Priority Subjects with Limited Terms — **Establishment of Optimal Risk Management**

Safety Assessment of LWR Systems

Background and Objective

In order to acquire social receptivity for nuclear power generation and continue operation, it is necessary to make ongoing improvements and introduce the latest knowledge for enhancement of safety based on the experience of the Fukushima Daiichi nuclear power plant accident. It is necessary to perform simulations using models which can assess detailed phenomenon and make quantitative risk evaluations of important hazards to clarify the potentially vulnerable portions for evaluation of additional equipment in regards to safety improvement, especially from an engineering viewpoint. In this subject, we aim to contribute to the safety and stable operation of LWRs, through quantitative evaluation and the sophistication of evaluation systems.

Main results

1 Evaluation on characteristics of severe accident analysis codes

We have completed input decks preparations of typical PWR and BWR plants for the latest version of severe accident analysis codes, MAAP*1 ver5 and MELCOR*2. Main accident sequences were analyzed by MAAP ver5 (Fig. 1) and calculation results were compared with previously obtained results by MELCOR code. It was confirmed that the input deck for MAAP code was appropriate for severe accident evaluations. Based on quantitative comparison of calculation results between MAAP and MELCOR*3, we also found that there are differences in the progression of molten core – concrete interaction (Fig. 2) and the failure mode of a containment vessel*4.

2 Evaluation of a spent fuel pool in a severe accident

In light of the Fukushima Daiichi nuclear accident, creating measures against severe accidents for a spent fuel pool is also a task which requires attention. As such, spent fuel pool accident analyses using MAAP 5 were performed to quantitatively understand event sequence of both loss-of cooling due to total station blackout and loss-of-pool water accidents. These analyses revealed that there is enough time to implement measures until fuels are uncovered in the case of a loss-of-cooling accident (Fig. 3). It is also revealed that fuel temperature increases, however, it can be cooled by natural circulation of air when the decay heat power of fuel assemblies is small in the case of a loss-of-coolant accident.

3 Evaluation on hydrogen/vapor ventilation in a BWR reactor building

We analyze the behavior of hydrogen, which is generated from a RPV (reactor pressure vessel) and moves through a PCV (primary containment vessel), in a BWR (boiling water reactor) building using CFD (computational fluid dynamics) code. A uniform distribution of hydrogen and temperature is obtained. With the assumption of uniformity, we developed a lumped system to evaluate hydrogen density of the building. The system can be adopted to evaluate not only hydrogen but also vapor generated from SFP (spent fuel pool).

4 Development of the level 1 PRA models for NPPs in consideration of common cause failures caused by external events

Level 1 PRA (probabilistic risk assessments for the estimation of core damage frequencies) models for the typical BWR and PWR were developed in consideration of the effects of internal events and seismic hazards to analyze the vulnerability of the nuclear power plants to the natural hazards such as seismic and/or tsunami events.

*1 MAAP: The severe accident analysis code being developed by EPRI (U.S.). It is mainly used by electric utilities.
*2 MELCOR: The severe accident analysis code being developed by U.S.NRC. It is mainly used by the regulation side in Japan.
*3 "Development of Level-2 seismic PSA methodology", JNES/SAE06-087, Japan Nuclear Energy Safety Organization (JNES), (2006)
*4 R.Hiwatari, et al., 8th Japan-Korea Symposium on Nuclear Thermal Hydraulics and Safety (NTHAS8), 2012-2

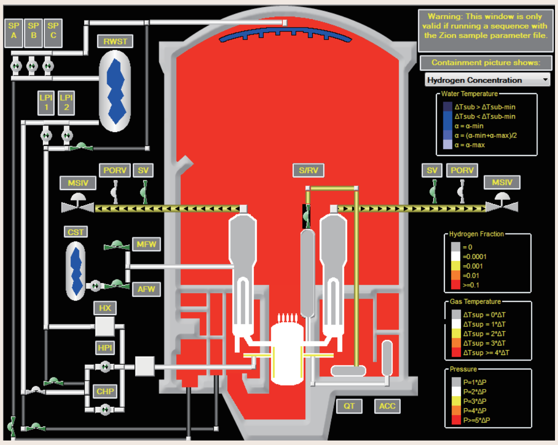

Fig. 1: Analytical model and graphics of calculation results for a typical PWR plant in the MAAP code

This figure shows an analytical model of a core, primary/secondary coolant systems and a containment vessel for a 4-loop-type PWR with dry containment in the MAAP code. Analytical results of temperature and pressure in the containment vessel and the coolant system loops are indicated by color mapping.

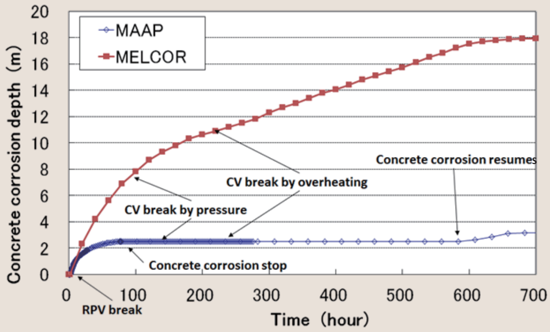

Fig. 2: Comparison between codes of the concrete corrosion depth in a containment vessel (CV) about the accident sequence "large rupture LOCA + feed water failure" in a PWR representation plant

The difference in the concrete corrosion depth in a CV following the breakage of a reactor pressure vessel suggests a difference in the physical model relating to molten fuel-concrete reaction, debris movement within the CV, etc.

* The result of MELCOR is created from reading values of the report "Development of Level-2 seismic PSA methodology", JNES/SAE06-087, JNES, (2006)*3

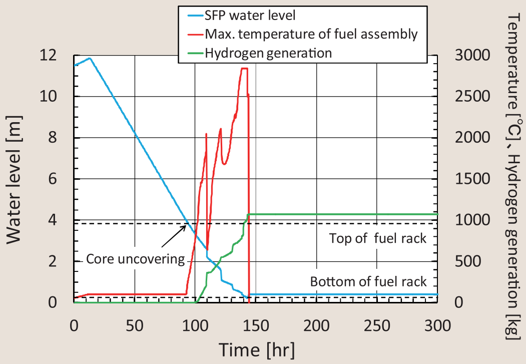

Fig. 3: Evaluation of spent fuel pool in a loss-of-coolant accident due to station power outage

A spent fuel pool accident analyses using the integral severe accident code "MAAP" were performed when the spent fuel pool loses its cooling function in the event of a station power outage. From these analyses, it is confirmed that there are about 89 hours until the fuels are uncovered under the severe thermal condition that the half of the spent fuel pool inventory is consisted of the fuel assemblies with cooling times of only 7 days after discharge from the core (Maximum burn up is 45,000 MWd/t, decay heat power is 8.6 MW).

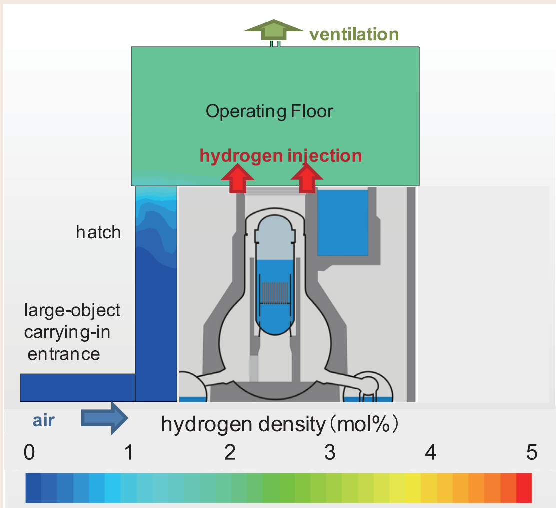

Fig. 4: Calculation on hydrogen behavior in BWR building

Three dimensional CFD analyses (static and transient) is conducted to evaluate hydrogen behavior in a BWR building. The hydrogen leaks from a PCV to the operating floor, mixes with the air from outside and is vented from the top outlet.