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

その他のタイトル	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
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論 文 の 内 容 の 要 旨

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)

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

As there are structural differences between PWRs and BWRs, some additional models are necessary in the BWR models compared to the PWR models. In addition, BWRs and their primary containments have unique features that must be modeled for best estimate severe accident analysis. The reactor core of a BWR consists of an array of fuel assemblies with cross-shaped control blades located in the bypass channel between these 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 as well as material interaction between the B_4C, stainless steel, and Zircaloy would have significant impacts on the melting and subsequent relocation of the control blade and channel box structures. In addition, there are a large number of independent parameters governing heat transfer in the reactor core under simulated uncovered core conditions, including the radiation heat transfer. The radiation heat transfer in these complicate structures is improved.

During the progress of the severe accident, due to the early depletion of injection water source, the core becomes rapidly heated up by the decay heat and oxidation reaction energy. There are a large number of instrumentation tubes inserted in the reactor core of BWRs, which are also performed as the pressure boundary of the reactor cooling system. There was high possibility that theses stainless tubes would fail once the core is heated-up, which build up a flow path directly from the reactor core to the containment. In addition, as the steam in the center of the core becomes hot, the downstream structures of the reactor coolant systems would be also heat-up by the high-temperature steam, which can also challenge reactor pressure boundary integrity. The temperature of some other structures, such as SRVs, the main steam line would also exceed the design limit and cause the failure of the pressure boundary. Failure time and leakage path of the BWR reactor pressure boundary before manual depressurization have huge impact on the accident progression and fission products release.

TEPCO announced that the safety relief valves of the Fukushima Daiichi NPP Unit 1 were never manually opened during the accident due to the shortage of DC power. However, the measured reactor pressure was about 1MPa at 2:43 of March 12. Such unanticipated depressurization might accelerate the core “uncovery” and also prevent early containment failure caused by the direct containment heating. We have evaluated the reactor depressurization caused by failure of the SRM guide tube and SRV gasket based on the Fukushima accident with SAMPSON code. As a result of this effort, two candidate mitigative strategies for the management of in-vessel events during the early-phase and late-phase (after-core degradation has occurred) of postulated BWR severe accidents were proposed to depressurize the reactor based on the sensitivity analysis.

The concrete findings of the present study are as below.

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, we have improved the code with regard to BWR severe accident phenomenology and translation. With proper consideration given to the special features of the 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 below.

(1) Core structures oxidation model

(2) Eutectic reaction of B4C and stainless steel

(3) Radiation heat transfer model

A radiation heat transfer model in the radial and axial direction has been developed to determine the relative importance of radiation heat transfer in the cooling of degraded BWR cores. (1) Radiation heat transfer in the radial direction plays an important role in the temperature distribution in the radius direction of the core, as well as core shroud temperature, which has significant impact on the core degradation and the integrity of the RPV during a severe accident. (2) Axial radiation has significant impact on the 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 1F1 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. The base-case Unit 1 analysis consistently predicts SRV gasket failure and the main steam line creep rupture as the mode of vessel depressurization. SAMPSON-predicted results compare favorably against the limited data available for the RPV, drywell, and wetwell pressures. A detailed analysis of the accidents are help to confirm the current understanding of severe accident plant response and guide future areas of research to enhance plant safety.

There are known areas of important phenomenological uncertainty in the analyses (melt progression and core relocation details) and observed areas of sequence sensitivity to uncertain input conditions and physical models. These uncertainties can produce differences in timing of the major events such as water boildown to top of core, onset of rapid zircaloy oxidation reaction, SRV gasket failure time, and lower head failure.

Candidate mitigative strategies for depressurization

Based on the improvement of SAMPSON code, SAMPSON code analyses was used to evaluate the accident management/severe accident management guidelines success likelihood or alternative mitigation strategies with 1F1 input model. We have evaluated the reactor depressurization caused by failure of the SRM guide tube and SRV gasket based on the Fukushima accident with SAMPSON code. Two candidate mitigative strategies for the management of in-vessel events during the early-phase and late-phase (after-core degradation has occurred) of postulated BWR severe accidents were proposed to depressurize the reactor based on the sensitivity analysis.

According to the present simulation on SRM guide tubes, it is not the best option for the Fukushima like long-term SBO accident management by opening SRV during core heat-up if additional heat sink is not available to cooling the reactor. However, instead of opening SRVs, the leakage caused by failure of the RCS pressure boundary (such as SRM guide tubes) can delay the accident progression during core heat-up with optimized leakage area. In order to prevent the early failure of the containment caused by leakage through SRM guide tube, the SRM guide tube should be connected to the suppression pool so that it can be used as a passive depression system under severe accident condition.

In addition, SRV gasket failure temperature has significant impact on the core degradation process. Early failure of the SRV gasket provides sufficient steam to melt part of the core in a short period. Meanwhile, the steam flow directly into the drywell and would cause over-pressure and over-temperature of the containment at the same time. Thus the new design of SRV gasket should be able to tolerate even higher temperature so it can also be used as a candidate of passive systems to depressurize the RCS pressure to prevent direct containment heating.