Severe Accident Scenario Uncertainty Analysis using the Dynamic Event Tree Method

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Abstract: Several types of uncertainties exist during the simulation of a severe accident. These may result from incomplete knowledge about the plant systems, accident progression and oversimplified numerical models. Among them, parameter uncertainty can be treated via Monte-Carlo-sampling-based methods. Besides, to tackle the severe accident scenario uncertainty, we must resort to advanced dynamic probabilistic risk assessment (PRA) methods. In this paper, authors reviewed the previous dynamic PRA methods and tools, and then performed a preliminary scenario uncertainty analysis, by using an integrated severe accident code (THALES2) and a scenario generator (RAPID, risk assessment with plant interactive dynamics), both being developed at JAEA. THALES2 is a fast-running code for the simulation of severe accident progression and source term in light water reactors. Typical scenarios of station-blackout(SBO)-initiated accidents in boiling water reactors (BWRs) are generated and simulated, with the coupling process. The dynamic event tree (DET) method is applied to consider the stochastic uncertainties during the scenario progression. Major groups of SBO sequences with the similar accident characteristics can be found. To provide a reference value for risk, a conditional core damage frequency is calculated accordingly. This is a preliminary analysis, as the first attempt, for severe accident scenario uncertainty quantification at JAEA, and further DPRA researches are in progress.

Keywords: Dynamic PRA, Severe Accident, Uncertainty, Dynamic Event Tree, System Interaction, THALES2, RAPID

1 INTRODUCTION

Uncertainties exist during nuclear reactor severe accident analysis because reality is ever more complex than any numerical model. Uncertainties may result from indefinite setting of severe accident scenarios, oversimplified models, uncertain input parameters, and other reasons such as incomplete knowledge of systems, undesirable errors during numerical analysis, and so forth. We are trying to apply the dynamic probabilistic risk assessment (PRA) methods to the severe accident uncertainty analysis, and then provide a more realistic risk quantification for the operation of a nuclear power plant.

Dynamic PRA identifies possible results of an accident and assigns probabilities to them, as what traditional PRA already does. Moreover, dynamic PRA provides a more straightforward way to evaluate the risk, with the computational simulation of accident consequences and the consideration of uncertainties, by coupling with integrated severe accident codes. It provides a platform to care for dependencies among plant systems, which may be ignored using integrated severe accidents codes only, human behavior, accident processes (the physical and chemical changes that take place during an accident), emergency preparedness for consequence mitigation the off-site consequences of accidents, and how external events can cause accidents, etc. To some extent, dynamic PRA overcomes traits in traditional PRA analysis, such as the expert-dependent setting of accident sequence, incomplete consideration of accident progressions, separate treatment of scenario generation and severe accident simulation, and so forth.

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The development of dynamic PRA has a long history which dates back to 1980s [1]. Related methods and tools are widely used for risk quantification and uncertainty analysis. These include DETAM (dynamic event tree analysis method) [2][3], DYLAM (dynamic logical analytical methodology) [4], ADS (the accident dynamics simulator) [5], ADAPT (the analysis of dynamic accident progression trees) [6][9], MCDET (Monte Carlo dynamic event tree)[10][11], RAVEN (risk analysis virtual environment) [12][13], SCAIS (simulation codes system for integrated safety assessment) [14], and PyCATSHOO (Pythonic object-oriented hybrid stochastic automata) [15][16]. The dynamic PRA is also treated as IDPSA (integrated deterministic and probabilistic safety assessment), and reviews and applications can be found in the references [17][18]. Under the framework of dynamic PRA or IDPSA, many methods can lead to a more realistic risk assessment as well as a reasonable treatment of uncertainties.

The DET method is one of the most common methods for scenario generation. Starting from an initiating event, the evolution of interested events can be determined by the DET method, for example, it can be decided by pre-assigned probability distributions, or by the intermediate outputs from the integrated severe accident code when dependencies are required to be modeled. The main idea of this method is to let the severe accident code determine the pathway of an accident scenario within a probabilistic environment. The method allows different accident scenarios to be “spawned” and simulated. The simulation result provides reliable estimate of accident consequence and the sampling-based scheme makes the conclusion of occurrence probability straightforward.

At Japan Atomic Energy Agency (JAEA), we have previously investigated and developed methods of parameter uncertainty and sensitivity analyses for severe accident simulation [19][20]. Parameter uncertainty analysis can make clear only part of uncertainties in simulation results. To extend the treatment of uncertainties during severe accident analysis, in recent years, we are trying to investigate and combine the dynamic PRA theories and methods to simulation tools. A tool, RAPID (Risk Assessment with Plant Interactive Dynamics), for dynamic PRA using the dynamic event tree method has been developed and then coupled to the integrated severe accident code, THALES2.

In this paper, as a tentative study on the DPRA of nuclear power plants, we use the DET method to quantify the stochastic uncertainties existing in the typical station blackout (SBO) accident of a BWR4/Mark-I nuclear power plant. The uncertainties are resulted from the uncertain failure status as well as the thermal-hydraulics-based functioning of some reactor auxiliary systems, which is also known as the “interaction” between different systems.

Section 2 provides a literature review of dynamic PRA methods and tools. Section 3 illustrates the design of the JAEA tool for accident scenario generation. Section 4 introduces the computational process for scenario uncertainty quantification by using the THALES2 code and the scenario generator, RAPID. Section 5 summarizes the results of current risk assessments.

2 LITERATURE REVIEW OF THE DPRA RESEARCH

Since the dynamic PRA research is experiencing a high-speed development, authors summarize the currently available tools and their applications to nuclear power plant risk assessment, based on the independent investigation of journal or conference publications. The Figure 1 shows the history of dynamic PRA development and events are marked when tools or methods are invented. Some of important tools and events are listed based on the authors’ understanding of the dynamic PRA research. After methodologies of PRA widely applied to nuclear power plant risk quantification, the dynamic PRA research aimed to fill the gap between probabilistic risk modeling and deterministic accident simulation.

In the paper of 1983 [1], the authors (G. Apostolakis and T.L Chu) modeled time-dependent transitions between accident sequences due to operator intervention and the failure of systems. The analysis pointed out one of limitations of the static event tree/fault tree methodology. After then, a number of methods and tools have been developed. The paper of N. Siu [3] discussed and reviewed methods for dynamic system analysis to overcome the weakness of the static event tree/fault tree methodology. The papers [21][22] discussed about the probabilistic reactor dynamics to supplement the deterministic reactor analysis when stochastic changes happened to the complex system. The paper [5] described the toll of ADS for full scale dynamic PRA. Factors such as plant thermal-hydraulics behavior, safety systems response and operator interactions are explicitly accounted and ADS uses discrete dynamic event tree as the main accident scenario modeling approach. Papers [10][11] introduced the MCDET method to achieve a more realistic modeling and analysis of complex system dynamics in the framework of probabilistic safety analyses. The MCDET is a combination of Monte Carlo simulation and the discrete dynamic event tree approach. Papers [6][7] presented the tool of ADAPT for automated accident progression event trees generation using the concept of dynamic event tress. The ADAPT code, for example, is coupled with MELCOR for dynamic PRA of accident scenarios. RAVEN is a software framework able to perform parametric and stochastic analysis based on the response of complex system codes [13]. It provides dynamic risk analysis capabilities the thermohydraulic codes, such as RELAP-7, MAAP, etc. In Japan, a dynamic tool, using a method which is named as continuous Markov chain Monte Carlo, has been developed and used to perform risk analysis for sodium-cooled faster breeder reactors. It is a standalone tool, by which the plant dynamics is simulated by using a simple meta-model, and stochastic scenario branching is treated by using event tree and Monte Carlo sampling [23][24]. A summary of widely applied tools is provided in Table 1. Because the dynamic event tree method can treat the plant dependencies with coupling with severe accident codes, so it motivates us to perform the scenario uncertainty analysis.

Figure 1: The development of DPRA tools

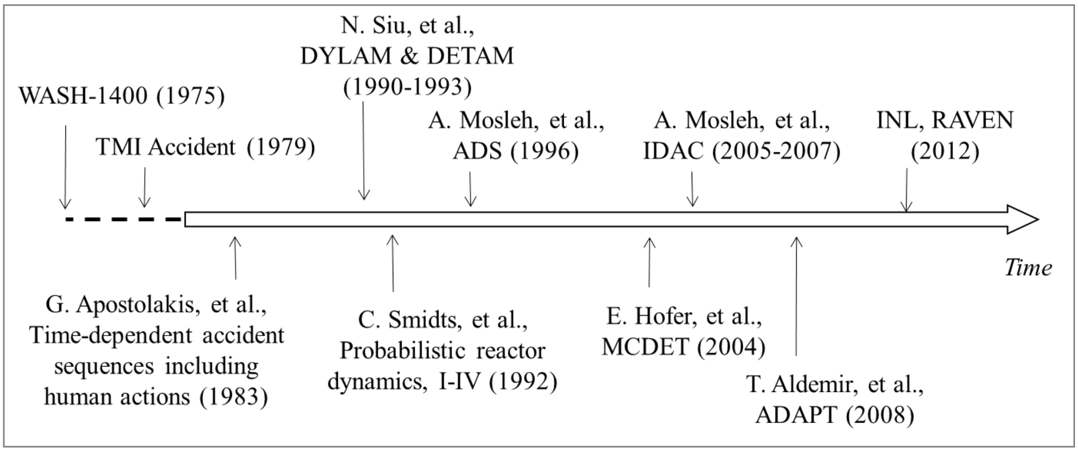


Table 1: Summary of well-known DPRA tools

	ADS-IDAC	MCDET	ADAPT	RAVEN	SCAIS	PyCATSHOO
Developed at	UMd (UCLA now), U.S.	GRS, Germany	OSU, U.S. SNL, U.S.	INL, U.S.	CSN, Spain	EDF, France
Started from[a]	1993	2002	2008	2012	2008	2013
Representative scenario generation approaches	Discrete dynamic event tree (DDET)	DET with Monte Carlo	DDET	DDET, hybrid DET, adaptive DET, etc.	DDET	Monte Carlo
Coupled thermal-hydraulic and SA codes	RELAP, TRACE, MELOR, etc.	MELCOR, etc.	MELCOR, etc.	RELAP, MAAP, etc.	MAAP	Standalone code for sodium-cooled fast reactor DPRA

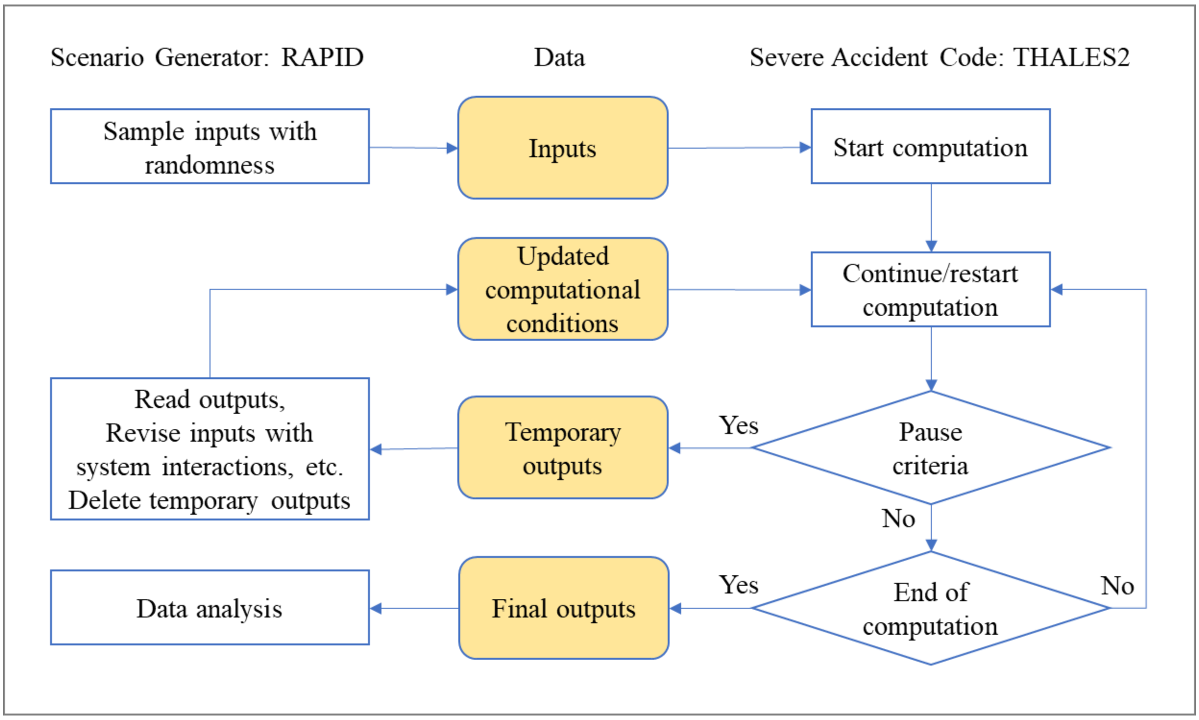
a: Start years are summarized according to available journal or conference publications.

3 DESIGN OF THE SCENARIO GENERATOR USING DYNAMIC EVENT TREE AT JAEA

To extend JAEA's capability of parametric severe accident uncertainty quantification, authors are applying the dynamic event tree method for scenario uncertainty quantification. The main idea of the scenario generator (RAPID) development is shown in Figure 2. When uncertain parameters in the severe accident code need to be considered, the scenario generator is able to produce a number of inputs for different accident settings. Because some models cannot be treated within the severe accident code, for example, the function and malfunction of safety-related systems, human interaction, etc., The scenario generator is designed to complement the limitedness of PRA modeling. The functioning of these models may depend on the simulation of severe accident codes or may be stochastic, so the scenario generator should communicate with the simulation, and adjust the computation accordingly. When simulations are finished, the tool should be able to perform data processing. The RAPID is programed using Python.

JAEA has been developing the THALES2 code to analyze the severe accident progression and estimate source terms for Level 2 probabilistic risk assessment. In recent years, an independent computer code of iodine chemistry simulation, KICHE, has been coupled with THALES2, through an interface program developed for the exchange of input/output between two codes. THALES2/KICHE is an integrated and fast-running severe accident code, which simulates the progression of severe accidents in light water reactor nuclear power plants, including simplified modeling of thermal-hydraulic response, core melt progression, and in-vessel and ex-vessel transport behavior of radioactive materials with the consideration of iodine chemical reaction kinetics in aqueous phase, etc. [25]. To test the dynamic PRA analysis, we only run the THALES2 part for the Level 1 PRA research.

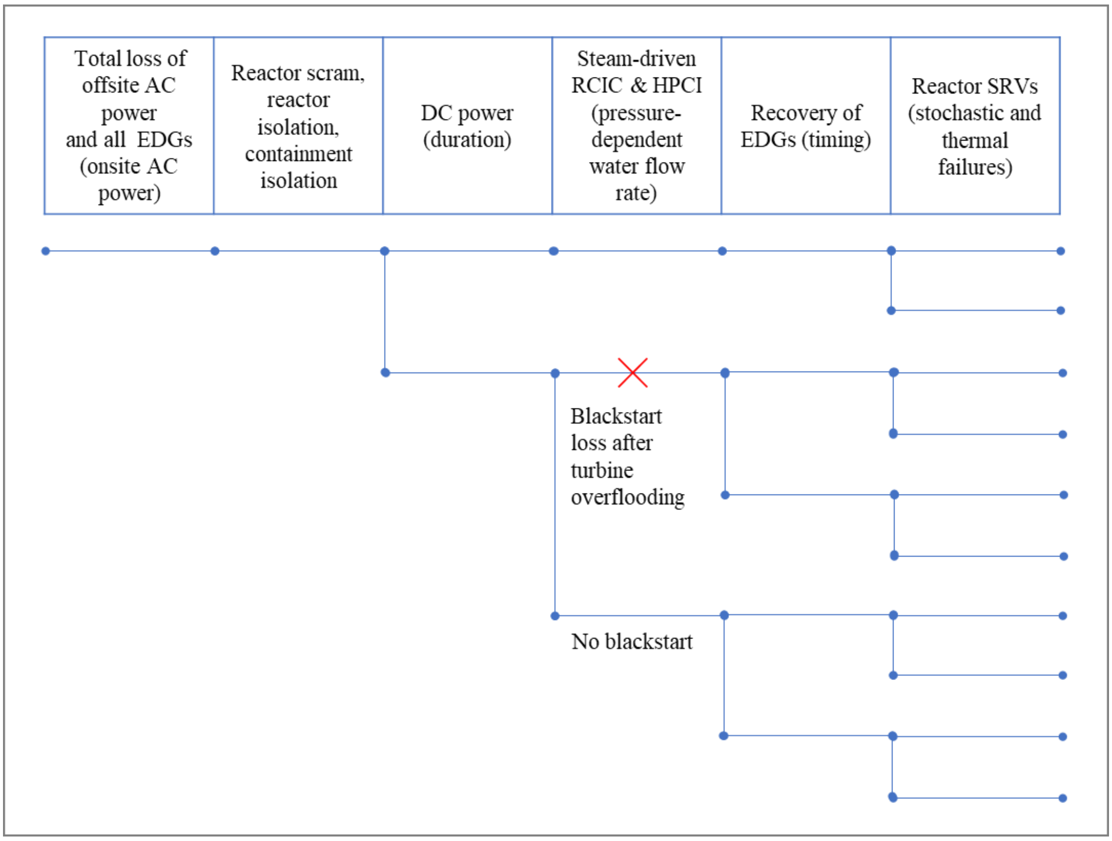
Figure 2: The combination of the scenario generator and the severe accident code, THALES2



4 SCENARIO UNCERTAINTY ANALYSIS INCLUDING SYSTEM INTERACTIONS

To practice the dynamic PRA and verify the understanding of related methods, we apply the scenario generator to a simplified station blackout accident [12]. The simplified event tree is shown in Figure 3, in which the number of top events is minimized, and failure modes of subsystems are assumed. The initiating event is a total loss of offsite alternating current (AC) power and all emergency diesel generators (EDGs). The depletion of direct current (DC) power and the recovery of the EDGs will affect the functioning of Reactor Core Isolation Cooling (RCIC) system, and then the core cooling status will be determined. We treat the duration of DC power as a stochastic distribution, and then assume a restoration probability of the EDGs. The operation of RCIC will be affected by the simulation results of primary coolant system. The flow rate is modeled as a function of the pressure of the primary coolant system. The modeling cannot be directly treated within the THALES2 code, so we let the scenario generator handle the dependency instead. The safety relief valves (SRVs) status will affect the accident consequence, so we assume two failure models for the SRVs, including stochastic failure and thermal seizure failure.

Figure 3: A simplified event tree model for the SBO accident



4.1. Stochastic distributions of DC depletion and EDGs recovery

The depletion of DC power is assumed to happen at the time defined by a truncated normal distribution, as shown in Table 2. The recovery of EDGs is also assumed with a Bernoulli distribution and the timing of the restoration is assumed with another truncated normal distribution. All parameters of these probability distributions, which do not reflect the actual plant operation experiences, are provided in Table 2.

Table 2: Probability distributions assumed for the depletion of DC power and recovery of diesel generator

Parameter	Reference value	Distribution	Range
DC depletion time	14400 seconds	Truncated Normal (μ = 14400, σ = 7200)	[7200, 21600]
EDGs recovery status	Yes or No	Bernoulli (p = 0.5)	N/A
EDGs recovery time before calculation ends	28800 seconds	Truncated Normal (μ = 28800, σ = 7200)	[0, 43200]

4.2. Thermal failure and stochastic failure of safety relief valve

A continuous depressurization of the reactor coolant system happens when an SRV fails to reclose. SRVs are activated at a predetermined opening pressure and deactivated when the pressure drops below another predetermined closing pressure. However, thermal expansion of the SRV would occur primarily during periods of gas flow (open cycles), which is known as the thermal seizure in the open position [26]. The valve is assumed to seize in the open position (fail to reclose) on the first cycle when the steam temperature is higher the criterion. The behavior leads to the uncertainty in the valve position, which is surely primary steam temperature and time dependent. Since there is certain model for the thermal seizure failure of SRVs, two hypothetical distributions are assumed to continue the DET modeling process, as shown in Table 3.

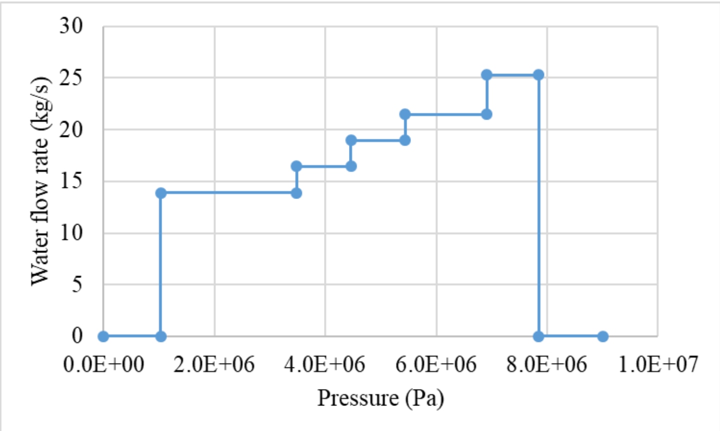
Table 3: Parameters for the thermal seizure failure and open area of SRVs

Parameter	Reference value	Distribution	Range
Temperature of thermal seizure failure	900 K	Truncated normal (μ = 900, σ = 100)	[800, 1100]
SRV open area	50%	Truncated normal (μ = 0.5, σ = 0.2)	[0.05, 1]

4.3. The pressure-dependent flow rate of RCIC

Since there is no explicit model for the dependency between RCIC flow rate and the pressure of the reactor coolant system, a hypothetical correlation is assumed for the modeling. The interaction needs to be modeled for the reduction of the uncertainty during the severe accident simulation. The assumed model shows a step-wise relationship between the RCIC flow rate and the pressure, and a sample of pressure-dependent flow rate is shown in Figure 4. Stochastic factors are assumed to reveal the operational relationship between safety-related systems and the reactor primary system.

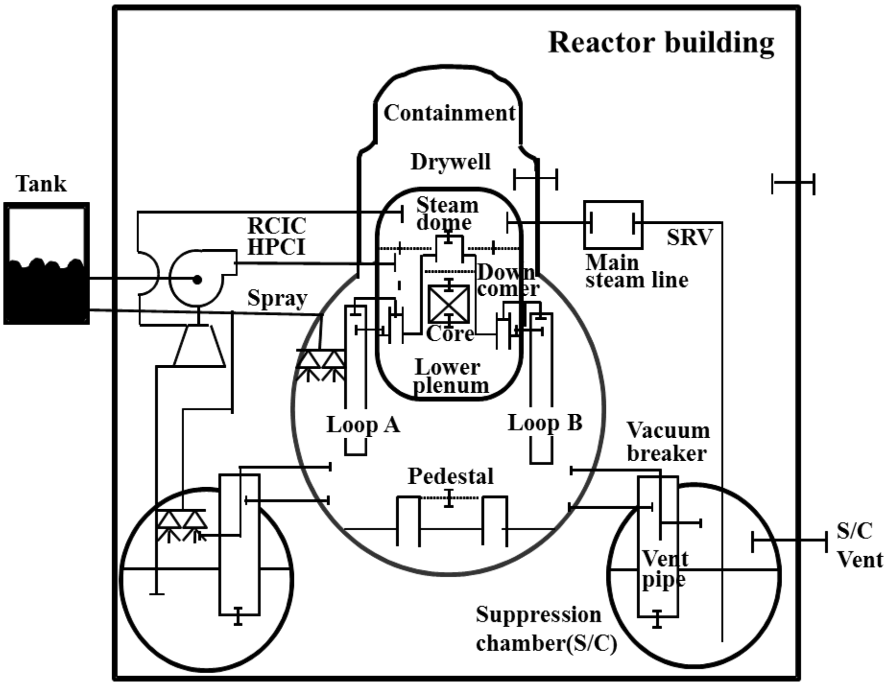
Figure 4: The pressure dependent RCIC water injection rate



4.4. Iterative thermal-hydraulic simulation for different accident sequences

Figure 5 shows the modeling of a BWR-4 plant with a MARK I containment via THALES2. The reactor cooling system is divided into seven volumes, consisting of reactor core, upper plenum, steam dome, downcomer, lower plenum and recirculation loops A and B. The containment vessel model comprises drywell (D/W), suppression chamber (S/C), pedestal and vent pipes that connect D/W and S/C. The environment volume is connected to the reactor building and S/C, which represent paths of the containment leak and S/C vent. The RCIC water injection is determined by the system simulation and an outside controller, which is conceptually represented by a stepwise model shown in Figure 4.

Figure 5: BWR modeling in THALES2



5 RESULTS AND SUMMARY

Using the method and tools, various accident sequences can be discovered. A representative accident sequence in Figure 6 illustrates the occurrence of key events and the change of maximum core temperature, which depends on the simulation conditions. It shows that, after the start of the SBO progression, the DG failure occurs at 11280 seconds, gap release of fission products at 19708 seconds, the core melts at 21306 seconds, the grid fails at 25877.4 seconds and the reactor pressurized vessel fails at 30843.7 seconds. Since the maximum of core temperature, including fuel pellets, cladding, control blade, and other structures, has been chosen to show the severity of accident progression, the temperature history does not decrease until the end of calculation. Given a criterion of core damage based on the fuel cladding temperature or fuel temperature, the status of core damage can be determined. In the paper, since the results of maximal core temperature are calculated via THALES2, a reference criterion is provided in Figure 6.

Five hundred of different accident sequences have been simulated and the history of maximum core temperature is obtained in Figure 7. Compared with the criterion for core damage status, the core damage frequency can be calculated straightforwardly. The history of maximum core temperature also gives us the information for scenario grouping, which can simplify the PRA analysis by using fewer representative accident sequences. The probabilistic and interactive simulation has the potential to quantify the scenario uncertainty for severe accident analysis.

The paper presents a preliminary dynamic PRA by using a scenario generator (RAPID) and an integral severe accident code (THALES2). Both tools are under development at JAEA. A more explicit PRA analysis using the proposed methodology and tools will be continued, and functions of the scenario generator will be enhanced. Especially, the numerical modeling of interaction during runtime will be well treated in the near future. In order to reveal which branching parameters more significantly influence the consequence of accident evolution, a sampling-based sensitivity analysis of the dynamic event tree model (coupled with THALES2) is expected to be included. More investigation of the dependencies between plant systems will be performed to reach more reasonable PRA results.

Figure 6: A typical SBO sequence sample simulated by THALES2

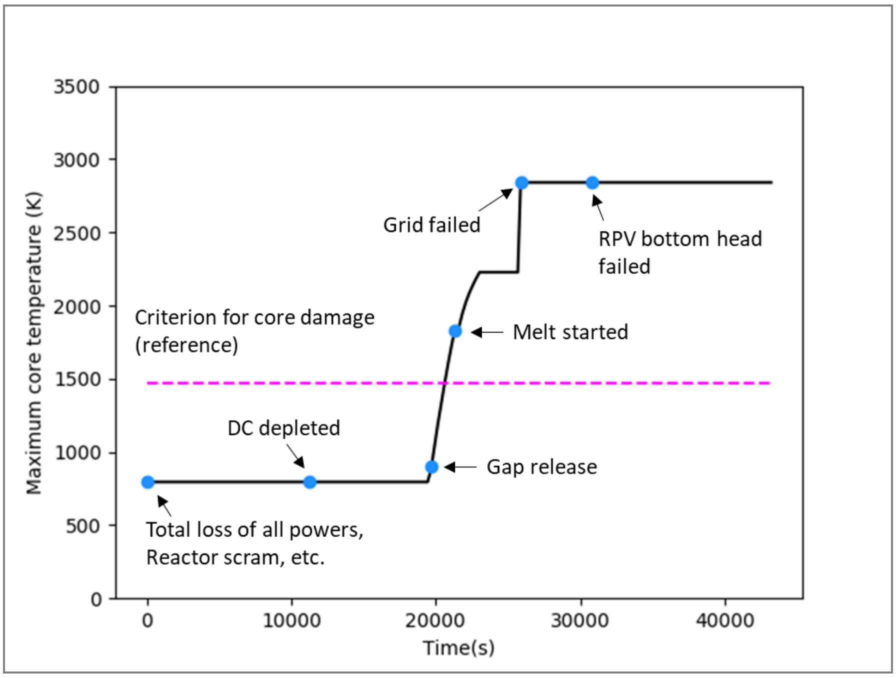
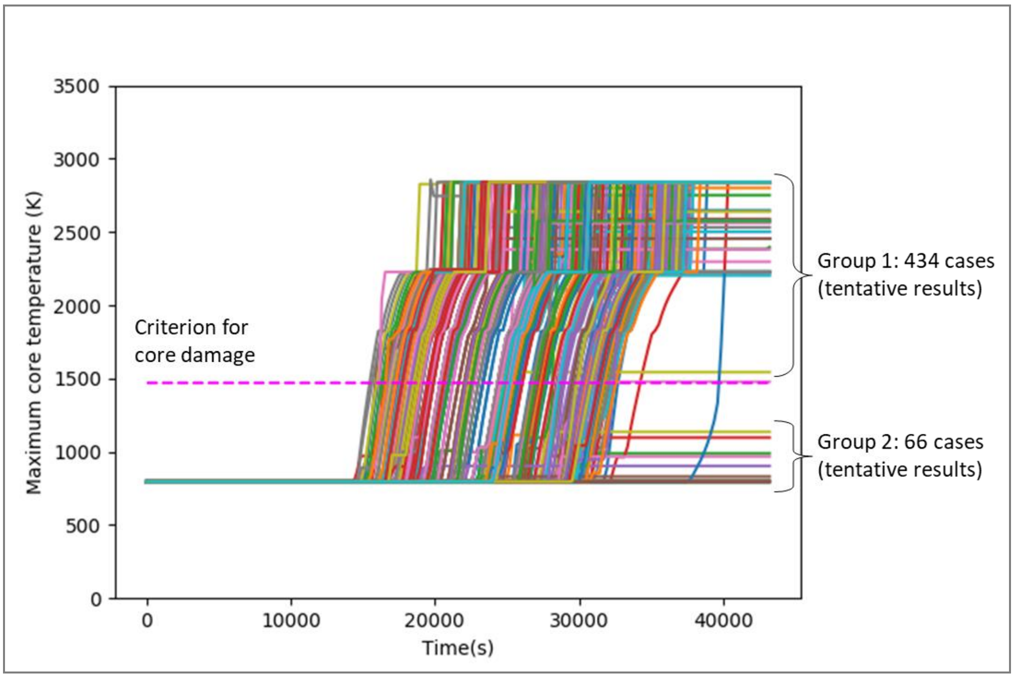


Figure 7: The history of maximum core temperature for different accident sequences (500 cases in total)



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