**Study on Severe Accident Progression and Source Terms in  
Fukushima Dai-ichi NPPs**

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**Abstract**. It has been past three years since the severe accident in TEPCO’s Fukushima Dai-ichi Nuclear Power Plants. After Fukushima Dai-ichi accident, several investigation reports were published. These reports pointed out lessons learned from the accident. Nuclear Regulation Authority (NRA) enacted new regulations last year, which require accident management and counter measures against severe accident, etc., to enhance safety of nuclear power plants and nuclear facilities. On the other hand, NRA launched new committee to investigate unsolved issues of the accident, which were pointed out previous investigation reports.

Decommissioning of severely damaged plants is in progress. In parallel, onsite R&Ds for investigation of the accident are undergoing. However, due to technical difficulties for investigation such as high radiation, leakage of contaminated water, etc., available information, especially for the inside of primary containment vessel, is limited. S/NRA/R is conducting both experimental programs and computational analyses to study important severe accident phenomena and accident progression of Fukushima Dai-ichi accident. This paper summarize current research activities.

# Introduction

On March 11, 2011, the Fukushima Dai-ichi Nuclear Power Plants (NPPs) Units 1, 2, and 3 were in operation and Units 4, 5, and 6 were shut down for refueling and maintenance. In Unit 4, the nuclear fuel was offloaded to its spent fuel pool. At 14:46, the Tohoku District-off the Pacific Ocean Earthquake occurred and resulted in the automatic shutdown of Units 1 to 3. As a result of the earthquake, offsite power was lost to the site and the emergency diesel generators (EDGs) started automatically at all six units. Approximately 40 minutes following the earthquake, the first tsunami wave inundated the site followed by multiple waves. The tsunami resulted in extensive damage to site facilities and a complete loss of AC power, known as station blackout (SBO), at Units 1 to 5. The air-cooled EDG in Unit 6 survived and successfully provided ac power for both Units 5 and 6. In the Units 1 to 3, however, cooling to the fuel was lost, resulting in damage to the nuclear fuel.

The Tokyo Electric Power Co., Inc. (TEPCO), the licensee of the Fukushima Dai-ichi NPS, evaluated the plant behavior during the accident in Units 1 to 3 using an accident analysis code, MAAP, based on the information on initial conditions, operation, etc. The results were reported to the Nuclear and Industrial Safety Agency (NISA), the regulatory authority at that time, on May 23, 2011 [[1](#Tok11)].

The NISA had directed the incorporated administrative agency, Japan Nuclear Energy Safety Organization (JNES), to perform the accident analysis independently to review the above report. JNES performed the analysis using the MELCOR developed by the U.S. Nuclear Regulatory Commission (NRC) and the results were reported to NISA. They were included in the report of the Japanese government to the International Atomic Energy Agency (IAEA) [[2](#Nuc11)].

Since then, the analyses have been continuously improved by taking into account new information on, such as operation of safety systems, leakage/failure of primary pressure boundary and containment, etc. However, it is still difficult to predict when and how much molten core fell into containment mainly due to large uncertainty in injection water flow rate into the core. We believe it is necessary to continue this analysis to understand the phenomena involved in the course of the accident and to take lessons learned fully from the accident. Also it is important to clarify what are needed to better understand the phenomena and what research is needed further. This paper describes current research activities of Regulatory Standard and Research Department of Nuclear Regulation Authority (S/NRA/R) on Fukushima Dai-ichi accident.

# Research activities of S/NRA/R

After Fukushima Dai-ichi accident, several investigation reports were published [[3](#BL32)] [[4](#BL31)] [[5](#BL33)]. These reports pointed out lessons learned from the accident. Nuclear Regulation Authority (NRA) enacted new regulations last year, which require accident management and counter measures against severe accident, etc., to enhance safety of nuclear power plants and nuclear facilities. On the other hand, NRA launched new committee to investigate unsolved issues of the accident, which were pointed out previous investigation reports. Since hydrogen explosion at Unit 4 is one of them, investigation results of inside of the reactor building of Unit 4 was reported to the committee.

S/NRA/R is conducting both experimental programs and computational analyses to study important severe accident phenomena and accident progression of Fukushima Dai-ichi accident. Our current subjects are described below.

## Thermal Stratification

The reactor core isolation cooling system (RCIC) had been operated for more than 60 hours and approximately 20 hours at Unit 2 and Unit 3, respectively. Water was injected into the core and steam was discharged through RCIC turbine to suppression pool (S/P). During RCIC operation, water in the S/P was locally heated up around the exit of RCIC exhaust line. It is assumed that thermal stratification occurred in the S/P, i.e. the temperature distribution of the S/P was not uniform. Although similar thermal stratification should occur in Unit 2, primary containment vessel (PCV) pressure of Unit 2 might be suppressed by flooding of sea water into torus room. Recent experimental study suggests formation of thermal stratification in the S/P after long term RCIC operation [[6](#BL35)].

To simulate this situation, nodalization of our calculation model is modified. The suppression chamber (S/C) is nodalized by vertical two volumes and the inflow of enthalpy from RCIC exhaust line is distributed to upper and lower volumes. The distribution ratio was determined so as to obtain better agreement with the observed dry well (D/W) pressure by sensitivity calculations.

## Leakage of primary pressure boundary and PCV

In the previous analysis [[2](#Nuc11)], temperature of gaseous phase in reactor pressure vessel (RPV) increased significantly after core degradation. In the calculation, superheated steam was discharged through safety and relief valves (S/RVs) to S/P. It is questionable whether the gasket of flange of S/RV is intact at high temperature. Therefore, in our study, a leakage from primary pressure boundary to D/W is assumed to be taken place if the temperature of gaseous phase at RPV exceeds 723 K. Alternatively, Sandia National Laboratory's calculation predicts creep rapture of main steam line [[7](#BL36)].

According to the survey in the reactor buildings (R/B) of Units 1 and 3, significant contaminations were observed [[8](#Tok111)]. It was reported that the radiation level was very high on the fifth floor of R/B at Unit 1 and near the mechanical hatch on the first floor of R/B at Unit 3. These contaminations strongly suggest fission products leaked from PCV to R/B. Potential leak locations of Mark-I type containment were discussed in Reference [[9](#Mas12)]. It was suggested that the sealing of penetrations and gasket of flanges were susceptible to over-temperature failure. On the other hand, the hydrogen explosion analyses of Units 1 and 3 done by JNES [[10](#Ogi12)], showed that the overall scales of the explosions could be reproduced if it was assumed that hydrogen leaked through the top flange of Unit 1 and mechanical hatch of Unit 3. Thus, over-temperature failure of PCV is assumed to occur if the PCV temperature exceeds 623 K. Leak paths were modeled from D/W to fifth floor of R/B at Unit 1 and from D/W to first floor of R/B at Unit 3. To identify leakage location of the PCV, we apply computational fluid dynamics (CFD) calculation for complement of lumped parameter model calculation.

As TEPCO recently published progress report of investigation, they found water leak locations from PCV [[11](#BL34)]. The updated information will be implemented to new calculation.

## External water injection

In this severe accident, external water injection was the most important accident management for prevention and mitigation of core damage. If whole external water, fed by fire engine, was injected to the core, core ought to have been cooled. However, in reality, all three cores were severely damaged and primary pressure boundaries were failed since RPV was not refilled. Hence, external water injection to the core was less than the record.

TEPCO recently reported identified potential bypass paths of external water injection lines [[11](#BL34)]. Actual amount injected to the core significantly affects core status, hydrogen generation, molten core concrete interaction (MCCI), etc. These energetic phenomena also influenced accident progression thereafter. Realistic estimation of external water injection is useful to reduce uncertainty of core status, MCCI ablation depth, etc.

## Source Terms

A number of inverse estimations of source terms were published. On the other hand, forward estimation, i.e. accident progression analysis, provides complementary information, especially during the period of no monitoring measurement is available. Environmental consequences were studied using OSCAAR code with source terms obtained by MELCOR [[12](#Hos12)]. Based on the accident progression analyses, radionuclides release behavior is summarized as below.

Firstly, noble gas is released to the environment due to core damage of Unit 1. Since leakage paths from PCV to R/B and from R/B to the environment, which are allowed in design, are considered, noble gas is released to the environment after core damage. Since PCV leakage owing to over temperature is presumed, fission products (FP) release rate increases about 17 hours. Based on the monitoring data, it is presumed that PCV vent is partially succeeded at Unit 1. Therefore, radionuclides are released to the environment by PCV vent at about 20 hours and 24 hours, respectively. On March 13, FP is released via PCV vent from Unit 3. A large amount of FP is released from Unit 2 on March 15, which mainly contributes high contamination of northwest part of Fukushima prefecture.

To complement between source terms calculated by accident progression analyses and dose rate measured at monitoring posts, we apply CFD calculation for estimation of air stream on Fukushima Daiichi site.

# Conclusion

Regarding severe accident progression including source terms, most of the phenomena that took place during the accident have become reasonably well understood by efforts made by various organizations.

The accident progression analyses have been updated and improved by taking into account leakage due to over-temperature of primary pressure boundary and containment, thermal stratification at suppression chamber, etc. Qualitatively speaking, calculated results are consistent with observed data. At the moment, it is still difficult to evaluate MCCI since it is difficult to evaluate the amount of molten fuel that fell into the containment. We will continue to update the analyses by taking into account the data that are expected to be obtained at the Fukushima site through R&D for decommissioning process.

New attempts, such as environmental consequence analysis using the source terms calculated by SA progression analysis, are expected to be able to obtain better understanding the accident progression and consequences.

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