BENCHMARK STUDY OF THE ACCIDENT AT THE FUKUSHIMA DAIICHI NPS BEST ESTIMATE CASE COMPARISON

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Abstract

The Great East Japan earthquake, occurred on March 11th 2011 at 14:46, and the subsequent tsunami led the TEPCO's Fukushima Daiichi Nuclear Power Station (NPS) to endure beyond design basis accident. After the accident, the Japanese government and TEPCO compiled the roadmap towards an early resolution to the accident including, among the main activities, the employment and development of severe accident (SA) computer codes. In the member countries of the Organization for Economic Cooperation and Development, Nuclear Energy Agency (OECD/NEA), SA codes have been developed after the accident at the Three Mile Island Unit 2 and widely employed to asses NPS status in the postulated SA conditions. Therefore, the working plans have been set up to conduct a benchmark study of the accident for the Fukushima Daiichi NPS units 1-3 with the country members of the OECD/NEA, using SA codes, constituting an international program named Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF).

The objectives of the BSAF project are: to analyze the accident progression of Fukushima Daiichi NPS, to raise the understanding of SA phenomena, to contribute to the improvements of methods and models of the SA codes and to define the status of debris distribution in the reactor pressure vessels and primary containment vessels for decommissioning.

The present technical paper summarizes the achievements obtained through the results' comparison, emphasizing the portions of the accident where all the participants reached a common consensus and identifying still open questions where future work should be directed. Consensus exists on the current condition of the Unit 1, where a large fraction of the fuel is assumed to have relocated ex-vessel. On the other hand larger uncertainties exist for Units 2 and 3 where in-vessel and ex-vessel scenarios produce a reasonable prediction of the accident progression.

Keywords: OECD/NEA, BSAF, SA codes, decommissioning

1. INTRODUCTION

The Great East Japan earthquake occurred on March 11th 2011 at 14:46. At the earthquake onset the units at the TEPCO's Fukushima Daiichi Nuclear Power Station (NPS) were successfully shut-down. However, the subsequent tsunami led the TEPCO's Fukushima Daiichi NPS to endure beyond design basis accident. Even though direct evidence of the core melt has not been attained yet, it is believed that the units 1 to 3 experienced severe accidents involving core meltdown, deriving from the total or partial loss of the core cooling capabilities.
After the accident, the Japanese government and TEPCO compiled the "Roadmap towards Restoration from the Accident at Fukushima Daiichi Nuclear Power Station" promoting the Research and Development Plan towards the Decommissioning of the Fukushima Daiichi NPS which includes the analysis of the accident progression at the units 1-3 and their current status.
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. Taking account of the above circumstances, the working plans have been set up to conduct a benchmark study of the accident progression for the Fukushima Daiichi NPS units 1-3 accident with some of the OECD/NEA member countries using computer codes and methods of analysis, constituting an international program named Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF). The BSAF phase I project was finally launched in November 2012 with the participation of sixteen organizations of eight countries (Table I).

Table I Participants and codes employed in the BSAF project phase I.

		CODE	COUNTRY	ANALYZED UNITS
1	CEA	analytical study	FRANCE	1
2	CIEMAT	MELCOR 2.1-4803[1][2]	SPAIN	1 – 2 – 3
3	CRIEPI	MAAP 5.01 [3]	JAPAN	2
4	EPRI	MAAP 5.01	U.S.A	1 – 2 – 3
5	GRS	ATHLET-CD/COCOSYS[4][5]	GERMANY	2 – 3
6	IAE	SAMPSON-B 1.4 beta [6]	JAPAN	1 – 2 – 3
7	IBRAE/ROSATOM	SOCRAT/V3	RUSSIA	1 – 2 – 3
8	IRSN	ASTEC V2.0 rev3 p1 [7]	FRANCE	1 – 2 – 3
9	JAEA	THALES [7]	JAPAN	2 – 3
10	KAERI	MELCOR 1.8.6 [1][2]	SOUTH KOREA	1 – 2
11	NRA(S/NRA/R)	MELCOR 2.1 [1][2]	JAPAN	1
12	NRC/DOE/SNL	MELCOR 2.1-5864 [1][2]	U.S.A	1 – 3
13	PSI	MELCOR 2.1_4203 [1][2]	SWITZERLAND	3

The objectives of the project are:
(1) To analyze accident progression of Fukushima Daiichi NPS utilizing the common information database,
(2) To raise the understanding of SA phenomena, which took place during the accident, through comparison with participants' analysis results and with measured plant data,

(3) To contribute the above results to improvement of methods and models of the SA codes applied in each participating organization, in order to reduce uncertainties in SA analysis and validate the SA analysis codes by using data measured through the decommissioning process,
(4) To contribute analysis results on accident progression, status in the Reactor Pressure Vessels (RPVs) and Primary Containment Vessels (PCVs), and status of debris distribution to a future debris removal plan.
Analysis range deals with in plant phenomena until 12:00 March 17th 2011.

2. Best Estimate Case Results

The BSAF project represents a unique example of benchmark against a complex accident scenario where the knowledge of the boundary conditions is still limited due to existing unknowns of the accident progression. Therefore, a common set of boundary conditions was proposed and discussed among the operative agent and the participants but institutes were free to modify them based on the their interpretations of the available facts. Participants were requested to provide a single best estimate case for the final comparison in which assumptions were made known for a deep comparison. Due to the length of the present paper the detailed explanation of the accident has not been included and interested readers might refer to TEPCO's reports [1] or other paper of the same series.

2.1 Unit 1

In the perspective of plant accident investigation the Unit 1 case represents an example to confirm the responses of an NPP in one of the most severe cases when almost no action was effective to avoid core degradation. Indeed, with the exception done for the Isolation Condensers (IC) employment, which basically served as ultimate heat sink until SBO maintaining high water level and pressure, the accident progressed without any intervention of the operator until the onset of core degradation. From the point of view of comparison among SA codes this translates in the chance to investigate the effect of intrinsic differences among codes (e.g. nodalization, physical models and failure criteria) without large interference introduced by different interpretations of the boundary conditions, since they are reduced at the minimum for this case.

The calculations are coherent computing the core water level decreasing monotonically without any effective way that could recover the water level until the end of the core degradation (Figure 1). It can be stated that the water level reached Top of Active Fuel (TAF) between 17:30 and 18:00 on March 11th, around 3 hours after scram. Calculations remain coherent until the prediction of the first corium (melt or particle) movement across the core, which is predicted to start between 18:45 to 19:15 on March 11th. It is remarkable that the core degradation phase started earlier than any possible attempt to inject water into the core for Unit 1, since the first injection (not effective) started at 20:50 on March 11th by the Diesel Driven Fire Pump (DDFP).

From the calculations the Reactor Cooling System (RCS) (pressure boundary) failure is likely to have occurred prior to melt attack in the lower plenum of the RPV. Such event might have been created because of hot gases flowing through the Main Steam Line (MSL) or hot temperature achieved in the core region. The former acting on the gasket, creep or seizure of the Safety Relief Valve (SRV), the latter weakening the materials of RPV penetrations.

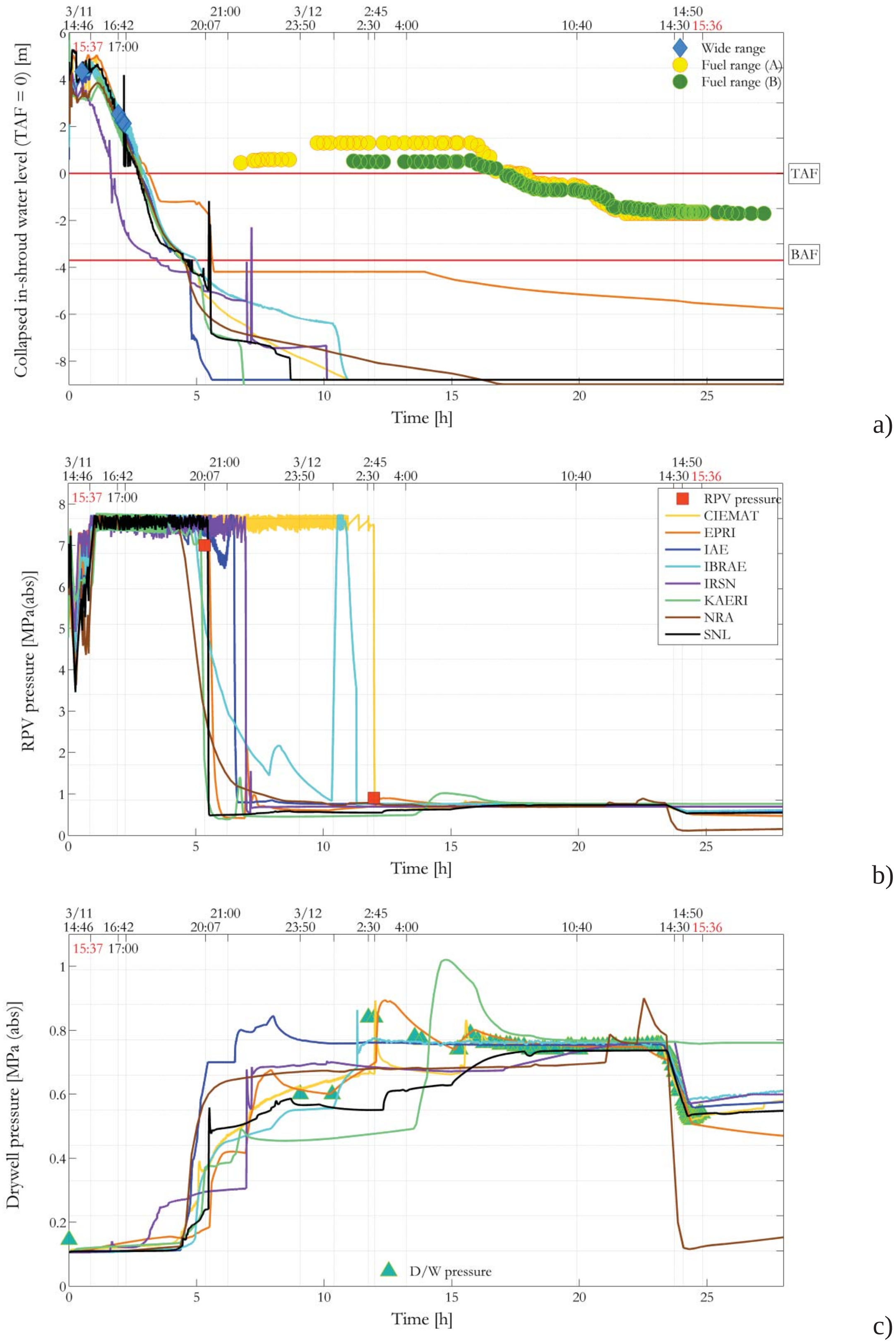


Figure 1 Comparison against measurements and simulations in Unit 1. a) In-shroud water level, b) RPV pressure, c) D/W pressure.

The RPV pressure reduction in Figure 1 b) denotes the timing and kind of adopted failure. Uncertainty exists on the timing and location of its occurrence and whether the RCS failure area would be large enough to trigger a leak or a quick depressurization. Four possible RCS failure points were identified: SRV seizure during hot gases flow, MSL creep rupture, SRVs gasket failure during hot gases flowing and instrumentation piping failure by buckling phenomenon. RCS failure results in the PCV pressure rise presented in Figure 1 c).

Because of the delay in the water injection, corium is expected to relocate eventually into the lower head. Thereafter RPV failure is predicted to have occurred in all the calculations. Uncertainty in the computations exists on the kind and timing of the lower head failure. In Figure 1 c) the failure of the lower head is identified by a sudden pressure rise in the calculations, showing the variability. Also values of failure area, discharge criteria are not similarly defined among calculations and appear to be largely different. In Unit 1 only one manual vent was effective at around 20 h. All calculations predict Molten Corium Concrete Interaction (MCCI) to have already started earlier than the venting activation and consequently the Dry Well (D/W) pressure increase would be dependent on the balance between gas generated during MCCI and leakage assumed in the D/W through top head flange and/or other penetrations (e.g. cable penetrations). Hydrogen generation masses agree among results as long as the core is intact, indicating a comparable generation rate. Thereafter, once the corium relocation begins, trends and absolute values are largely different among computations with a mass variation from around 350 kg to around 1000 kg. Large variability exists for the gases generated during the MCCI phase. The variability might be created because of the uncertain composition of the concrete (e.g. siliceous or limestone based concrete, presence or not of steel bars) whose information was not been available at the beginning of the phase I. Refined consideration of the MCCI will be attempted in the phase II of the same project.

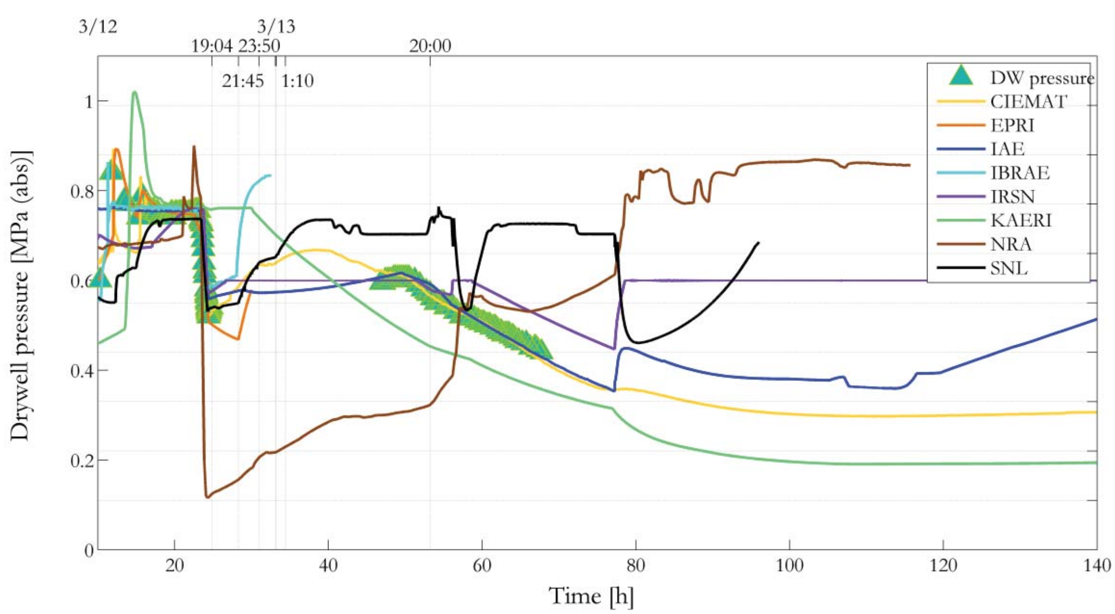


Figure 2 Unit 1 PCV pressure until the end of the transient.

2.2 Unit 2

Regarding the accident progression in Unit 2, several unanswered questions exist. A still debated is the possibility of the beyond design operation of some safety systems, such as the Reactor Core Isolation Cooling (RCIC) turbine-driven-pump, in what could be defined self-controlling conditions without the availability of DC power. Several reasons coexisted to guarantee such behavior, such as the possible robust RCIC turbine design in two phase flow operation, low pressure increase in the Suppression Chamber (S/C), and the disabled shut down signal due to DC loss. With the above assumptions all the calculations could recreate satisfying predictions of the water level (Figure 3 a) and RPV pressure until 70 h (Figure 3 b) from scram when all calculations assumed the RCIC turbine to stop working. All institutes assumed heat loss at the S/C, which has been modelled as heat loss to the torus room assumed flooded, to reproduce the relatively low pressure increase in the S/C.

Unit 2 accident progression represents also an example to illustrate the importance of the procedures to reflood the core possible in a Boiling Water Reactor (BWR) by means of external water injection (without boron addition). It is likely that the water level decreased to TAF just before the RPV depressurization. After depressurization the core experiences high temperatures and oxidation started, partially melting the fuel earlier than any efficient reflooding of the core. Attempts to reflood the core by injecting small amounts of water could have enhanced the core degradation due to augmented water-zirconium interaction.

During the core degradation and attempt to reflood large hydrogen has been generated increasing the pressure around 0.8 MPa (Figure 4) when the PCV, due to the failure of manual venting activation, is predicted to leak by all the institutes as a self-venting at the top head flange. Around 90 h then a loss of integrity has been assumed by the majority of the calculation in order to predict the relatively quick pressure decrease observed. It is likely that the PCV had failed in the weak parts of the structure (e.g. penetration cables, head flange) but the extent of the failure cannot be estimated so far.

The analyses show a wide spread of results related to the core degradation, melt generation and relocation into the lower plenum. Three calculations predict RPV failure and melt release into the pedestal, and two of this group compute MCCI occurrence thereafter. Six of the calculations compute in-vessel retention of the melt. In general the extent of the core degradation seems to be the smallest comparing all three units. Whether RPV lower head failure due to melt attack has occurred or not cannot be judged from the present computations. A small failure at a pipe penetrating the RPV bottom cannot be excluded as water injected into the RPV reaches the pedestal. In general this remains an open issue.

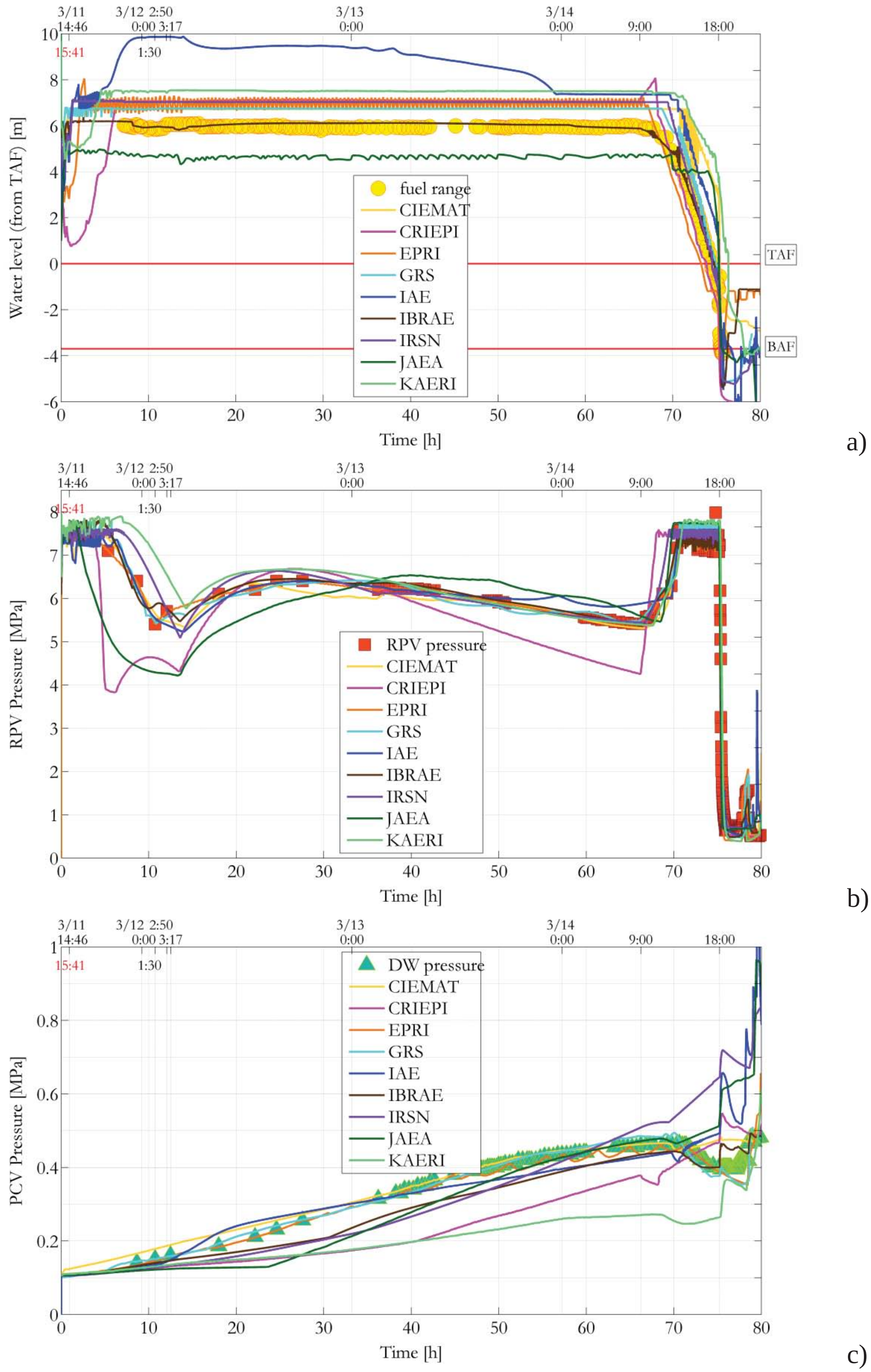


Figure 3 Comparison against measurements and simulations in Unit 2. a) In-shroud water level, b) RPV pressure, c) D/W pressure.

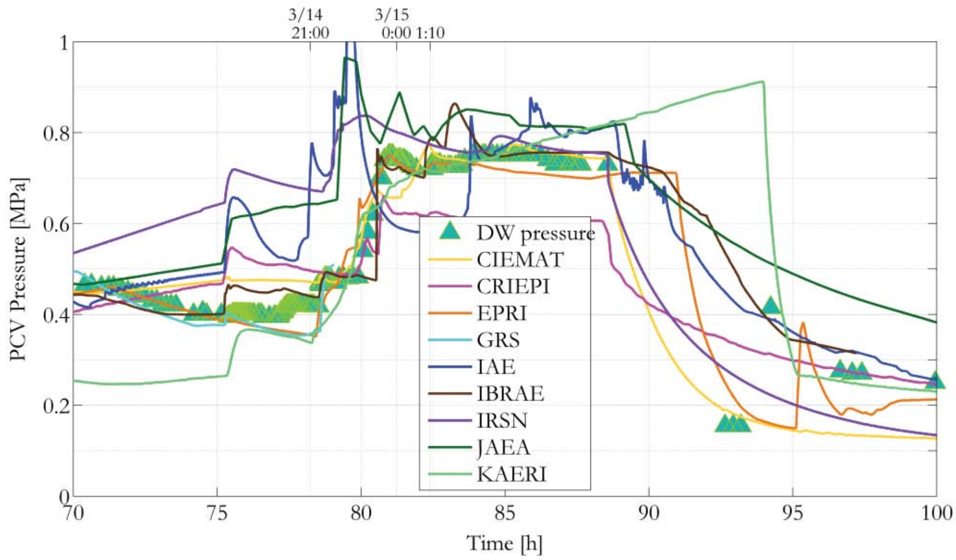


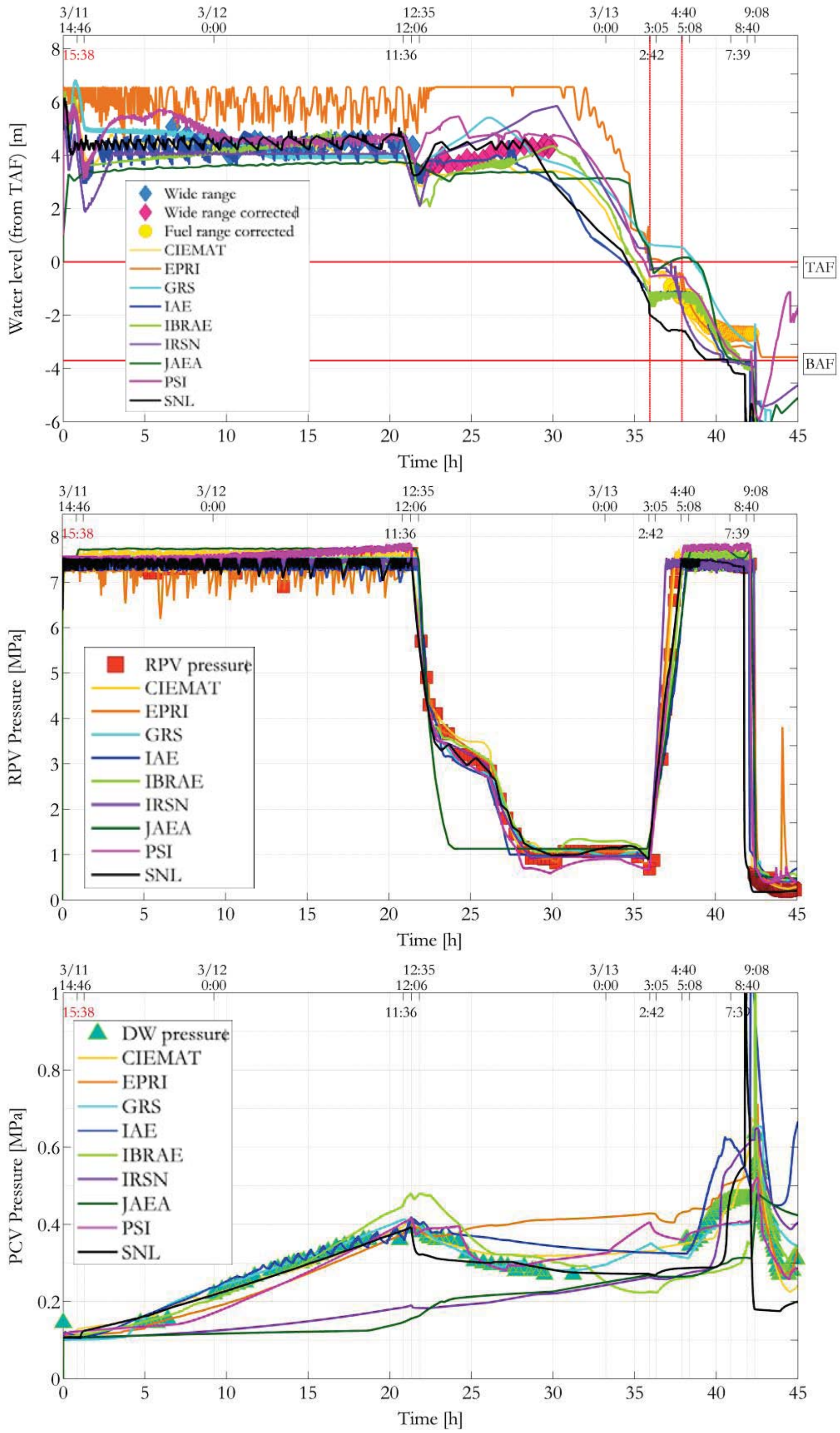
Figure 4 Unit 2 PCV pressure until the end of the transient.

2.3 Unit 3

Regarding the transient occurred in the Unit 3 several open issues exist related to the operation of the safety systems (e.g. RCIC, High Pressure Core Injection "HPCI" and spray) during extensive operation with limited DC power available. In contrast to Unit 1 and Unit 2, the available DC power in Unit 3 is the reason for a larger amount of measured data (e.g. pressure strip chart was working for around 2 days, water level values, PCV temperature) and therefore the case against which the codes performance might receive the largest validation.
On the other hand the degradation of water injection capabilities during the operation of HPCI and the loss of measurements during such conditions, introduce large uncertainties that can influence widely the prediction of the core relocation and subsequently to in-vessel or ex-vessel continuation of the accident.

It could be stated that the RCIC operation is well interpreted by all the calculations and predictions of the RPV water level and pressure reproduce the observations (Figure 5 a and b). The pressure rise in the S/C is still debated. The majority of the calculations obtain a satisfying agreement assuming either steam leak from the D/W to the Wet Well (W/W) or modeling the effect of stratification in the S/C. The uncertainties existing on the degradation of HPCI and also regarding the temperature excursion phase during SRV cycling result in divergence in the prediction of the core melt. During the HPCI phase the RPV water level starts to differ between the calculations with considerable variations in the time to reach TAF and Bottom of Active Fuel (BAF) (Figure 5 a). We can assume a considerable difference whether core degradation starts before or after RPV depressurization time. RPV depressurization is assumed a characteristic time because it is discriminating the possibility to start external water injection.

Most of calculations assume, in accordance with the benchmark prescriptions, that the Automatic Depressurization System (ADS) was operated at around 9:00 on March 13th. ADS opens 6 SRVs simultaneously which provide a good agreement with the depressurization time.



a)

b)

c)

Figure 5 Comparison against measurements and simulations in Unit 3. a) In-shroud water level, b) RPV pressure, c) D/W pressure.

However the possibility was brought up by SNL that one MSL might have failed due to creep providing also a good agreement with the quick pressure decrease. Details of the creep rupture model are not known but it is coherent among calculations of the same institute between Unit 1 and Unit 3.

After the depressurization the external water injection started by mean of fire truck with several interruptions due to fuel or water depletion and hydrogen explosion on March 14th at 11:01. Participants' results indicate that large uncertainty exists on the amount of water effectively transferred to the RPV recirculation line. Computations assume from 50% to less than 10% of the water discharged from the pump, whose integral amount is known.

For most of the calculations the vents are assumed to start at the timing which was provided by the BSAF operative agent, and from TEPCO investigation [9], with some exceptions. In particular doubts exist on the activations from the 3rd vent. Also the mass flow rate is considerably different among calculations and reason might be the description of the valve based on the open area which might not be general for all the participating codes. In the direction to uniform the information provided among participants the valve characteristic will be provided in the next phase of the benchmark.

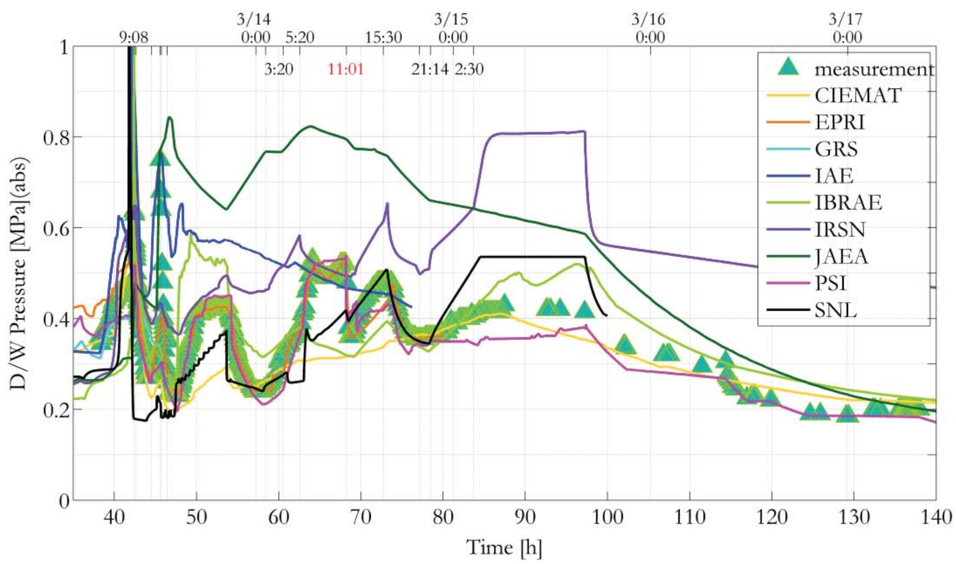


Figure 6 PCV pressure until the end of the transient.

Given the large variability of the results the conditions of the RPV and PCV are scattered among participants and a unique expected condition of the plant cannot be quantified. However in a general synthesis we could state that in case in-vessel scenario is predicted the assumption of a reasonable vent size would be enough to qualitatively reproduce the measured PCV pressure transient, until the time before the hydrogen explosion, while in case of MCCI occurrence given the much larger non condensable gas generation compared to the in-vessel case, the assumption of PCV failure is a necessary assumption to predict the containment pressure transient. Given the large variety among results regarding the core degradation phase, the boundary integrity is also scattered among results. As a general trend those calculations that compute failure of the RPV assume either PCV failure or relatively long leak to maintain realistic pressure. On the other hand in-vessel scenario cases do not usually require the assumption of a failure of the containment and leak is assumed by head flange self-venting only when the pressure exceeds for a limited time period a threshold pressure. It should be noticed that also in the case of in-vessel retention, as presented by two participants, the possibility of PCV failure exists. Indeed, the large temperatures produced also considering the only in-vessel phase might result in PCV loss of integrity. Since a predominant tendency to either ex-vessel or in-vessel cannot be identified and several cases in both scenarios provide an acceptable agreement to the PCV pressure, the boundary integrity is for this unit is still an open issue.

3. Implications on the Reactor Defueling and Decommissioning

The BSAF phase I did not intend to give detailed information of the core debris characterization such as morphology (e.g. particulate size, crust), temperature and movability. The results, presented in Chapter 4, focus more on global values representing the localization of the debris in the three main region of importance (i.e. core region, lower head and PCV) though the debris compositions were also provided. In addition the BSAF project does not aim to indicate the decommissioning strategies, such as which unit to begin the defueling with, which part of the plant should be challenged first or other methods (e.g. dry or wet decommissioning). However, it gives vital information which could be necessary to inform and develop the above strategies, reducing costs and exposure time.
It should be mentioned that ex-vessel scenario are expected in the Fukushima Daiichi accident and that the units in the site add a further level of complexity for the defueling activities, which is represented by the complex configuration of the lower head. Characterization of the crust created inside and outside the structures and on the lower head wall might be the starting point of a novel approach to defueling, if compared to what done in the past.

Every calculation predicted large damage occurred at the Unit 1 core region (large relocation of core debris into the lower head) and further relocation in the pedestal. Results present almost the totality of the core debris in the pedestal region where it eroded the concrete with a variable size.
Figure 7 present the results of the debris composition in the containment cavity at the end of the calculations, that is to say around 144h or when the calculations reached stable conditions. Around 50% of the debris is composed of UO_2. For almost all calculations around 10% of the relocated zirconium is in oxidized form while other 10% in metallic form in accordance with fast evolving transient, as in the Fukushima case. For all calculations the amount of oxidized Fe is small or nearly negligible while stainless steel ranges from 10% to 30% for almost all calculations. Also the presence of B_4C should be expected in the cavity but all calculation demonstrate that this amount, in accordance with the initial inventory (IN. INV in the figure), represents an extremely low value in percentage.

The majority of the calculations presented above denote that large core debris retention in the lower head is likely in Unit 2 but still possibility exists that debris could be transferred ex-vessel, likely as frozen materials on the structures and penetrations.

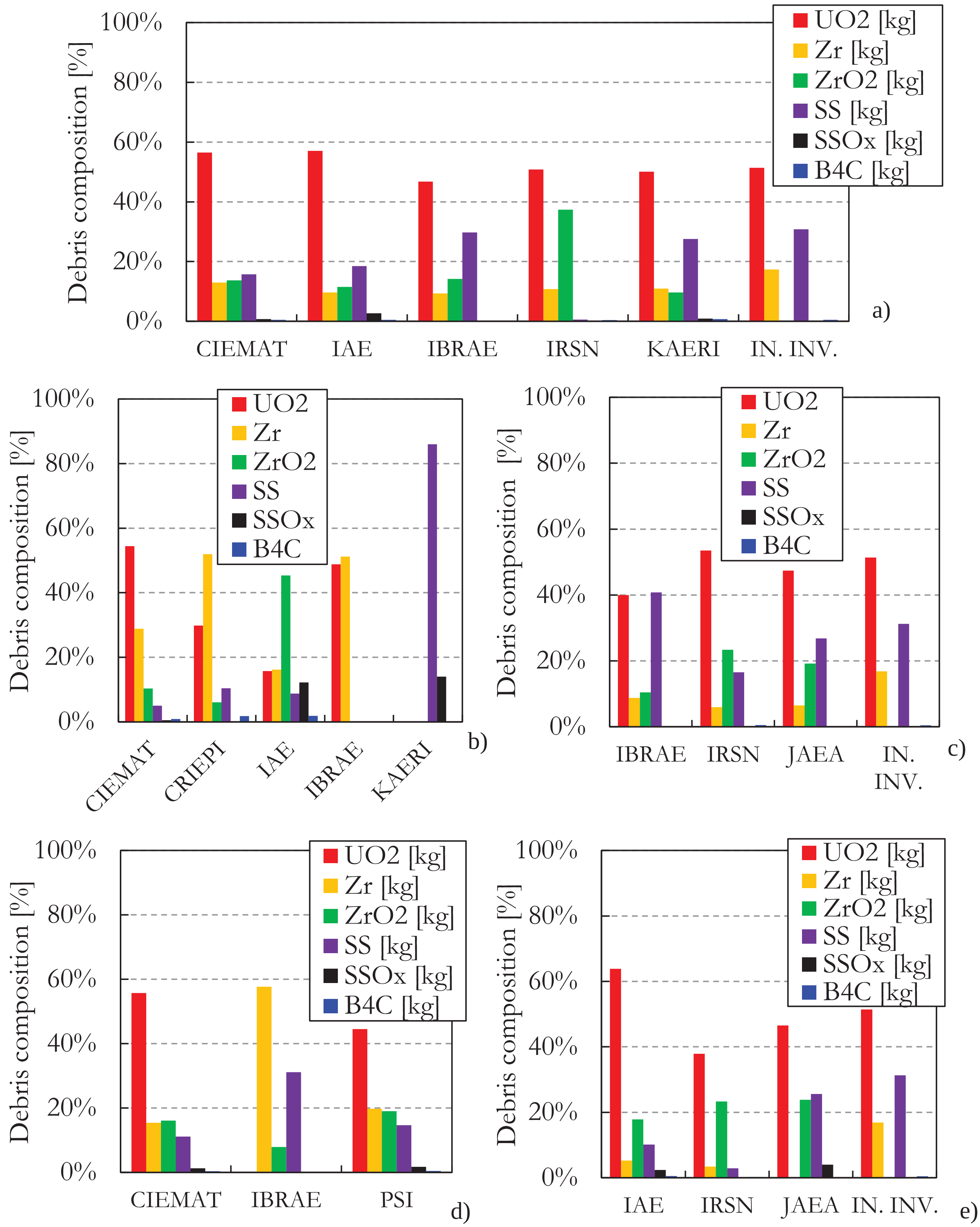


Figure 7a) Unit 1 ex-vessel, b) and c) Unit 2 lower head and ex-vessel, d) and e) Unit 3 lower head and ex-vessel.

The debris composition in the lower head presents variability for those institutes that predict in-vessel scenario (i.e. CIEMAT, CRIEPI, IAE and KAERI). In particular the grade of oxidation (Zr or SS) in the composition is variable. CIEMAT and CRIEPI present relatively little oxidation of both metals and the majority of the debris is composed of Zr and UO_2 (IBRAE presents only UO_2 and Zr). IAE results, on the contrary, present majority of large oxidation in the lower head of both Zr and SS. Institutes predicting in-vessel scenario present B_4C masses in the lower head which might be larger than in the case of ex-vessel scenario. The present consideration might have implication on the decommissioning activities including development of defueling tools. As in Unit 1, the compositions in the pedestal (in case of ex-vessel scenario) agree qualitatively among the calculations and with the initial inventory (IN. INV in Figure 7 c). Most of the calculations present a considerable amount of oxidized zirconium and negligible B_4C masses.

As introduced above the current status of Unit 3 is probably the most uncertain among the other three units, at least from the point of view of the results. Calculations give comparable level of agreement against measurements in the two hypothesized cases of lower head integrity and failure. Further investigations, estimation of the external water injected and model improvement are needed to reach a larger consensus among analysts. Results of the composition are presented for both the lower head and cavity in Figure 7 d and e). Despite the different accident progression assumed by CIEMAT and PSI regarding the amount and timing of core reflooding, the two results present a comparable agreement in the grade of oxidations of zirconium and SS and the percentage of UO_2. On the contrary IBRAE, assuming a much quicker reflooding indicates that UO_2 did not relocate into the lower head and only a small amount of ZrO_2 could relocate into the lower head.

Regarding the ex-vessel scenario, as in the previous two units, comparable agreement exists among results presenting a relatively large grade of oxidized Zr, majority of UO_2 and almost negligible B_4C.

3.1 Possible Repercussions on the Decommissioning Activities

Defueling is the essential step in the whole decommissioning of the plant and is a necessary phase to:

- Preclude the possibility of recriticality in the reactor
- Reduce the contamination potential to the environment.

Three Miles Island Unit 2 (TMI-2) is the only example of an operating power Light Water reactor (LWR) plant, which experienced fuel damage and underwent an extensive defueling phase. It is straightforward to take it as reference point for our purposes. In TMI-2 the real condition became known through both measurements and internal visualization while in case of Fukushima Daiichi and the present project, computational analyses were intended to provide this picture. As described in the results important points of agreement in the development of the accident scenario and in the debris location and composition can be drawn comparing the best estimate results. However, as in the nature of the SA codes, the obtained information represent integral values (not local) and might reach a larger consensus through model refinement and further availability of measured data.

Two examples of possible features of BWR accident core debris that may pose challenges during recovery are suggested here:

1. Effect of larger zirconium inventories;
2. Effect of boride and carbide inclusions in debris.

It shall be remembered that, at the present state of knowledge, there are no assurances that these differences will be encountered in the recovery of the Fukushima reactors nor is there assurance that the two above are the only differences that may be encountered during the defueling process.

3.1.1 Effect of larger zirconium inventories

BWRs have larger inventories of zirconium metal than Pressure Water Reactors (PWR) do. Whereas nearly all the degraded zirconium in the TMI-2 reactor core debris was probably oxidized to zirconium dioxide (ZrO_2) in the slowly progression accident, this may not be the case in debris produced in the damaged Fukushima reactors. The debris produced in BWR accidents will be initially a molten mixture of Zr, ZrO_2 and UO_2, as well as some stainless steel. The high temperature melt will have a substoichiometric composition usually depicted as $(U,Zr)O_{2-x}$. The phase stability region of substoichiometric mixtures of this type narrows dramatically upon cooling. That is, upon solidification a single-phase mixture of $(U,Zr)O_{2-x}$ will become unstable and will segregate into two phases: a nearly stoichiometric oxide phase, designated $(U,Zr)O_{2.00}$ and a metallic alloy phase composed of uranium, zirconium and possibly stainless steel constituents (Fe, Cr, Ni, Si, etc…). The metallic alloy is likely to be present as nodules embedded in the oxide matrix. Comminution or fragmentation of the debris will expose the high surface area metallic nodules. Underwater, these nodules will react to form hydrogen. In air, these nodules may be pyrophoric and react with air producing voluminous aerosol of largely UO_2 (U_3O_8) contaminated with various fission products.

3.1.2 Effect of boride and carbide inclusions in metallic debris

BWRs use boron carbide (B_4C) control material encased in stainless steel cruciform blades. It is anticipated that during core degradation the boron carbide will dissolve in stainless steel. The interaction of boron carbide with stainless steel leads to an exothermic heat of solution of boron carbide in the steel. There are deep eutectics between the boron and carbon and the constituents of stainless steel. Consequently, fluid melts are formed from the control blades at modest temperatures. These melts will flow from the fueled regions and solidify on lower, cooler structures in the core. The solidified material will not be a single phase material. It will be instead a multiphase mixture of stainless steel with embedded borides and carbides of iron and chromium. There will also be precipitated borides in the solidified melts. Such two-phase mixtures are notorious for being extremely hard (e.g. typical metal carbides have Knoop hardness in excess of 1500). They pose challenges to usually cutting tools. The metallic matrix will tend to gall and clog cutting surfaces. The hard carbide precipitates will tend to abrade cutting surfaces – even diamond cutting surfaces and especially diamond bonding materials and mounts. Cutting such solidified metal will present completely different characteristics compared to ordinary stainless steel.

Furthermore, the solidified metals within the damaged core may well be quite porous especially if formation of melt took place while the reactor coolant system was pressurized. Molten metals have a much higher solubility for gases, such as hydrogen, than do solidified metals. Gases dissolved in the molten metals will come out of solution during solidification to produce large voids (perhaps over a centimeter in diameter). These voids will *"grab"* cutting tools and complicate removal of the solid materials.

4. CONCLUSIONS

As in the previous units large uncertainty in the physical interpretation is given at the onset of debris relocation within the core region. Uncertainties are represented by criteria, interactions with the structures, relocation paths and assumed/calculated surface area. Such models have deep impact on the hydrogen generation rate during degradation, possibility and timing of core plate failure, shroud failure and RPV breach.

In the present chapter a comparison of the computed results was attempted. Similarities and differences were explained and discussions provided. It has been highlighted, as a typical result of SA codes comparison, that in case the boundary conditions (e.g. geometry, input values for safety systems) are known, all the codes provide comparable agreement of the thermal-hydraulics phase, as well as the fuel temperature excursion phase. It has been underlined however that divergence exists once the geometry is altered during the relocation process. An attempt to identify the influence of the employed models during relocation and common results has been performed during the discussion even though, due to the many dependency of the problem, uncertainties still exist.
Regarding the effective development of the accident in the three units and the current status several common understanding was reach in the computations, but various differences in the computations exists also in macroscopic descriptions, such as in-vessel or ex-vessel possibilities in Unit 2 and Unit 3. It is necessary to remark that in the present phase only similarities and differences were objectively presented without attempting to filter insights on the accident based on those calculations which provide a better agreement against the measured values. The reason is that the main objective of the present phase was to advance the understanding of the accident and bring up the main limitations of SA codes and a requirement for the decision of more or less correct calculations was not stated for all the participants during the development of the activity. Nevertheless, in the next phase it might be considered minimum requirements for the simulations to be considered in the discussion. This general approach could incentive the achievement of better agreement and might skim several results, among the multitude of data, to produce more concrete information to benefit the decommissioning activities.

The above considerations imply that caution shall be taken during the recovery of debris from the Fukushima Daiichi reactors to consider the risk of pyrophoric reactions, the possibility of hydrogen production during debris reduction and the production of aerosol and dust containing radioactive materials. Also it might be prudent to garner some experience with the remote cutting of solidified, porous, multiphase metals prior to attempting recovery of metals from the damaged reactor cores.

5. ACRONYMS

ADS	Automatic Depressurization Signal	MSIV	Main Steam Isolation Valve
AO	Air Operated	MSL	Main Steam Line
APD	Active Personal Dosimeter	NPS	Nuclear Power Station
BAF	Bottom of Active Fuel	RCIC	Reactor Core Isolation Cooling
BSAF	Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station	RCS	Reactor Cooling System
BWR	Boiling Water Reactor	RHR	Reactor Heat Removal
CCS	Containment Cooling System	RPV	Reactor Pressure Vessel
CST	Condensate Storage Tank	PCV	Primary Containment Vessel
DC	Direct Current	PWR	Pressurized Water Reactor
DDFP	Diesel Driven Firer Pump	SA	Severe Accident
D/W	Dry Well	SBO	Station Black-Out
HPCI	High Pressure Core Injection	S/C	Suppression Chamber (torus)
IC	isolation Condenser	SRV	Safety Relief Valve
LWR	Light Water Reactor	TAF	Top of Active Fuel
MCCI	Molten Core Concrete Interaction	W/W	Wet Well

6. REFERENCES

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