenomena were further examined in this project. A comparison of the applications of PSA based approach and ROAAM to two severe accident issues, such as hydrogen combustion and in-vessel retention, was made. A general conclusion is that there is no requirement for further major development of the PSA and ROAAM methodologies in the modelling of SAM strategies for a variety of applications as far as the technical aspects are concerned.

III. Methodology and Results

The severe accident management effectiveness evaluation methodology is performed using PSA and ROAAM, and the execution procedure is performed as shown in Fig 1. The main concept of this study is to verify and evaluate the effectiveness of the existing severe accident management strategies performed by operator using probabilistic manner.

III.1 Selection of Event Sequences

The event sequences to be analyzed were selected to be the accident sequences that resulted in the containment failure - which should be controlled by the severe accident management. The analysis results of the reactor containment integrity analysis for the target nuclear power plant were used to select the accident type and the accident occurrence condition that the reactor containment failure probability was more than 1%

of the all occurrence frequencies (2.48E-7 / RY). Small Loss of Coolant Accident (SLOCA) was chosen as initiating events for this analysis. In final paper, Station Blackout (SBO) will be included.

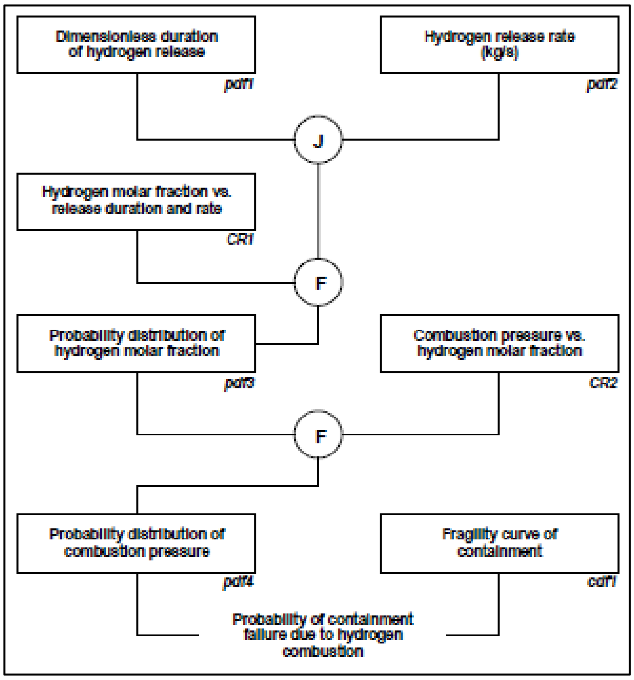

Fig.1 Probabilistic framework for containment failure due to hydrogen combustion

III.2 Identification of Severe Accident Prevention and Mitigation Function

Reactor Containment Failure Accident Sequences include Severe Accident prevention function and mitigation function. Therefore, these functions are separated to reconstruct the event sequences. Severe Accident prevention function is a function to prevent event progression from initiating event to core damage. There are operator actions that use engineered safety feature according to emergency operating procedure. Severe Accident mitigation functions consist of operator actions that utilize the Severe Accident mitigation feature to mitigate event progression from core damage to Reactor containment failure. (Fig.2)

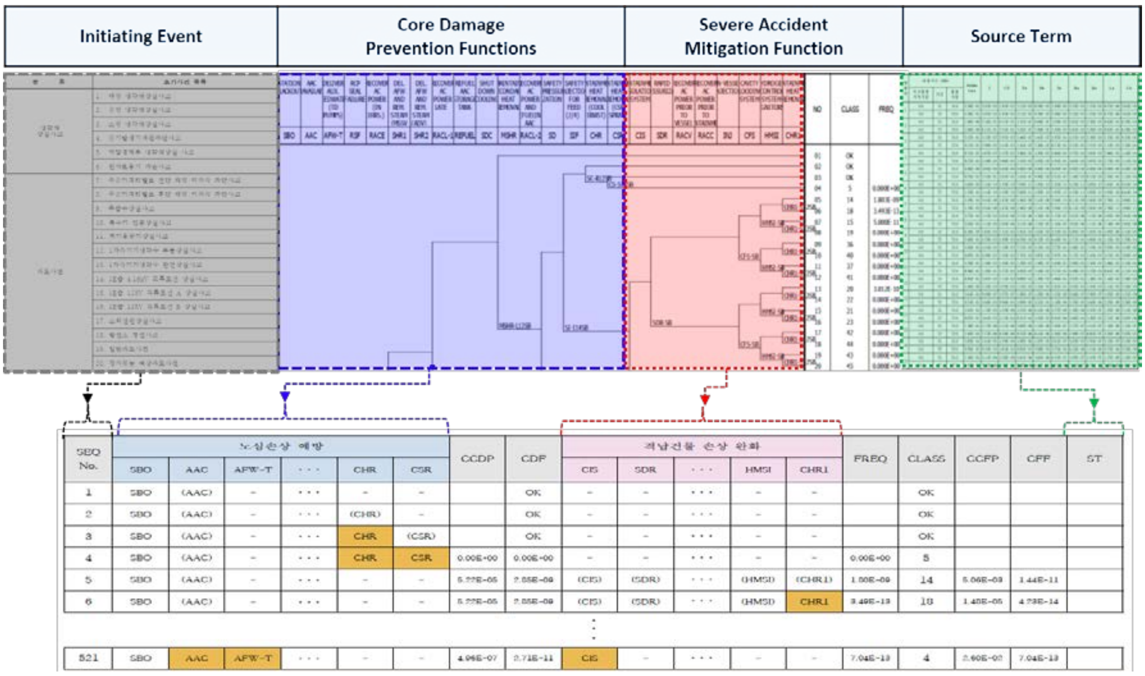

Fig.2. Identification of Severe Accident Prevention and Mitigation Function from Level 2 PSA

III.3 APR1400 Application Results

This methodologies had been applied to APR1400 which was designed by Korea Hydro and Nuclear Company (KHNP). The APR-1400 is an evolutionary Advanced Light Water Reactor which is based on the previous OPR-1000 design. Under Korean conditions, the reactor produced 1455MWe gross electrical power with a thermal power capacity of 3983 MWt (4000MWt nominal). As severe accident prevention and mitigation features, APR1400 has cavity flooding system (CFS), passive auto-catalytic recombiner (PAR), emergency containment spray backup system (ECSBS) and alternate diesel generator (A-DG).

Firstly, the existing level 2 PSA process is reconstructed by removing the plant damage status to conserve the information of the ignored accident sequences which have low CDF (Core Damage Frequency), but high CCFP (Conditional Containment Failure Probability). Korean level 2 PSA adopted the grouping logic in level 1 end states connection part, named plant damage grouping, which is one of the generally accepted PSA method. Secondly, all end state sequences of level 2 PSA are displayed on the chart of CCFP vs. CDF, to check the effectiveness of the severe accident management strategies easily at glance. The procedures of this step are as follows. The sequences of level 2 PSA are plotted on the CCFP vs. CDF at first. The location of the point on the chart is changed if a strategy is applied and the direction moved from the original position indicates the qualitative effectiveness of the strategy. Finally, source terms are analyzed for the representative accident sequences. In this analysis, the source terms are grouped with 17 categories and their release fractions are calculated using the contribution ratio of containment failure modes from the PSA results. And the release fractions are plotted on the other chart by containment failure frequencies.

The methodology was applied to the SBO (Station Blackout) sequence analysis of APR1400 as an example. Some recommendations have already been introduced and the prioritization of accident management strategies were suggested. This kind of methodologies can be used to various area, for example, future nuclear power plant design, accident management program training, backfiting severe accident mitigation measure to operating reactors, etc. As future works, the quantitative success possibilities of severe accident management strategies are going to be analyzed using MELCOR and uncertainty analysis.

The probabilistic importance chart of the failure probability of the severe accident prevention function and the failure probability of the mitigation function was developed and the results are shown in Fig3. The x axis of the probabilistic importance chart is the conditional containment failure probability (CCFP) and the y axis is the core damage frequencies (CDF). The sensitivity results of containment heat removal under SLOCA is shown in Fig.4.

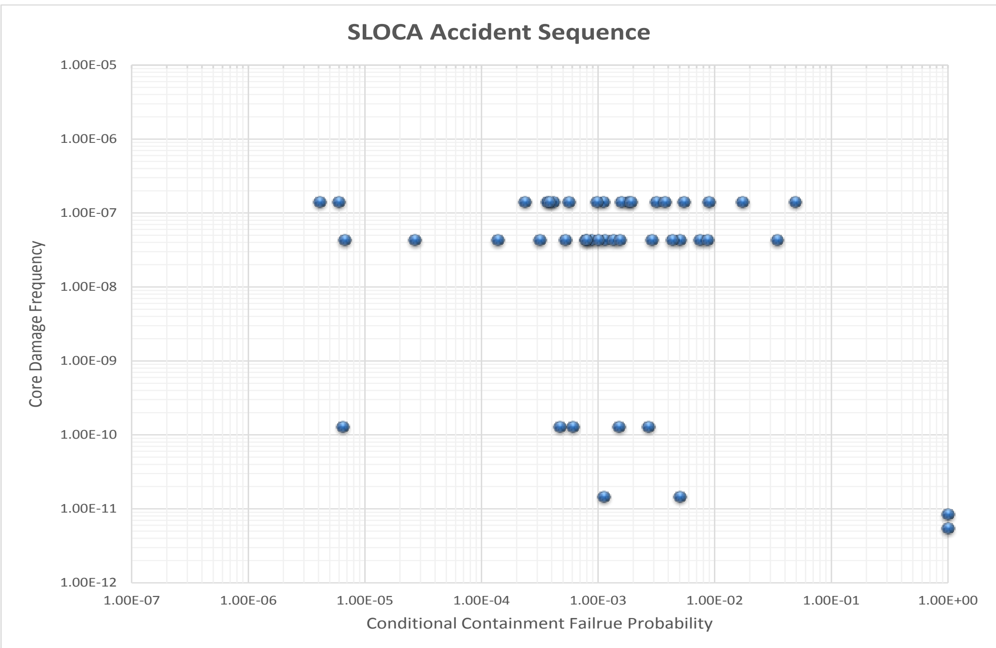

Fig.3. APR1400 SLOCA Results

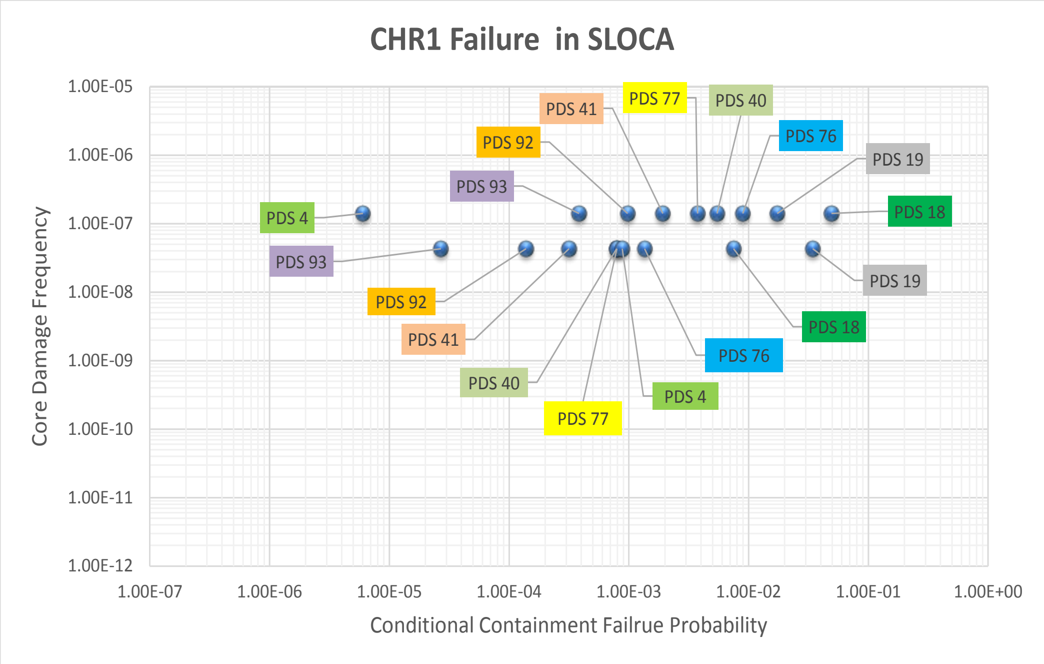

Fig.4. APR1400 SLOCA Containment Heat Removal Sensitivity Results

For SBO (Station Blackout) sequences, the sensitivity studies were performed for the strategies relevant to ‘late containment spray failure (CSLATE)’ and ‘ex-vessel debris cooling(DBCOOL)’. Figure 5 shows the cases for CSLATE success (colored square) and failure (gray triangle). The figure shows that the late containment spray is very important accident management strategy to prevent containment failure.

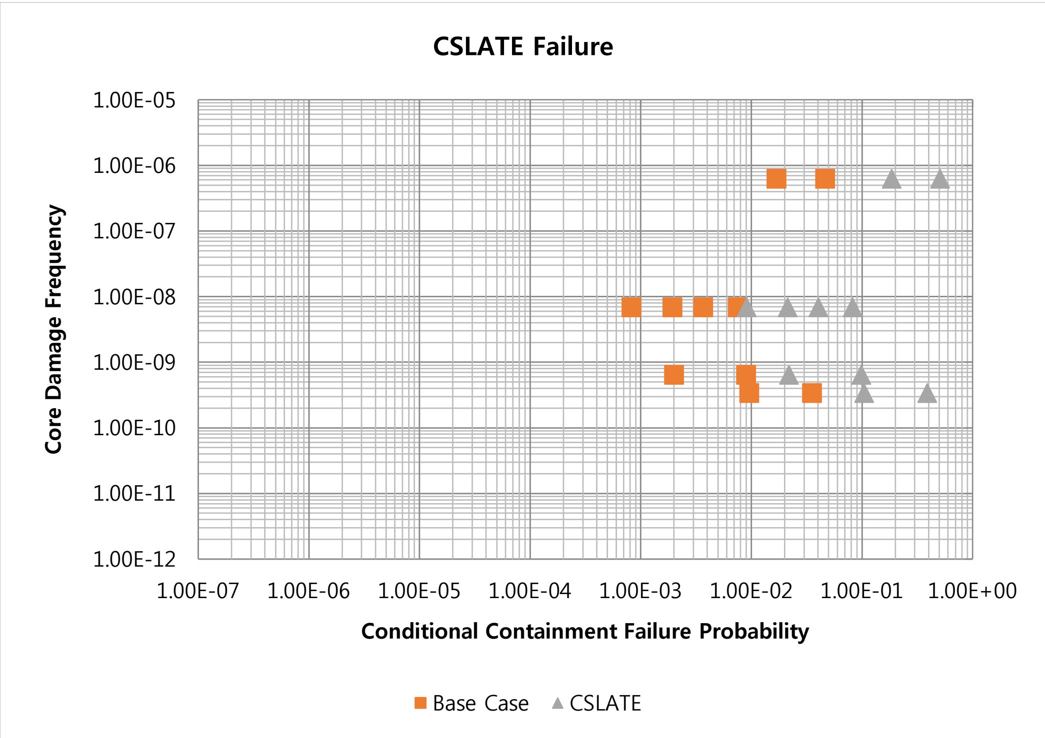

Fig.5. APR1400 SBO Late Containment Spray Sensitivity Results

In the case of ex-vessel debris cooling for SBO sequences, the sensitivity study results shows in Figure 6. Figure 6 shows the cases for DBCOOL success (colored square) and failure (gray triangle). As shown in Figure 6, the Ex-vessel Debris Cooling is relatively less important than CSLATE accident management strategy to prevent containment failure.

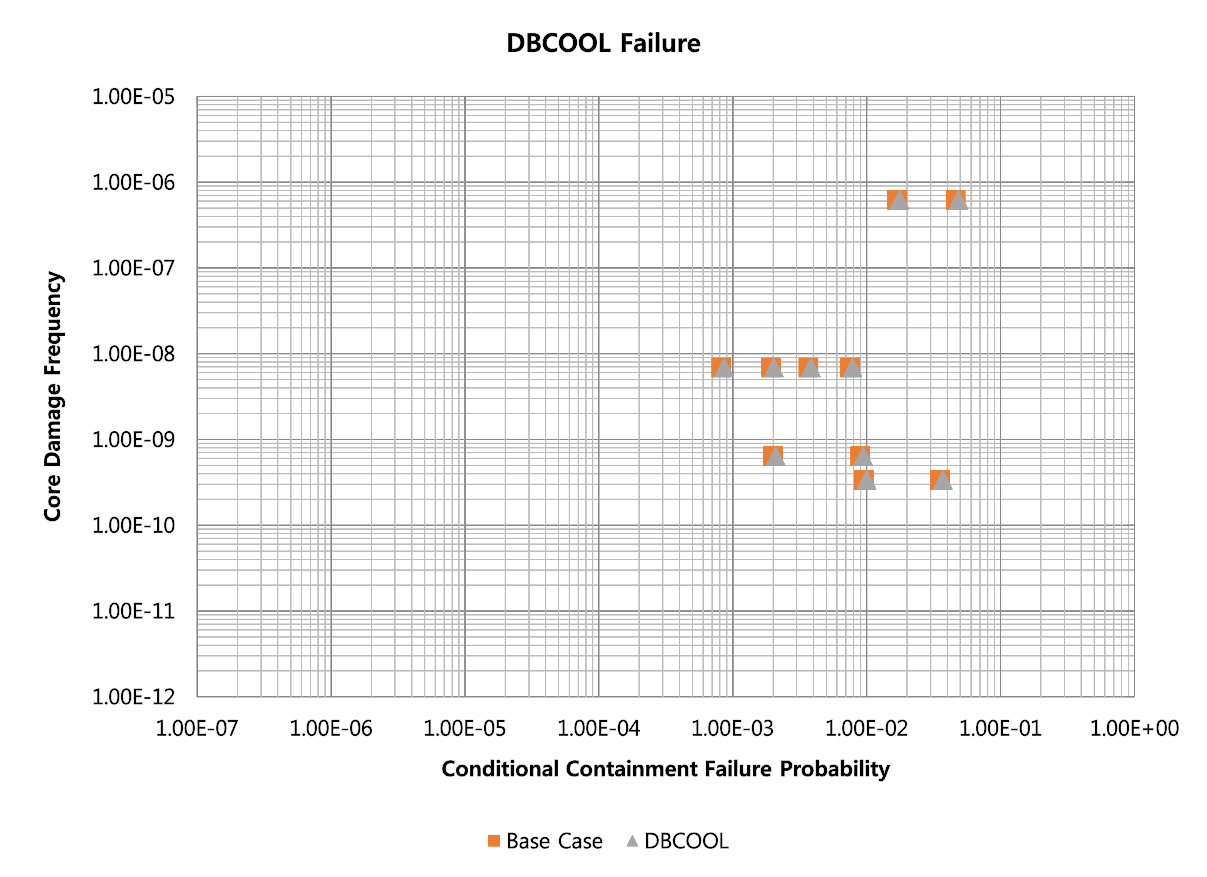

Fig.6. APR1400 SBO Ex-vessel Debris Cooling Sensitivity Results

Comparing two strategies sensitivity studies, the prioritization of various accident management strategies can be possible and this result can be used to operating nuclear power plant equipment improvement for severe accident mitigation.

IV. Concluding Remarks & Future Work

In this paper, the effective evaluation methodologies for Severe Accident Management strategies were proposed and the evaluation procedure for this methodology was established. As future research items, the followings will be performed.

1. In this paper, CET(Containment Event Tree) and DET(Decomposition Event Tree) models were implemented in this framework only for Station Blackout and will be implemented for the whole plant damage states bins.

2. Severe accident management evaluation methodologies developed in this paper will be extend to develop an accident radiological source term analysis that is released after containment failure. Finally, the chart showed in Fig. 3 and 4 will be changed three axis chart (x-axis is CDF, y-axis CCFP, and z-axis source term).

3. Success probabilities of DET branches related to severe accident phenomena will be quantified by

severe accident analysis computer code using uncertainty analysis method (ex. MELCOR, MAAP, etc.)

4. Success probabilities of DET branches, related to operator actions will be quantified, and the prioritization table of those actions will provided.

References

J.H.Scobel, et al(1998), Application of the risk oriented accident analysis methodology (ROAAM)to severe accident management in the AP600 advanced light water reactor

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