

Safety Assessment of LWR Systems

Background and Objective

To increase the safety of a nuclear power plant, it is necessary to take measures such as remodeling and adding equipment which can improve safety by revealing vulnerabilities through performing simulations and probabilistic risk assessments (PRA) which can grasp detailed phenomena. In regards

to this issue, we will enhance the technique for evaluating the effectiveness of safety improvement measures, and contribute to continual improvement of the safety of nuclear power plants by using this technique.

Main results

1 Development of a Seismic PRA model^{*1}

NRRC has been improving the methods for seismic evaluation and fragility evaluation. NRRC has also been developing a seismic accident sequence model for the target plant (Fig. 1). The objective of these research activities is to conduct seismic PRA. Furthermore, NRRC has been quantifying

CV Failure Frequency and developing a model for source term evaluation. The trial seismic PRA for the target plant (an actual NPP) demonstrated the feasibility of evaluating relative vulnerabilities according to seismic acceleration and effectiveness of SA countermeasures.

2 Development of Probabilistic Risk Analysis Library

In a level 1 PRA that assesses core damage frequencies, fault trees/event trees (FT/ET) are typically utilized. To validate numerical accuracy in the assessment, we developed a new library which can analytically evaluate success paths in ETs without approximation errors using Binary Decision Diagrams. Since this library can evaluate analytic

sequence occurrence frequencies, approximation errors in the minimal cut set method which is commonly used in a PRA can be quantified (Fig. 2). Hence, this library can accurately perform a seismic PRA, which previously had potential large numerical errors due to the minimal cut set method (O14001).

3 Development of a PRD method applied to level-2 PRA^{*1}

A method of Phenomenological Relationship Diagram (PRD) is applied to the quantification of probabilities of core damage events identified in the containment event tree (CET) used in the level-2 PRA, which evaluates frequencies of the release rate of fission products during the containment failure. The PRD evaluates a probability of the core damage phenomenon selected as a top-event of the CET, and is expanded to the lower component event by employing cause relations described by physical

equations. The probability of the top-event can be obtained by piling up the probabilities of the lowest elementary event combined by physical equations. In this study, the PRD is applied to the re-criticality events considered during the core damage. It is found that the PRD can quantitatively evaluate probabilities to non-linear phenomena such as effective multiplication factors and reactivity insertion rates which determine the released energy during the re-criticality event.

4 Evaluation of transient critical heat flux simulated reactivity initiated accident (RIA)

RIA is a phenomenon in which reactor power rapidly increases due to control-rod fall etc. Since the cooling limit of the fuel rod during RIA is defined by the critical heat flux on the fuel rod surface in a transient state, it is important to evaluate a transient critical heat flux. To evaluate this, experimental apparatus was designed and manufactured to evaluate the transient critical heat flux under low pressure conditions with a rod bundle heated by

direct current heating (simulated fuel rod bundle with fuel rods arranged in lattice pattern) (Fig. 4a). The transient critical heat flux was evaluated in the range of initial water temperature and flow rate assumed during RIA under low pressure condition, and an experimental database of the transient critical heat flux in consideration of cross-flow and outer wall effects was constructed.

*1 The research activity is a part of the publicly offered research on PRA sponsored by METI.

*2 Phenomenological Relationship Diagram: The method of calculating the probability of the ET branch that describes the physical chemistry phenomenon.

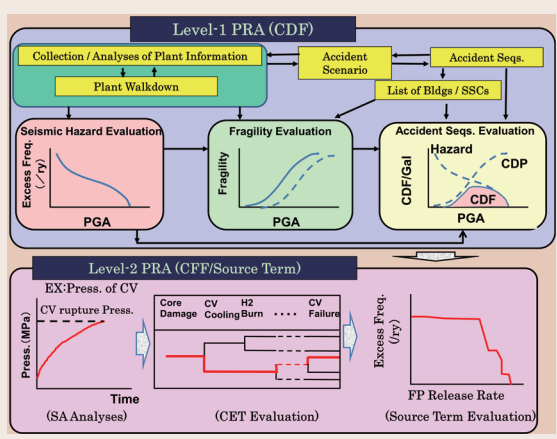


Fig. 1: Outline of seismic PRA method
 Seismic PRA evaluates seismic hazard based on seismic characteristics, fragilities of buildings & SSCs based on seismic capacities & responses, accident sequence evaluation with the results of seismic hazard / fragility evaluation, CFF & source term evaluation based on frequencies of PDSs quantified in accident sequence evaluation using CET and severe accident analyses. With the results of these evaluations, it is possible to identify relative vulnerabilities and to evaluate effectiveness of countermeasures. Since these results are inputs to risk informed decision makings, it is important the results are provided by evaluation models faithful to plant conditions.

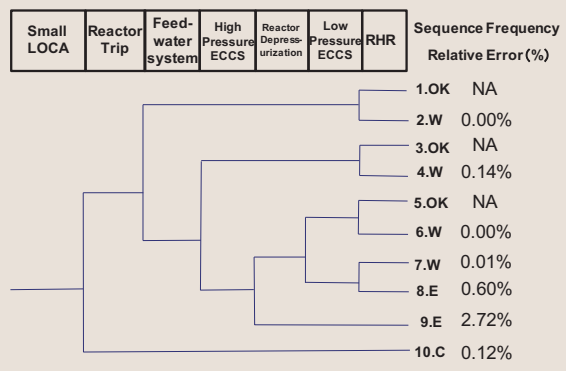


Fig. 2: Numerical Errors in small LOCA model with the minimal cut set method
 Small LOCA is a loss-of-coolant accident resulting from rupture of a small pipe. By analyzing an FT/ET model for small LOCA using the minimal cut set method and the binary decision diagram, we quantified numerical errors in the minimal cut set method. We verified that numerical errors in this model, which stems from an approximation in the minimal cut set method, are small and this small LOCA model is evaluated accurately.

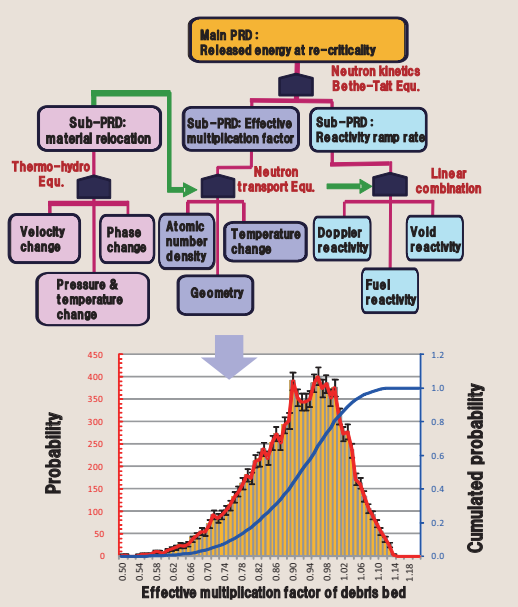


Fig. 3: PRD to evaluate the probability of re-criticality events during severe accidents
 The PRD evaluating the probability of released energy during a re-criticality event consists of three sub-PRDs describing material relocation, effective multiplication factor, and reactivity insertion rate for debris beds (upper part of the figure). The probability distribution function (PDF), shown in the lower part of the figure, of the effective multiplication factor of debris beds is determined by PDFs of the particle diameter (geometry), porosity (material relocation), and uranium/metal ratio (atomic number density) of the debris bed, based on the neutron transport equations as the cause relation between elementary events.

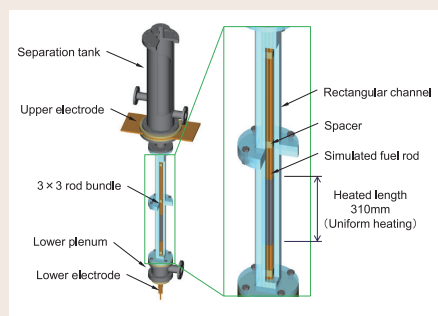


Fig. 4a: Experimental apparatus of RIA under low pressure condition

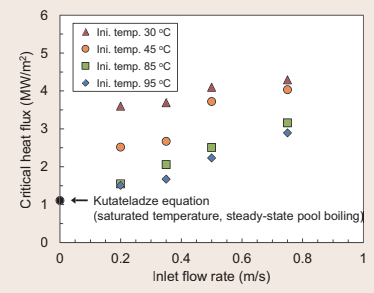


Fig. 4b: Transient critical heat flux during RIA under low pressure condition

The experimental apparatus was designed and manufactured to evaluate a transient critical heat flux (Fig. 4a). The 3x3 rod bundle simulating BWR fuel rod bundle was uniformly heated by direct current and the transient critical heat flux was evaluated by the temperature elevation of simulated fuel rod. The experimental data was acquired in the range of initial water temperature and flow rate assumed during RIA under low pressure condition. The results show that the transient critical heat flux increased with the decrease of initial water temperature, or the escalation of flow rate.