博士論文（要約）

論文題目

STUDY ON CORE DEGRADATION AND MELT PROGRESSION IN BWR SEVERE ACCIDENT CONSIDERING THE DEPRESSURIZATION CAUSED BY SEVERE LOADINGS

（極限荷重による減圧を考慮した BWR のシビアアクシデントにおける炉心損傷と溶融に関する研究）

氏名　劉 茂龍

SAMPSON code is state-of-the-art, system-level, severe accident analysis code developed by the IMPACT Project in Japan to investigate severe accident phenomena in light water reactors. SAMPSON code integrates various analysis modules into a single code. The present study focuses on gaining greater insight through SAMPSON code analyses of the severe accident that occurred at Fukushima Dai-ichi Unit 1.

As there are structural differences between pressurized water reactors (PWRs) and boiling water reactors (BWRs), some additional models are necessary for BWR simulation compared to PWR simulation. In addition, BWRs and their primary containments have unique features that must be modeled for best estimate severe accidents analysis. The reactor core of a BWR consists of an array of fuel assemblies with cross-shaped control blades located in the bypass channels between the assemblies. Each fuel assembly consists of a fuel rod bundle surrounded by a Zircaloy channel box. Under severe accident conditions, oxidation reaction of these metals with high-temperature steam material interaction among the B_4C/stainless steel, and B_4C/Zircaloy have a significant effect on the melting and subsequent relocation of the control blades and channel box. In addition, there are a large number of independent parameters governing heat transfer including radiation heat transfer in the reactor core uncovered due to low water level. In the present study, radiation heat transfer model between the core structures is improved to predict the reactor behavior more accurately in such severe conditions.

During the progress of a severe accident, due to the early depletion of injection water, the reactor core is rapidly heated up by decay heat and oxidation reaction energy. BWRs have a large number of instrumentation tubes inserted in the core that act as the pressure boundary for the reactor. Once the core is heated up, there is a high possibility that theses stainless steel tubes will fail. Failure opens a flow path directly from the reactor core to the containment. In addition, as steam in the core becomes hot, it heats the downstream structures of the reactor coolant systems, which challenges integrity of the reactor pressure boundary. The temperature of other structures, such as safety relief valves (SRVs) and the main steam line, could exceed design limits and cause failure leading to leakage of fission products and steam. Failure time and leakage path flow area before manual depressurization have huge effects on accident progression and release of fission products.

TEPCO announced that the SRVs of the Fukushima Dai-ichi NPP Unit 1 never were manually opened during the accident. However, reactor pressure measured about 1.0 MPa at 2:43 on March 12. Such unanticipated depressurization might accelerate uncovering the core and might prevent early containment failure caused by direct containment heating. Using SAMPSON code, I evaluated the reactor depressurization caused by failure of the source range monitor (SRM) guide tube, the SRV gasket and the main steam line based on the Fukushima accident. Based on the sensitivity analysis, I propose two candidate mitigation strategies to depressurize the reactor for the management of in-vessel events during the early phase and late phase (after core degradation has occurred) of postulated BWR severe accidents.

Concrete findings of the present study are as follows.

1. Improvement of severe accident response models for BWR

The purpose of the SAMPSON code is to study the progression of potential BWR severe accidents that extend beyond the design basis. In the present study, I have improved the code with regard to BWR severe accident phenomenology and translation. With proper consideration given to the special features of BWR design, it is now possible to calculate a reasonable estimate of the response of a BWR facility under severe accident conditions. Nevertheless, many areas of uncertainty remain, and there is a clear need for additional experimental verification of existing models.

The modified physical models of SAMPSON code in the present study are as follows.

(1) Core structures oxidation model

(2) Eutectic reaction of B_4C and stainless steel

(3) Radiation heat transfer model

A radiation heat transfer model in the radial and axial direction was developed to determine relative importance of radiation heat transfer in the cooling of degraded BWR cores. (a) Radiation heat transfer in the radial direction plays an important role in temperature distribution in the radial direction of the core structures, which has significant impact on core degradation and integrity of the reactor pressure vessel (RPV) during a severe accident. (b) Axial radiation has significant impact on cooling of the fuel rod and control rod because it significantly enhances heat transfer in the axial direction of the core.

(4) Fuel cladding creep rupture model

(5) Stainless steel tubes creep rupture model

2. Understanding of Fukushima Dai-ichi Unit 1 accident progression

The present study embodies several predictive component failure treatments for the SRM guide tube, SRV gasket under high temperature, and potential rupture of the main steam line. Fukushima Dai-ichi Unit 1 analysis consistently predicts SRV gasket failure and main steam line creep rupture as the mode of vessel depressurization. SAMPSON-predicted results compare favorably against the limited data available for RPV, drywell, and wetwell pressures. A detailed analysis of the Fukushima accident is helping to confirm the current understanding of severe accident plant response and to guide future areas of research to enhance plant safety.

There are known areas of important phenomenological uncertainty in the analyses: (1) melt progression and core relocation details, (2) input conditions of observed accident sequences, and (3) physical models. These uncertainties can produce differences in timing of major events, such as water boildown to top of core, onset of rapid Zircaloy oxidation reaction, SRM guide tube and SRV gasket failure time, and lower head failure.

3. Candidate mitigative strategies for depressurization

Based on improved SAMPSON code, sensitive analysis was carried out to analyze the influences of failure criteria and leakage area on the reactor pressure boundary. Based on the sensitivity analysis, I have proposed two passive depressurization systems in the present study. Those systems are (1) Passive SRM guide tubes, and (2) Safety heat-up rupture valve.

According to the present simulation on passive SRM guide tubes, in Fukushima-like, long-term, station blackout (SBO) accident management, it is not the best option to open SRVs during core heat-up if additional heat sink is not available to cool the reactor. However, leakage caused by failure of the reactor pressure boundary, such as passive SRM guide tubes, can effectively depressurize the reactor but also can delay accident progression during core heat-up by causing an optimized leakage area. To prevent early failure of the containment caused by leakage through the passive SRM guide tube, the latter should be connected to the suppression pool so that it can be used as a passive depression system under severe accident conditions.

In addition, the safety heat-up rupture valve failure temperature has significant effect on the core degradation process. Failure of the safety heat-up rupture valve can depressurize the reactor but also provides sufficient steam to melt part of the core in a short period. Meanwhile, the steam flow directly into the drywell causes simultaneous over-pressure and overheating of the containment. The safety heat-up rupture valve should be connected to the suppression pool, so it also can be used as a candidate of passive systems to depressurize the reactor pressure to prevent direct containment heating.

I also compared the depressurization strategy of current severe accident management guideline (using SRVs) with the passive depressurization strategies to evaluate the capability of the passive depressurization strategies in the management of an accident. For all strategies, the reactor pressure at the moment of RPV failure is lower enough to avoid direct containment heating. The rapid depressurization by opening all SRVs could reduce the reactor pressure very quickly. However, high rate of gas discharge causing excessive loss of water inventory can accelerate core degradation that leads to early failure of RPV. In comparison, the calculations with passive depressurization systems delay the time to RPV failure and they may provide more time for the operators to recover either AC power or auxiliary feedwater, or to find an emergency source of feedwater to maintain the integrity of the RPV. Therefore, passive depressurization strategies might be regarded as a better strategy compared to the manual depressurization using SRVs.

Leakage area and time of initiation of leakage are two key parameters for passive depressurization systems. With the optimization of these parameters, proposed system is capable to (1) depressurize the reactor and prevent direct containment heating, (2) reduce the total mass of hydrogen generation comparing to the depressurization strategy using SRVs, and (3) delay the RPV failure time. There are uncertainties on the effective leakage area for both passive in-core guide tubes and passive SRV. Further investigation on the failure condition and leakage area is necessary for the application of these passive depressurization systems.