OVERVIEW AND OUTCOMES OF BENCHMARK STUDY OF THE ACCIDENT AT THE FUKUSHIMA DAIICHI NPS (OECD/NEA BSAF PROJECT)

Fumihisa Nagase
Japan Atomic Energy Agency, Nuclear Science and Engineering Center
Shirakata-shirane 2-4, Tokai, Naka, Ibaraki, 319-1195, Japan
nagase.fumihisa@jaea.go.jp

Randall O. Gauntt
Sandia National Laboratories, Severe Accident Analysis Department
P.O. Box 5800, MS 0748, Albuquerque, New Mexico, 87185-0748 USA
rogaunt@sandia.gov

Masanori Naito
The Institute of Applied Energy, Nuclear Engineering Power Center
1-14-2 Nishi-Shimbashi, Minato-ku, Tokyo, 105-0003 Japan
mnaito@iae.or.jp

ABSTRACT

The OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) Project has been established in November 2012 with the support of the Agency of Natural Resource and Energy, the Ministry of Economy, Trade and Industry, Japan. Fifteen organizations of eight countries calculated thermo-hydraulic behavior with severe accident integral codes for the time span of about six days from the occurrence of the earthquake. The primary objective of this benchmark study is to estimate accident progression, status in the reactor pressure vessels and primary containment vessels, and status of debris distribution for a debris removal plan at the Fukushima Daiichi nuclear power plant. A common information database including plant specifications and timeline plant operation data was provided by Japan and boundary conditions were discussed among the participants.
Finally the calculated results submitted by the participants were compared and evaluated to estimate the accident progression and status inside the reactors though the results showed wide variations. Still remaining uncertainties and data needs that are useful to the communication between analysts and decommissioning activities are also summarized as the output from the project. The present paper describes overview and main results of the BSAF project together with some implications to the decommissioning activity at the Fukushima Daiichi NPS.

KEYWORDS
OECD/NEA, BSAF, SA codes, benchmarking, Fukushima Daiichi NPSs

1. INTRODUCTION

According to the "Roadmap towards Restoration from the Accident at Fukushima Daiichi Nuclear Power Station", the Agency of Natural Resource and Energy, the Ministry of Economy, Trade and Industry, Japan (METI/ANRE) has been promoting the R&D plan towards the decommissioning, which includes the analysis of the accident progression and their current status of the NPS. In a number of member countries of Organization for Economic Cooperation and Development, Nuclear Energy Agency (OECD/NEA), severe accident (SA) analysis codes have been developed after the accident at the Three Mile Island Unit 2 reactor and the codes are also used to analyze the accident at the Fukushima Daiichi NPS. Taking account of the above circumstances, the Committee on the Safety of Nuclear Installations (CSNI) of the OECD/NEA decided to support the Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF), and the project was launched in November 2012 with the participation of France, Germany, Japan, Republic of Korea, Russian Federation, Spain, Switzerland, and the United States.

The objectives of the benchmark are to analyze accident progression and current status, including fuel debris distribution in the reactor pressure vessels (RPVs) and primary containment vessels (PCVs) for preparation of fuel debris removal, and to improve SA analysis codes. The calculated results submitted by the participants were compared and evaluated to estimate the accident progression and status inside the RPVs and PCVs. The participants discussed about uncertainties still remaining in the understanding of the accident and data needs from the view point of analysts, in order to enhance the communication between analysts and decommissioning activities. The present paper describes overview and main results of the BSAF project together with some implications to the decommissioning activity at the Fukushima Daiichi NPS.

Further technical details of the results of the BSAF project are presented in a separated paper [1].

2. OVERVIEW OF THE PROJECT

2.1. Scope

The accident events at Fukushima Daiichi NPS include a very wide range of phenomena and plant behaviors which are beyond current analysis functions of a single SA integral code or a coupled code system. In addition, information about the accident such as accident progression, actions of the operators and operation of safety facilities, which can be the input data and verification data for the analyses, has not been fully available and would be obtained after many years. Thus, a *phased approach* is applied in this project as in the case of other NEA benchmark exercises. The range of analysis in the phase-1 is as follows:

- To conduct full scope analyses of Fukushima Daiichi NPS Units 1 to 3 using currently available SA integral codes.
- To use a time span for analysis of accident events of about six days from the occurrence of the earthquake (or reactor scram) until about noon on 17 March 2011; this end time is chosen because stable and continuous cooling was attained by alternative water injection by then and after that the change in the unit statuses was considered to be negligible.
- To analyze in full scope the following phenomena:
    (1) Initial transient from rated condition to core heat-up
    (2) Core heat-up
    (3) Core melt
    (4) Behavior of core internals (core shroud)
    (5) Core status including debris behavior
    (6) Molten debris-coolant interaction in the lower plenum (if necessary)
    (7) RPV failure
    (8) PCV thermal-hydraulics
    (9) MCCI (molten core concrete interaction)
    (10) Hydrogen generation (excluding the hydrogen explosions)

2.2. Input data and boundary condition

In order to conduct full scope analysis of accident progression, Tokyo Electric Power Co. Inc. (TEPCO), Toshiba Co., Ltd., Hitachi-GE Nuclear Energy, Ltd., Global Nuclear Fuel-Japan Co., Ltd., and Nuclear Fuel Industries, Ltd. jointly prepared a common information database consisting of plant specifications, and timeline of plant operation data and measured data during a defined period from the start of the accident.
A web portal (https://fdada.info/) was also established to share the information among the participants as a part of activities of this project. It is open to the public, though it has a protected members' area, and provides access to the technical information on accident analysis and decommissioning activities at the Fukushima Daiichi NPS.
In the accident at the Fukushima Daiich NPS, the information on in-reactor conditions, operation of equipment, on-off state of valves, and effects of emergency measures is quite limited or cannot be quantitatively specified. However, initial conditions and boundary conditions have to be fixed for execution of the analysis. A complete list and possible treatment of the boundary conditions were organized by the operating agent and discussed among the participants.

2.3. Participants

The participants of the BSAF project are the Japan Atomic Energy Agency (JAEA), the Nuclear Regulatory Authority (NRA), the Central Research Institute of Electric Power Industry (CRIEPI), the Institute of Applied Energy (IAE), Japan; the Institut de Radioprotection et de Sûreté Nucléaire (IRSN), the Commissariat à l'Energie Atomique et aux Energies Alternatives (CEA), France; the Gesellschaft für Anlagen-und Reaktorsicherheit (GRS), Germany; the Korea Atomic Energy Research Institute (KAERI), Korea; the Russian Academy of Sciences Nuclear Safety Institute (IBRAE), The State Atomic Energy Corporation ROSATOM, Russia; the Consejo de Seguridad Nuclear (CSN), Spain; the Paul Scherrer Institute, Switzerland; and the United States Department of Energy (DOE) and the United States Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute (EPRI), the United States of America. Table I shows the list of the computer codes used by the participants.

Table I. List of computer codes used by participants

Country	Institutes	Codes
France	CEA	Analytical study
	IRSN	ASTEC V2.0 rev3 p1
Germany	GRS	ATHLETE-CD/COCOSYS
Japan	CRIEPI	MAAP 5.01
	IAE	SAMPSON-B 1.4 beta
	JAEA	THALES 2
	NRA(S/NRA/R)	MELCOR 2.1
Republic of Korea	KAERI	MELCOR 1.8.6
Russian Federation	IBRAE/ROSATOM	SOCRAT/V3
Spain	CSN/CIEMAT	MELCOR 2.1-4803
Switzerland	PSI	MELCOR 2.1-4803
US	EPRI	MAAP 5.01
	NRC/DOE/SNL	MELCOR 2.1-5864

The following points are the minimum obligation of the participants:
- To conduct a full scope analysis for at least a single unit among the Units 1 to 3 with the fixed single set of boundary conditions.

- To share the results of the analysis, models and methodologies used, and their physical discussions among all the participants.
- To submit a written report.

3. MAIN RESULTS

3.1. Common case

Since there still exists large uncertainties and controversial opinions on plant behavior, it is not surprising that results of the analyses by the participants scatter in such a wide range. Therefore, the project decided to analyze a “common case” first with a common set of boundary conditions, determined by calculating with simplified approach of mass and energy balance, before the “best-estimate analysis”. The common case analysis was considered to be useful to identify differences in assumptions and physical modeling among the severe codes and analysts.

Figure 1 shows examples of the results from the common case analysis for Unit 1. The figures compare the collapsed water level in-shroud and the RPV pressure during the isolation condenser (IC) operation before the station black-out (SBO) between the analyses and measurement. The calculated collapsed water level in-shroud is remarkably different among the codes. The main cause of the differences is the initial water inventory, namely the operation of IC, employed in the common case by each participant. This issue is critical in Unit 1 where alternative water injection was not accomplished until around 14 h. The common case analysis also showed difficulty induced by the treatment of boundary conditions determined by the participants, especially associated to failure of systems. In the analyses for the RPV pressure in Figure 1, EPRI, KAERI, IAE and SNL assume SRV (Safety Relief Valve) flange gasket leak and SRM penetration pipe failure, IBRAE considered only SRV flange gasket leak, KAERI and IRSN predicted MSL (Main Steam Line) failure, and CIEMAT and IAE predict RPV lower head failure. With little information regarding water level and pressure in this unit at the time of the depressurization it is hard to demonstrate which depressurization method was most likely to have occurred.

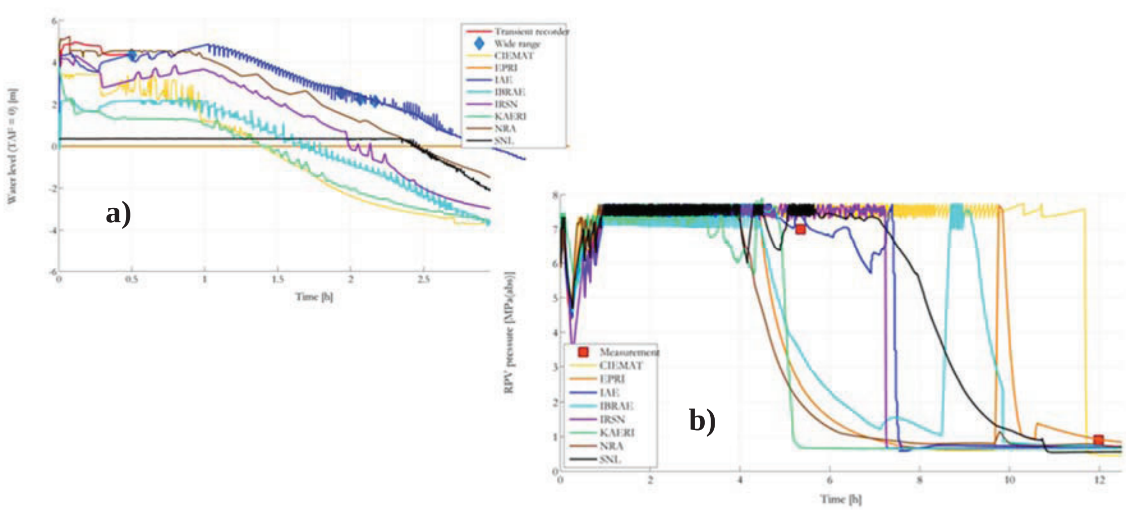

Figure 1. Unit 1 common case results. a) Water level and b) RPV pressure.

As for Unit 2 and 3 where uncertainties of boundary conditions are greater, the differences among the analytical results were much greater than Unit 1 due to “constraint” of the pre-determined boundary conditions in the common case analysis. Detailed technical descriptions and discussion are made in the reference [1].

3.2. Best estimate case

The common case analysis showed relatively wide variation in the results which were caused not only by the difference in the assumptions and modelling among the codes and analysts but also by the "constraint" effect of the pre-determined boundary conditions. In the best estimate analysis, the participants were allowed to adjust boundary conditions, e.g. operation of safety systems, timing of depressurization, leak conditions, the amount of alternative cooling water, according to the interpretation or expectation of the accident for the most uncertainty plant parameters.
The main outputs from the best estimate analysis are:

- Coolant level including timing of the reaching Top of Active Fuel (TAF)
- Hydrogen generation
- Initiation and progress of melt in fuel bundle and control blade
- Timing of SRV flange gasket leakage, TIP leak, MSL rupture, SRV stuck open
- Core plate and RPV failure
- Distribution of molten and solidified materials at three locations (above core plate, on the lower head of RPV and out of the RPV)
- Composition of molten and solidified materials
- Progress of MCCI

though the computable items are not always common among the computer codes.

Figure 2 shows examples of comparisons for the calculated results among the codes. Similarities and differences are technically explained and discussions in the reference [1].

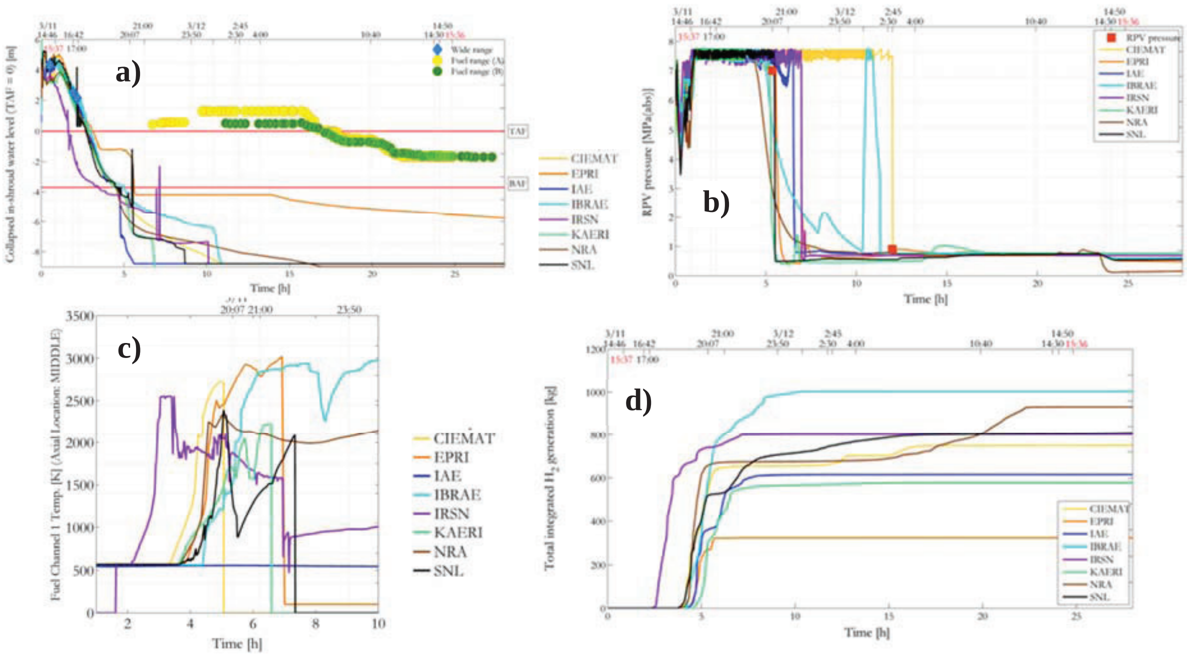

Figure 2-1. Examples of comparisons for calculated results among the codes. a) In-shroud water level, b) RPV pressure, c) Fuel temperature at mid-height, d) Hydrogen generation in Unit 1

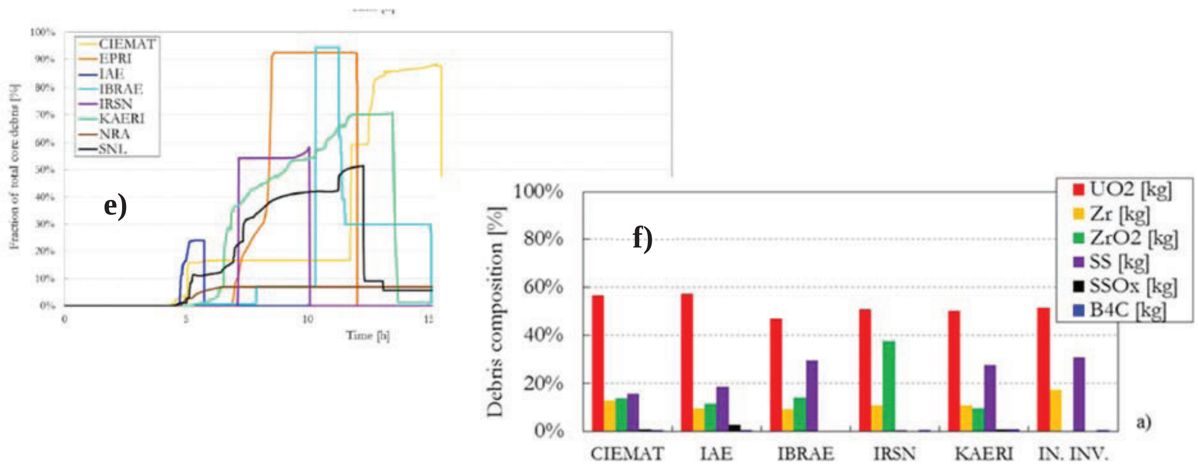

Figure 2-2. Examples of comparisons for calculated results among the codes.
e) Material relocation to RPV lower head, f) Debris composition in Unit 1

3.3. Accident scenario estimated from the benchmarking

In the present paper, the most probable accident scenarios in the three units are briefly described on the basis of the results of the benchmarking.

3.3.1 Unit 1

After the tsunami hit the NPS, all AC and DC onsite power supply was lost. Consensus is reached that the IC which was activated by the operators was finally kept off so that further heat removal from the core became impossible. The RPV pressure increased, limited by an operating SRV, and the water level monotonically decreased. It is likely that the RPV water level reached TAF around 19:00 on 3/11 (around 4 hours after scram). The core temperature increased to levels that the fuel bundle would have melted and collapsed at around 20:00 on 3/11 (around 5 hours after scram) at high system pressure as shown by all code calculations. It is also a consensual agreement that in Unit 1 the pressure boundary failed during the core degradation phase. The effect on the RPV pressure history depends on the assumption of its location. The MSL failure and SRV seizure tend to depressurize the reactor before lower head failure, while penetrations and SRV gasket leakage tend to maintain a relatively large RPV pressure and cause high pressure melt jet ejection into the pedestal region of the containment after the lower head failure.
The detailed mechanisms of core degradation and timing of events are hard to determine from the results of the available calculations. However, consensus is reached that a large portion of the fuel, control rods and core structures melted even though the calculated timing and sequences are slightly different and uncertain to some extent. Thereafter, the RPV failed possibly by either pipe rupture and/or by RPV wall melt-through, which allowed a large amount of corium debris to move into the cavity, where the MCCI phase started (Figure 4). This melt progression to the MCCI likely occurred within 15 hours from scram, before water injection was successfully delivered to the RCS/RPV or PCV.
Consensus among the codes has been reached that the first significant pressure rise in the Dry Well (D/W) was likely associated with the RPV/RCS boundary failure, discharging steam and hydrogen into the PCV. It is possible that the PCV failed after the failure of the lower head, resulting in the second large pressure peak in the PCV, as suggested also from the estimated pressure transients in RPV and PCV (figure 3), even though the location and the causes have not been indicated by the analyses. The timing of this event is very scattered among results.

Water injection by mean of fire trucks on 4:00 on March 12th is likely to be not effective until around 80 hours from scram. However, the further core degradation progression in the core and the pedestal was terminated by the water injection from fire trucks and long-term stable conditions have been reached. The range of overall core degradation (March 17th at 12:00) was between 80% and 100% of the initial in-core inventory. As a common indication from the calculations, the masses contained in the lower plenum were almost instantaneously released to the reactor cavity at the time of the RPV lower head failure.
The MCCI phase most probably started before the water injection by fire truck became effective. Extensive oxidation of the metals contained in the cavity and of the reinforcing bars in the basement is expected. The concrete erosion is supposed to be of a size comparable to the pedestal wall region in the radial direction, with the possibility that the pedestal walls had been weakened by the interactions with the corium. All codes predicted that the cavern erosion in the vertical direction was not extended to the liner and no failure is predicted in this benchmarking.
The calculations are in relative agreement about the volume of generated flammable gases around the time of explosion of the reactor building. The calculated total mass of hydrogen generated by the time of explosion is around 2000 kg (in core production plus MCCI) and that of CO is around 1000 kg (by the corium concrete interaction).

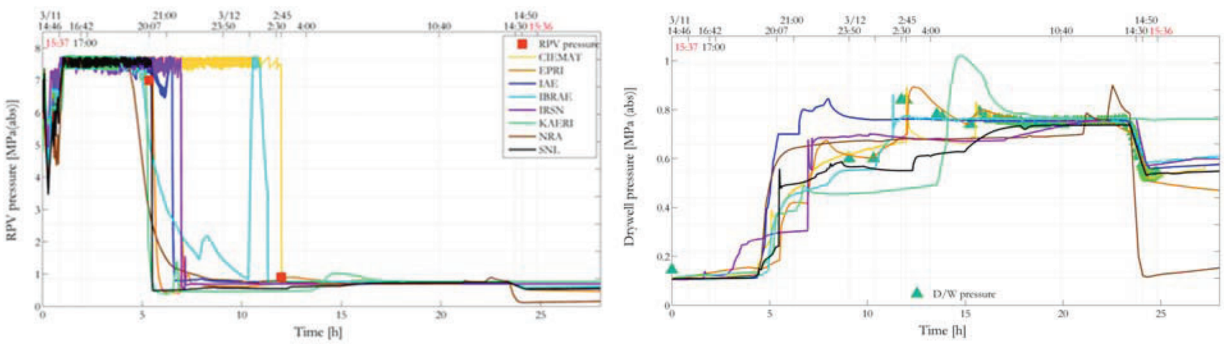

Figure 3. Estimated RPV pressure and D/W pressure in Unit 1

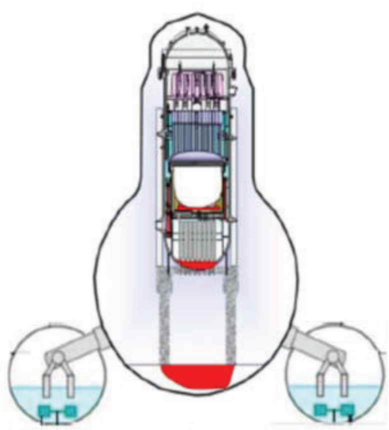

Figure 4. Qualitative description of the plausible Unit 1 status

3.3.2 Unit 2

The Units 2 and 3 have different specifications including reactor power and design of safety systems from Unit 1, as well as conditions that the units experienced during the accident.

The Reactor Core Isolation Cooling (RCIC) turbine was successfully activated to remove the residual heat from the core after the reactor scram just before the attack of the tsunami causing an SBO. Because of the complete loss of AC and DC, the control valve in the steam line remained fully opened and the RCIC system continued its operation even though not controllable either manually or automatically. Consensus exists among the participants that the RCIC worked in design conditions until the RPV water level reached the main steam line at around 2 to 3 hours after scram. The water level would have reached the main steam line and a two phase flow mixture entered the turbine, resulting in decrease in the RCIC efficiency, and on March 12th at 3:17 the water source was changed from the Condensate Storage Tank (CST) to the Suppression Chamber (S/C) due to the low water level observed in the tank. The calculations indicated a significantly lower value of the integral amount of water from the CST, compared to the estimations based on the observed remaining water inventory in the tank. Therefore, there still remain uncertainties for the measured values and the time of water source switch as well as the integral amount consumed at the CST.

After the water source was switched, the RPV pressure remained between 4 to 6.5 MPa(abs), well below the SRV activation level, and the in-reactor conditions were relatively stable until around 68 hours when the RCIC degradation was assumed by all the participants. Calculations of the accident progression with this assumption inferred that steam flow was maintained to the turbine even after the pump stopped injecting water. Eventually also the steam flow to the RCIC system was terminated due to a closing valve possibly by mechanical trip. After RCIC failure, the RPV pressure increased to the SRV relief pressure set point and thereafter the RPV water level started decreasing. All the codes predicted that the RPV water level reached TAF around 75 h. At this time the reactor was depressurized when operators connected an external battery to manually open a SRV, resulting in a faster drop in core water level.

There is consensus among the participants that water injection by means of fire truck started immediately after the manual RPV depressurization and the water was effectively injected into the recirculation line, even though the quantity reaching the RPV may have been lower than the indicated value discharged from the pump. Despite the alternative water injection, the water level continuously decreased to the BAF (Bottom of Active Fuel) after the depressurization followed by fuel temperature increase. The water-zirconium reaction which may be enhanced by the moderate water injection caused the temperature excursion and the core started melting. The majority of the participants predicted that around 20 to 70% of the initial core inventory melted and relocated into the lower head of RPV though the relocated molten materials were completely retained on the lower head (Figure 5). On the other hand, some participants predicted that the whole core mass relocated into the pedestal, resulting in occurrence of MCCI. This discrepancy is considered to be mainly generated by differences in efficiency of the alternative water injection, models associated to molten material relocation from the core plate to the lower head, and models associated to the lower head failure of RPV.

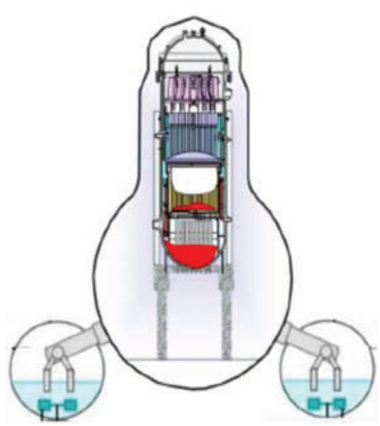

Figure 5. Qualitative description of the plausible Unit 2 status, estimated by the majority of the participants

3.3.3 Unit 3

The RCIC was activated and operated in a controlled manner owing to the availability of DC batteries in Unit 3. The controlled operation of the RCIC turbine prolonged the availability of battery voltage and maintained the RPV water level to some extent. On March 11th at 11:36 the RCIC stopped working because of the increased S/C pressure which triggered automatic trip. The measured PCV pressure shows a much faster increase than generally predicted, especially during RCIC operation and cycling SRV. Though causes of the quick pressure increase are not clarified through the computations, two possible scenarios can be considered; one is steam bypass in the S/C due to the creation of stratification and local hot spot around the sparger in the pool, and another is direct leak from the RPV to the D/W.

In order to decrease the S/C pressure the spray system was activated at 12:06 on March 12th. The spray probably worked with reduced mass flow rate around 10 kg/s and low efficiency due to the relatively large droplet spray. Once the RPV water level reached L-2 the HPCI (High Pressure Core Injection) system was automatically activated at 12:35. The results of the calculations suggest that the HPCI was started with reduced water injection in order to successfully compete with the operation of the RCIC. However, because of the large steam draw of the HPCI and the associated depressurization of the RPV which affected the instruments, the water level was indicated to rise relatively quickly. Seeing the indication, the operator decreased the steam sent to the turbine below the nominal value at low pressure (which is 8 kg/s). Eventually the pressure decreased to a point where the turbine performance is assumed to degrade, resulting in much reduced water injection to the RPV. The extent of the pump-turbine degradation could not be clarified by the computations but a mass flow rate between 0 and 4 kg/s, except for the flow to the CST, was assumed for this phase of the transient by the major part of the participants. The reduced water injection lead to the water level decrease and the measured value was below TAF when HPCI was manually shut down to attempt depressurization. Thereafter, upon the failure in the depressurization the RPV pressure increased and the water level remained unchanged while the fuel temperature slowly increased. Upon reaching the RPV pressure for the SRV operation, the water level decreased and the fuel temperature increased relatively quickly. The temperature increase rate varied among the calculations at those high RPV pressures; however it is likely that the core degradation began around 40 hours from scram, which might be connected with the neutron detection near the main gate of the NPS around that time.

Unlike Unit 1 and Unit 2, some discrepancies are seen among the codes in the prediction of the initiation of the core melting, due to uncertainties related to the degradation of HPCI injection and the resulting core water level, and therefore related to the core temperature excursion phase during SRV cycling prior to RPV depressurization.

The reactor was depressurized for approximately 2 minutes around 9:00 on March 13th, possibly by activation of the Automatic Depressurization System (ADS) or by rupture of the MSL by creep. At this time water injection by the fire truck was started. It was pointed out later that there were by-pass flows to the main condenser and condensate storage tank which decreased the effective mass of water sent to the RPV. The percentage of the injected water that actually reached the core is unknown. Some codes calculated that the lower head of RPV might be intact if 30% of the average water mass flow from the fire truck reached the core, while it might fail if the lower amount reached the core.

Two scenarios for the melt progression are possible from the computations for this unit due to the several remaining uncertainties (Figure 6). In the first scenario, the RPV remained intact and melt retained on the lower head, so that MCCI did not initiate. The molten mass is limited to 40 to 60% of the inventory in this scenario. On the other hand, the predicted total debris masses in the second scenario ranged approximately from 60 to 100% of the core inventory. RPV breach and core debris relocation to the pedestal occurred with onset of MCCI and gas generation in this scenario. It was predicted that 1000−1500 kg of hydrogen was generated in the first scenario while 2500−3500 kg of hydrogen and more than 4000 kg of CO were generated in the second scenario.

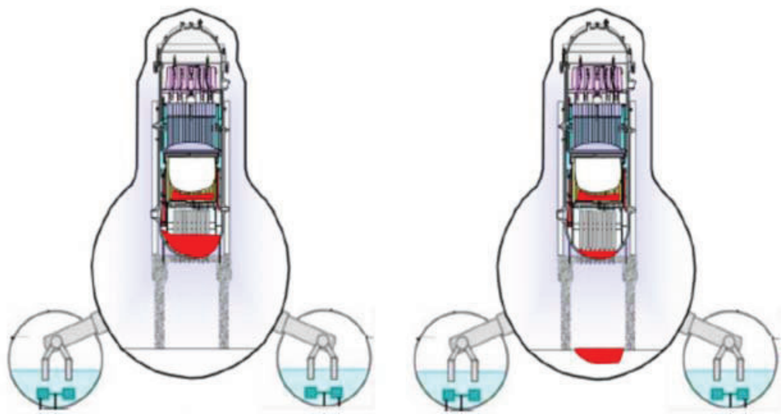

Figure 6. Qualitative description of the plausible Unit 3 status
The first scenario predicts the RPV to remain intact and melt retained into the lower head, while the second scenario predicts RPV breach and core debris relocation to the pedestal

3.4. Implications to the decommissioning at the Fukushima Daiichi NPP

Decommission is steadily pursued by TEPCO in the Fukushima Daiichi NPS. The Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF) works out the strategy for the decommissioning and the plan for the related R&D. The International Research Institute for Nuclear Decommissioning (IRID) manages and conducts R&D [2,3]. In the R&D, characterization of fuel debris, development of methods and equipment for retrieving fuel debris and core internals, development of technologies for criticality control and methods for nuclear material accountancy of fuel debris, and researches for accommodating, transferring and storing of fuel debris are conducted in addition to the development and improvement of the SA computer codes. The information on peak core temperatures, cooling conditions, atmosphere (oxygen potential) inside the reactors, distributions of fuel debris and radioactive materials released from the fuel, compositions of relocated molten materials at different locations is useful in accomplishing the R&D and the decommissioning. The present benchmark study was able to provide information on most probable accident progression and current status inside the RPVs and PCVs under the boundary conditions currently available, though the accuracy of the results is still inadequate due to large uncertainties in the boundary conditions [1].

Based on the results from the present benchmark study, it is estimated that the reactor core was severely damaged and the major portion of the core materials melted and relocated to the lower head of the RPV and further to the pedestal in Unit 1. The low-melting- temperature materials such as components of control blades relocated earlier than the fuel rod materials. However, the macroscopic homogeneity of the fuel debris relocated to the pedestal may be relatively high in this unit, considering the remarkable extent of accident progression. The composition of the relocated debris predicted by the codes was UO_2 (50−60%), zirconium metal (10%), zirconium oxide (10−40%) and stainless steel metal (10-30%). In all calculations the amount of oxidized iron was negligible. All the calculations demonstrated the presence of B_4C in the cavity of the pedestal; the amount was, however, extremely small (<2%). Therefore, the presence of the boron and carbon that can significantly increases hardness of fuel debris may not have the great impact on the mechanical removal of fuel debris, depending on the distribution of these elements. On the other hand, the reduced amount of boron in the fuel debris would increase the difficulty of re-criticality management. The other information such as porosity and chemical homogeneity which are important from the view point of re-criticality management cannot be obtained in analytical studies.

The majority of the calculations suggested that a large portion of molten materials more likely was retained in the lower head of Unit 2 though there still exists possibility of debris release out of the RPV. If most the debris is retained inside the RPV and reflooding can be achieved, simpler methodologies like the TMI-2 case may be adopted for debris retrieval. Decontamination of the reactor building and PCV is another problem in this unit. In the case of a lower extent of core damage and molten material relocation, the fuel debris may not be macroscopically homogeneous and molten materials may be solidified inside or between core components, which would require complicated tools and methodologies in the fuel debris removal.
The current status of Unit 3 is probably the most uncertain, at least from the point of view of the results from the present benchmark study. The calculations gave comparable level of agreement against measurements, the two hypothesized cases of lower head integrity and failure. Further investigations inside the facilities, estimation of the external water injected and model improvement are needed to reach a larger consensus among specialists.
Water levels inside the reactors are estimated from the investigations of the Fukushima Daiichi NPS [4]. They are 3 m, 0.3 m and 6.5 m above the bottom of the pedestal in Unit 1, 2 and 3, respectively. Locations of leakages are being specified; however, additional leakages will be found at higher elevations as the leakage locations at lower elevations are repaired and the water level is increased. In that case, in-air retrieval is the realizable method [2] though various R&D are required to conduct it and various difficulties are expected. Results of analytical studies become more important in those situations.

The present benchmark study did not only provided information about the progress of the accident and the current status inside the reactors, but also it clarified causes of uncertainties for variations in the estimations. They are missing of information on operation and failure of facilities, open/close of valves, etc., missing of information on failure or leakage of PRV and PCV, missing of information on efficiency of alternative water injection, and lack of information on several phenomena and insufficiency of modelling in codes, including BWR-specific phenomena such as failure of core support plate, melt relocation through lower plenum to RPV lower head, steam-zirconium interaction during core degradation, failure behavior of lower head, MCCI including termination of interaction by flooding, RCIC characteristics under beyond-design conditions, RCIC turbine pump efficiency by injecting two-phase flow with low void fraction. The provision of such information is consider to enhance communication between decommissioning and analysis as well as development of further investigations.

Based on the results of the present benchmark study, the next phase of the project is prepared. The objectives of the phase 2 project is to provide information and analysis results on accident progression, FP behavior, and source term estimation to support safe and timely decommissioning at Fukushima Daiichi NPS as well as to contribute to improvement of methods and models of the SA codes. The following items are conducted in the phase 2 project.

- Evaluation on transport of radioactive FPs from degraded fuels, through RPV, PCV, R/B to environment for 3 weeks
- Identification of leakage path from RPV, PCV to environment
- Evaluation on amount of hydrogen generated and accumulated in R/B
- Evaluation on amount and chemical form of FPs released into environment
- Active interactions with related research areas including prediction of environmental diffusion of radioactive materials

4. CONCLUSIONS

The OECD/NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF) Project was established in November 2012. Fifteen organizations of eight countries calculated thermo-hydraulic behavior with severe accident integral codes for the time span of about six days from the occurrence of the earthquake. The calculated results submitted by the participants were compared and evaluated to estimate the accident progression and status inside the reactors. Still remaining uncertainties and data needs that are useful to the communication between analysts and decommissioning activities are also summarized as the output from the project.

In actuality, it is very difficult to predict the accident progression in the Fukushima Daiichi NPS, since the accident progression is a consequence of complicated combinations of many phenomena. However, computer codes are valuable tools to estimate the status inside the contaminated reactors that are hard to access. In order to increase the accuracy and obtain reliable results for the decommissioning, boundary conditions and models are required to be continuously improved based on the information and data from the Fukushima Daiichi NPS and related studies.

The next phase of the project is conducted with the aim of:

- To provide information and analysis results on accident progression, FP behavior, and source term estimation to support safe and timely decommissioning at Fukushima Daiichi NPS, and
- To contribute to improvement of methods and models of the SA codes.

NOMENCLATURE

ADS: Automatic Depressurization System
BAF: Bottom of Active Fuel
CST: Condensate Storage Tank
D/W: Dry Well
HPCI: High Pressure Core Injection
MSL: Main Steam Line
PCV: Primary Containment Vessel
RCIC: Reactor Core Isolation Cooling
RPV: Reactor Pressure Vessel
SA: Severe Accident
SBO: Station Black Out
S/C: Suppression Chamber
SRV: Safety Relief Valve
TAF: Top of Active Fuel
TMI-2: Three Mile Island Unit 2

ACKNOWLEDGMENTS

The contents of the present paper are the summary of the OECD/NEA benchmark study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF project) conducted with strong contributions of the specialists of eight participant countries. The authors acknowledge and express their appreciation to the experts, Mr. Greg Lamarre, Mr. Andrew White and Mr. Toshiyuki Koganeya who devoted the time and efforts as the NEA secretariats.

REFERENCES

1. M. Pellegrini et al., “Benchmark study of the accident at the Fukushima Daiichi NPS, Summary report,” *Proceedings of The 16th International Topical Meeting on Nuclear Reactor Thermal Hydraulics (NURETH-16),* Chicago, IL, USA, August 30-September 4, (2015).
2. H. Yamana, “Rebuilding the Strategy for the Decommissioning of Fukushima Daiichi NPP, Recent Situation of NDF’s new Approach,” *Proceedings of International Experts’ Meeting on Strengthening the Effectiveness of Research and Development in the Light of the Fukushima Daiichi Power Plant Accident*, Vienna, Austria, February 16 – 20, (2015).

3. "Research and development, International Research Institute for Nuclear Decommissioning," http://irid.or.jp/en/research/.
4. J. Matsumoto, "Current status on Fukushima Daiichi NPS," *Proceedings of Physics of Reactors Topical Meeting 2014 (PHYSOR2014),* Kyoto, Japan, September 28-October 3, (2014).