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Overview and outcomes of the OECD/NEA benchmark study of the accident at the Fukushima Daiichi NPS (BSAF) Phase 2 – Results of severe accident analyses for Unit 1

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ABSTRACT

Phase 2 of the OECD/NEA Project "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF)" was established in mid-2015. The objectives have been similar to Phase 1 of the project but with an extended analysis period of 3 weeks from the occurrence of the earthquake, a major focus on fission product (FP) behaviour and releases to the environment and the comparison to various data including radiological data and results of backwards calculations of the source term. Nine organizations of six countries (Ciemat Spain; IAE, JAEA and NRA Japan; CEA, IRSN France; IBRAE Russia; KAERI Korea; NRC/DOE/SNL U.S.A.; VTT Finland) submitted results of their calculated severe accident scenarios for Unit 1 at the Fukushima Daiichi site using different severe accident codes (ASTEC, MAAP, MELCOR, SAMPSON, SOCRAT, THALES-KICHE). This paper describes the findings of the comparison of the participants results for Unit 1 against each other and against plant data, the evaluation of the accident progression and the final status inside the reactors. Special focus is on RPV status, melt release and FP behaviour and release. Unit specific aspects will be highlighted and points of consensus as well as remaining uncertainties and data needs will be summarised. The results for Units 2 and 3 are presented next in separated papers.

1. Introduction

The Great East Japan earthquake occurred on March 11th 2011 at 14:46 (Japan time zone); one minute later, the three reactors in operation (Units 1 through 3) automatically scrammed. A tsunami followed and resulted in a beyond design basis accident at the three units in TEPCO's Fukushima Daiichi NPS, with the worst conditions of safety systems availability occurred in Unit 1. The data recorded during the accidents (and later through visual inspections and sample analyses) do have an immense value for enabling further development and validation of the analytical tools that since the TMI-2 accident have been developed. However, many aspects of the accidents unfolding and the final state of Units remain uncertain since the database gathered is far from being as complete as necessary to make an accurate diagnostics. By using these analytical tools in a forensic way, key insights on both aspects might be gained and technical assistance to decommissioning might be given through the best characterization possible of corium and fission product distribution in each Unit. Likewise, a suitable sampling preceding and during the decommissioning phase might provide essential information to validate approximations and models

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embedded within severe accident codes.

Inspired by the potential of forensic studies and the expected support to the units decommissioning, a benchmark study of the accident progression for the Fukushima Daiichi NPS Units 1–3 accident has been conducted under the frame of OECD/NEA BSAF Project. Coordinated by IAE (Institute of Applied Energy), the project has developed in two phases. Phase 1 (Pellegrini et al., 2015; OECD/NEA, 2015) focused on accident progression analysis during the first 6 days of the accident, with emphasis on the thermal-hydraulics, core degradation and debris distribution in RPV (Reactor Pressure Vessel) and PCV (Primary Containment Vessel). Phase 2 extended substantially the scope of the project and, besides deepening in the aspects dealt with in Phase 1, addressed fission product release and distribution all over the plant (Reactor Building included - R/B) during the first three weeks of the accident, with the final intention to estimate Source Term to the environment. All the partners had full access to plant proprietary data and all the data recorded during and after the accedent (Tokyo Electric Power Company, 2014a, 2015a, 2015b, 2015c, 2015d, 2015e, 2017, 2014b, 2013, 2015f).

The present paper describes the main findings from Phase 2 of OECD/NEA BSAF project concerning Unit 1. The comparison of the participants' results against each other and against plant data, the evaluation of the accident progression and the final status inside the reactor will be shown. Special focus is on RPV status, melt release and FP behaviour and release. Unit specific aspects will be highlighted and points of consensus as well as remaining uncertainties and data needs will be summarised.

2. Analysis method

Table 1 shows the list of the institutes and the codes used to generate results for the calculation of the Unit 1 accident progression.

It is worth noting, though, that although codes are the main analysis tools and system behavior and operator actions have been as detailed as possible within the benchmark, scenario modelers have faced with the additional challenge of postulating uncertain boundary conditions and setting equipment failure criteria. As a consequence, some of the differences shown in Section 4 have a lot to do with these different postulates and not just with using different codes or opting for different choices in the input deck of the same code.

3. Major challenges of forensic analysis

Probably, the biggest challenge posed by Unit 1 accident progression is the very scarce information recorded during the first 12 h of the accident. Just two data points distant about 7 h from each other were measured in RPV pressure (P_{RPV}) and only four in PCV pressure (P_{PCV}), the first of which was recorded after more than 8 h of the accident onset. Few data were also recorded of in-reactor core water level; however, given that no water was injected during this period a monotonous decrease is assumed to have happened. There were two more periods of pressure data scarcity: the first just after venting, which lasted a bit less than one day; and then, later (after 2.5 days approximately since the Unit scram), for almost 4 days.

Final corium and debris location once the unit is in a cool steady state is also a challenge. Despite muon inspection confirmed that little (if any) highly dense material remains in the core (Tokyo Electric Power Company, 2015a), robot entries in the PCV have not provided detailed information yet about how corium might have distributed between pedestal and DryWell (D/W) floor. Nevertheless, some thick sediment layer deposits have been observed out of the pedestal, which thickness is notably larger just in front the pedestal slit (Tokyo Electric Power Company, 2015b, Tokyo Electric Power Company, 2015c; Tokyo Electric Power Company, 2015d, 2015e, 2017).

As for boundary conditions, like water injection and leaking paths, forensic analysis is also strongly challenged, since they might determine to a good extent accident evolution estimates. As for water injection, the information available is not conclusive. Even though several attempts were done along the first days to inject water and apparatus readings seemed to indicate a successful injection into the Unit is uncertain that, due to leakages and piping connections, any of that water got in the RPV. The first signals that might clearly indicate external water injection have been recorded at about 270 h (more than 10 days) after the scram. At that time, both P_{PCV} and T_{BH} (RPV bottom head temperature) showed a noticeable and consistent change with water injection.

As for leaking paths between large Unit 1 components (location, single or multiple, dimensions, evolution, etc.), like RPV to PCV and PCV to R/B, quite limited information has been gained by robotic access to PCV and R/B. The latest inspection of R/B has shown that some leakages between D/W and Torus Room might have happened (Tokyo Electric Power Company, 2014b, 2013); this seems to be consistent with the high radiation levels found in the second floor of R/B at that specific azimuthal location (Tokyo Electric Power Company, 2015f). However, it is unlikely that this is the only leaking path between PCV and R/B, and one cannot be certain either that this happened during the accident and not afterwards.

4. Thermal-hydraulics and core degradation analyses

The Unit 1 accident may be seen as a response of a BWR3 reactor technology to an unmitigated total loss of power (SBO) accident. With the exception of the employment of the Isolation Condensers (IC), which served as ultimate heat sink during the very early times of the accident sequence, during core degradation no operator action was successful.

The only two available RCS pressure measurements until 30 h indicate that a depressurization event occurred between 5 h and 12 h. The timing of the RCS leakage or RPV failure in the simulations varied depending on the hypothesis made by modelers (Table 2). Most leakages, though, were predicted to occur between 3 and 4 h, regardless the path chosen. The differences related to the leakage assumptions (code boundary conditions) during the pressure vessel degradation are visible in the RPV pressure evolution. The computed RPV pressure trends present coherent behavior with the two measured data values in Fig. 1a; however, given the difference in the RPV depressurization via postulated, each calculation has a different way to match the two pressure data points. It should be noted that whenever a leak is assumed in the modeling, the leak cross section becomes an unknown that the code user has to set, sometimes being even time dependent. As seen in Table 2, two modelling teams assume a main steam line failure (at 5 h and 6.1 h respectively) resulting in an immediate reactor depressurization following the failure. All other institutes assume the RCS leak occurs at some penetration(s) (SRV, SRM, TIP or PLR) presenting a gradual variation of the pressure. And some of them even do assume two leakage pathways. Any of the potential RCS failure modes might be mechanically feasible under the anticipated conditions by the codes

Table 1
Participants and codes employed for Unit 1 calculation.

	Organization	Country	Code
1	CEA	France	TOLBIAC (MCCI analysis)
2	CIEMAT	Spain	MELCOR 2.1–4803
3	IAE	Japan	SAMPSON-B 1.4 beta
4	IBRAE	Russia	SOCRAT/V3
5	IRSN	France	ASTEC V2.0 rev3 p1
6	JAEA	Japan	THALES
7	KAERI	South Korea	MELCOR 1.8.6
8	NRA(S/NRA/R)	Japan	MELCOR 2.1
9	NRC/DOE/SNL	U.S.A	MELCOR 2.1–5864
10	VTT	Finland	MELCOR 2.2–9607

Table 2
Event time of RCS leak or failure.

	CIEMAT	IAE	IBRAE	IRSN	JAEA	KAERI	NRA	SNL	VTT
SRM/TIP pipe [h]	1.0	–	3.6	–	2.47	–	4.1	–	–
SRV gasket [h]	4.61	4.1	10.9	3.68	–	–	4.2	–	–
PLR pump leak [h]	–	–	–	–	–	–	–	–	1.0
MSL failure [h]	–	–	–	–	–	5.0	–	6.1	–

used, although not a specific component mechanical analysis has been conducted. Unfortunately, data in RPV and PCV are so scarce in the period between 5 h and 12 h that screening out of some of the postulated scenarios would not be reliable.

D/W pressurization is the result of water, steam and hydrogen (generated from zirconium-steam reactions during the core degradation) from the RPV. As expected, different RPV depressurization patterns results also in different ways to match the PCV pressure data available (Fig. 1b). Those calculations that do not assume direct steam release into the D/W (KAERI and SNL), show a mild pressurization until a peak is displayed at the time MSL breaks and, from then on, a subsequent pressure increase as long as some water remains in the RCS (Reactor Coolant System). Assuming the leakage in the RCS as either a result of the temperature increase (SRV, SRM) or from equipment leakage (PLR pump) will generally produce a slight depressurization of the RPV and continuous containment pressurization from steam discharging into the D/W. The results of CIEMAT, assuming double leaking of water and steam, capture the pressure rise to 0.6 MPa as in the measurements. Anyway, Fig. 1b shows that regardless how close calculations are to capturing D/W pressure at 10 h, from around 4 h to 10 h the scenarios modeled lead to drastic pressure differences, as large as a factor of two at around 6 h.

It should be also highlighted that most codes predict consistently data in between 15 h and the PCV venting time (around 24 h). Such a steady state is interpreted in most cases as a natural PCV pressure "self-regulation" balancing the pressure increase resulting from MCCI and the PCV leaking through the flange of the upper head of the D/W into the reactor building.

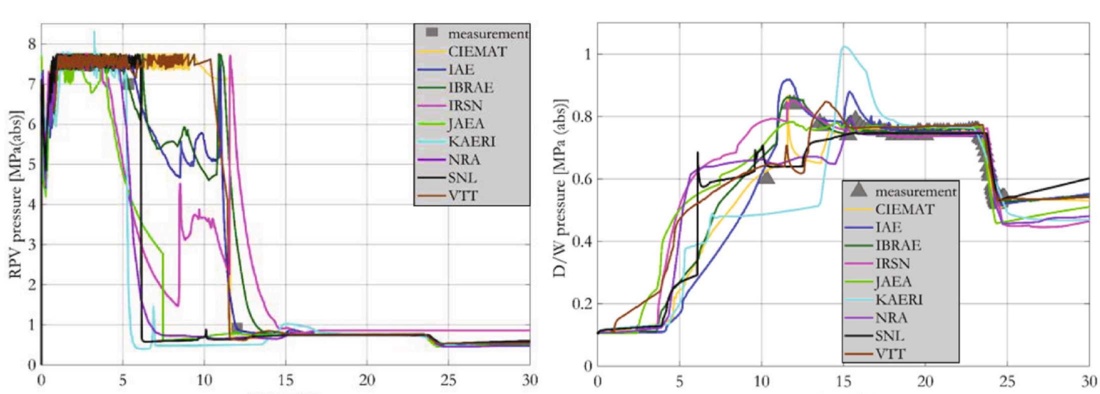
There is some variation in the prediction of the RPV failure time and mode between the calculations (Table 3). CIEMAT and VTT predict the lower head failure time, by melt through or penetration failure respectively, coincident with the timing of the largest measured PCV pressure at around 11.5 h. IRSN, IBRAE and IAE justify the PCV pressure increase with the assumption that the rise is generated by the slumping of debris in the lower head, while JAEA associates this pressure rise with MCCI progression.

Fig. 2 shows the H_2 generation partially responsible for PCV pressure increase during the first 10 h. The results look spread, although if the highest (IBRAE) and lowest (JAEA) calculations are considered outliers, then the scatter reduces roughly to a factor of 2.0. Anyway, note that most calculations predict close to or over 600 kg of H_2 produced.

Before S/C venting (23.73 h) all the calculations estimate RPV lower head failure, onset of MCCI and concrete erosion. The majority of institutes assumes instantaneous corium spreading within the pedestal or in the sump (2.5 m − 3 m) as a default in the code and evaluates the erosion from this point. Only two organizations calculated corium spreading out the pedestal into the D/W.

After the S/C vent closure (24.68 h), the gases released from MCCI drives pressure rise in PCV until at 50 h the trend reversed and started a decreasing period that lasted for more than 200 h until it sharply started growing again at 270 h from the reactor scram. All the calculations follow the measurements (Fig. 3) by assuming a containment leak of variable cross section from 50 h on; just JAEA does not make such a hypothesis and postulates venting from the suppression chamber. Among the institutes assuming a containment leaking, some of them (3) assume that the leak occurs at a penetration while all the others (5) assume that the leak is the result of MCCI erosion of the D/W liner. Calculations present different decreasing slopes depending on gas mass flow rate imbalance between MCCI and leaking flow rate. At around 270 h a few calculations do achieve the tendency change by imposing an effective water injection; however, as shown in Fig. 3b, they agree on an injection rate lower than 4.5 kg/s, but they differ in the specific injection profile. Failure of the PCV due to MCCI attack to the liner seems to be supported by the current observation of water leakages from PCV into the sand pit and the high radiation dose rates measured around the piping of the reactor building closed cooling water; nonetheless, there is no direct evidence of when this leak path was set or how large the initial size was.

At the time of RPV lower head failure most predictions calculated a total degraded core mass released in between 120 and 190 tons; in other words, the codes consistently predict a massive amount of material at very high temperatures is relocated in the pedestal. In addition, despite no in-situ evidences indicate so, most calculations showed that interaction of the molten material with concrete is still ongoing at 500 h (Fig. 4).



a. RPV b. PCV

Fig. 1. Pressure evolution in Unit 1 (0–20 h).

Table 3
Event time of RPV failure and failure mode.

	CIEMAT	IAE	IBRAE	IRSN	JAEA	KAERI	NRA	SNL	VTT
RPV failure [h]	11.58	14.97	15.10	17.16	7.45	13.50	13.6	12.5	11.42
Failure mode	Melt through	Penetration failure	Melt through	Creep	Melt through	Penetration failure	Melt through	User specified	Penetration failure

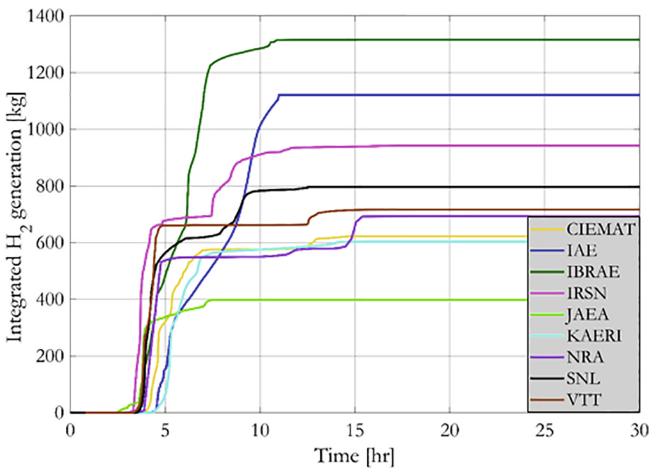


Fig. 2. In-vessel hydrogen generation.

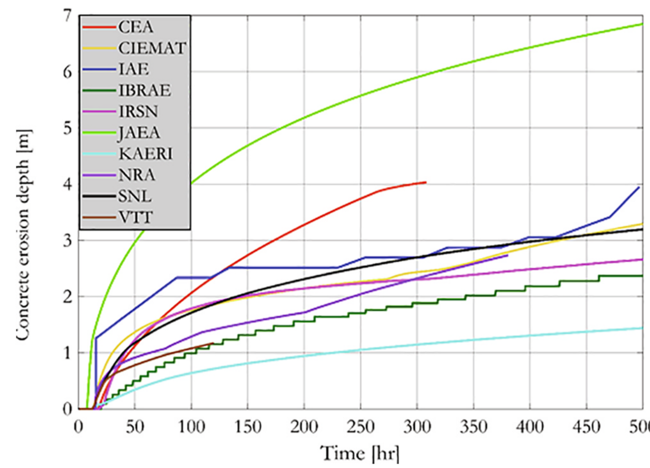


Fig. 4. Concrete erosion depth (origin at the bottom of the sump).

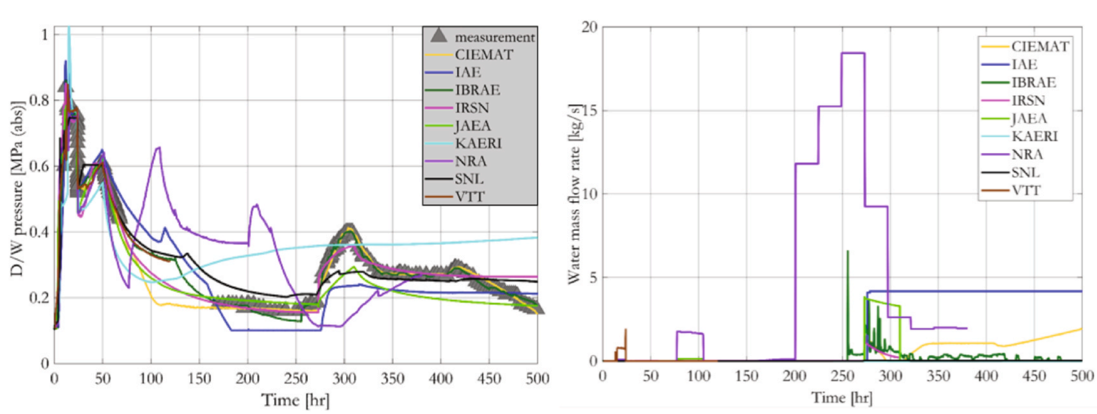
5. Fission product release and behaviour

Codes calculate most relevant fission products, either as individual RadioNuclides (RN) or grouped in classes according to their chemical similarity. However, given the safety significance of cesium (Cs) and iodine (I) isotopes (Cs^{137} and I^{131}, respectively) and the necessary limited extension of a paper, the analysis here is focused on these two elements. The initial inventory of every element was specified by the benchmark coordinator and in the case of cesium and iodine, they were 154.0 kg and 11.98 kg, respectively.

Fig. 5 shows the Cs (a) and I (b) releases from the fuel in terms of fraction from the initial inventory. Given their volatile nature, Cs and I releases are tightly related to the onset of fuel rod damages. Most calculations draw a fast release with or without subsequent steps according to core degradation progression up to getting an asymptotic value, which in most cases is over 80%.

The results of FP distribution in the containment are conditioned by the assumptions made concerning the leaks from RPV to D/W, although in most cases RNs enter PCV through the Safety Relief Valve (SRV) discharge in the suppression pool. Most Cs and I are in aerosol form when enter the PCV and once there they settle down from the atmosphere and deposit on D/W structures or get efficiently scrubbed when passing through the suppression pool. Typically, most analyses predict less than 10% airborne concentration, except in those cases that assumed a direct leakage from RPV gas phase into D/W. It is worth emphasizing that significant deposition was predicted on D/W structures and some remobilization was noted in some calculations either because of PCV depressurization or heating up of structures beneath deposits.

For most calculations, Cs and I are very efficiently scrubbed in the suppression pool during the early injection through the SRVs (Fig. 6). Given the carrier gas composition (i.e., steam rich) and velocity, those calculations in which most FPs were injected through quenchers reached higher values of the decontamination factor than those in which a good fraction of the FP got into the pool through the down comers (non-condensable rich). In the end, between 20% and 60%, roughly, of Cs and I are estimated to have been scrubbed in S/C water.



a. PCV pressure vs. time

b. Alternative water injection mass flow rate vs. time

Fig. 3. PCV evolution over 500 h period.

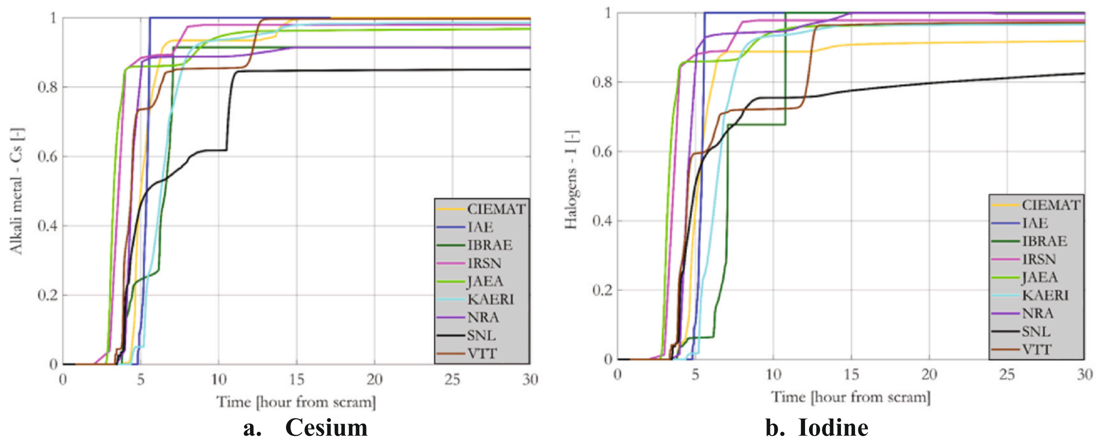


Fig. 5. Fraction of FP released from fuel.

During the accident, dose rate in the containment was measured by Containment Air Monitoring System (CAMS). These measurements were not directly comparable to masses resulting from severe accident codes and an approximate methodology was developed to account for contributions from the different RNs depending on their location (airborne, deposited on structures or solved in water). It is accepted that given the approximations made, the uncertainty associated to this conversion is at least a factor 2.0. In addition, some behaviors shown by measurements, like the signal drop to zero in D/W between 120 and 200 h, are hard to explain. Despite everything, comparison to calculations allows realizing that qualitatively predictions captured the same order of magnitude and trend as measurements', particularly in the long run (Fig. 7). Nonetheless, some calculations show strong oscillations in D/W and most of them displayed close to an order of magnitude underpredictions in the W/W.

Finally, the source term to environment may be compared to inverse analyses based on land and ocean measurements and carried out with atmospheric transport codes (hereafter denoted as GRS and WSPEEDI). For Unit 1 the comparison is relatively straightforward at the beginning of the accident because until around 35 h (time of Unit 3 postulated core meltdown) there cannot be other release than from this unit. The main releases from Fukushima Daiichi NPP accident in Unit 1 were detected by the monitoring posts at the time of PCV venting and during the subsequent hydrogen explosion. The first significant release was predicted around 12 h after the accident has started by GRS and about 14 h by WSPEEDI. Nevertheless WSPEEDI present at around 15 h a considerably larger release compared to GRS which will reach WSPEEDI values with a second release from around 17 h. Fig. 8 presents the comparison of the cumulative releases against GRS and WSPEEDI codes Cs and I. First, it should be noted that results of GRS and WSPEEDI codes tend to reach the same value at the time of venting and subsequent hydrogen explosion, although with about an order of magnitude difference in the case of I.

Almost all calculations present a first release at around 10 h, when the PCV head flange is predicted to start leaking, and FPs are released to the environment through the assumed leakage from the building. Nevertheless these releases are much lower than the value estimated by GRS and WSPEEDI. The integral release after the PCV venting at 24.7 h is predicted reasonably well in what concerns the time and magnitude within a range of about 2 orders of magnitude ($6{\cdot}10^{13}$ Bq – $5{\cdot}10^{15}$ Bq). Most calculations estimate that less than about 2% of Cs and 5% of iodine might have been released to the environment, although the best fit to inverse calculations is matched by those with Cs in between 0.5 and 1.0% and I around 1%.

Tables 4 and 5 compile the Cs and I distribution, respectively, all over the Unit. As already discussed, these radionuclides distribution is highly dependent on the scenario postulated by participants, but all of them got a significant fraction on RPV surfaces and suppression pool, in the case of Cs, and most of them estimated minor Cs releases to the environment. In the case of I, all participants predict a significant, sometimes substantial, amount of I in the suppression pool and higher releases to the environment than in the case of Cs.

6. Final remarks

An outstanding progress has been achieved in the second phase of BSAF with respect the first one. Even though there seem to be several plausible scenarios, in general terms the predictions scatter has been

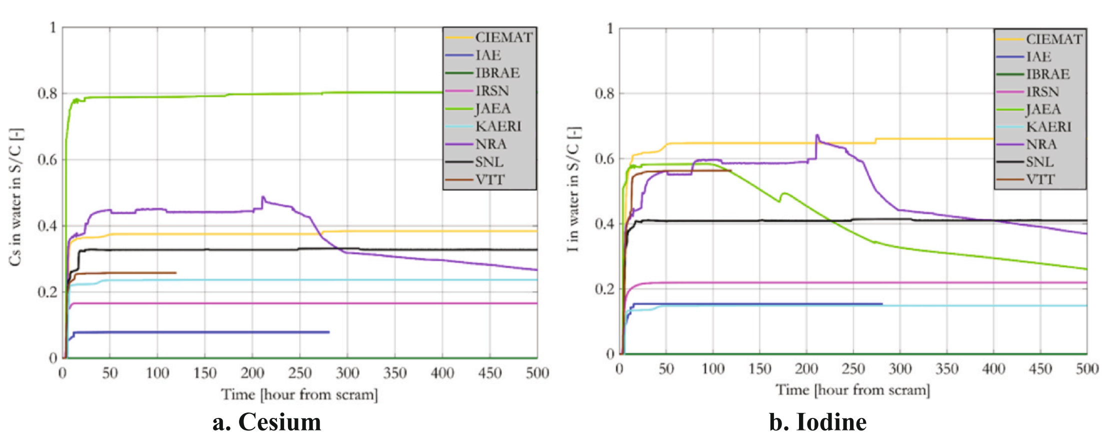


Fig. 6. Fraction of FP trapped in the suppression pool.

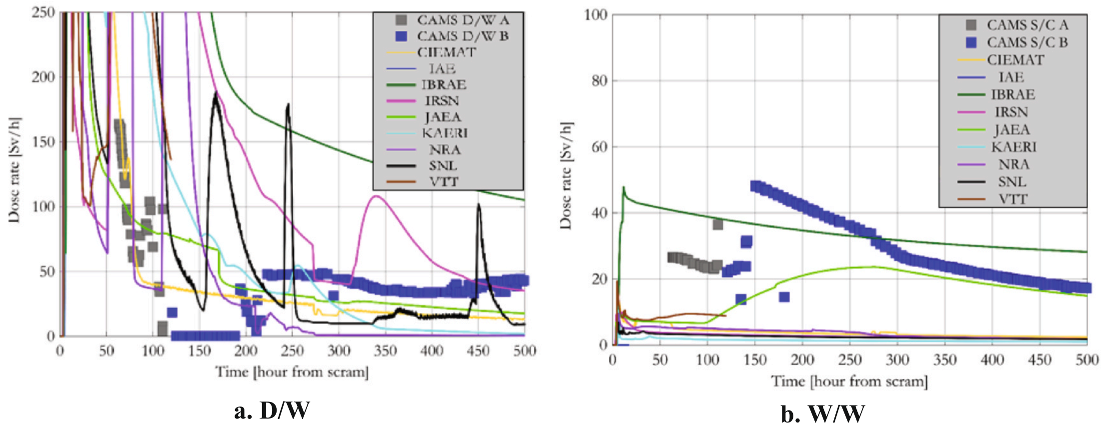


Fig. 7. Dose rates in PCV compared to CAMS measurement.

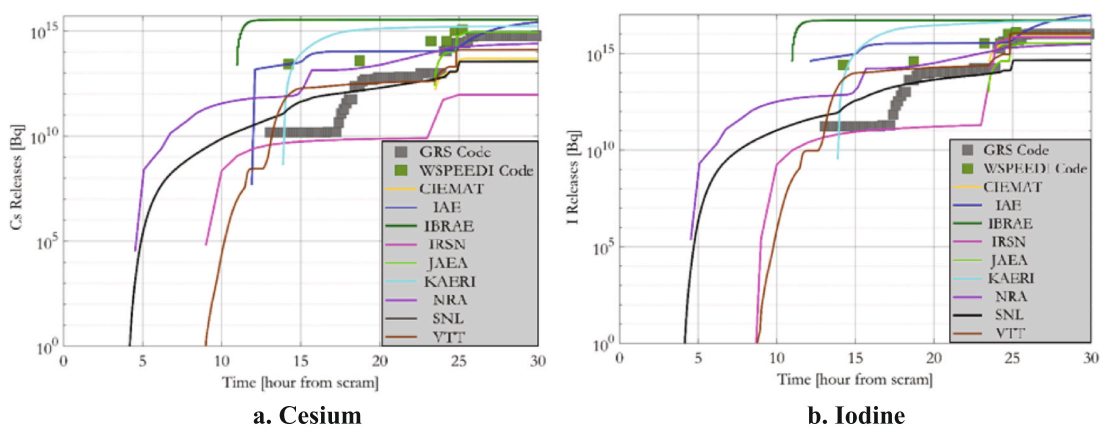


Fig. 8. FP releases from Unit 1 and comparison to WSPEEDI/GRS data.

Table 4
Distribution of Cesium in unit 1 at the end of the calculation (% of i.i.).

	VTT	NRA	CIEMAT	KAERI	IRSN	JAEA	SNL	IAE	IBRAE
Fuel debris	0.3	6.6	0.7	0.0	2.1	0.58	0.0	0.0	0.0
Reactor	68.2	35.3	54.3	0.0	25.0	10.6	11.4	11.4	10.1
Steam line	0.0	N/A	–	N/A	~0.0	–	0.0	0.0	N/A
D/W	0.01	12.6	5.2	27.2	55.4	6.32	77.1	70.1	67.5
S/C	27.5	25.1	33.2	64.1	17.1	80.92	9.2	9.2	20.8
R/B	3.1	19.6	4.2	2.6	0.001	0.63	0.3	0.3	0.0
Environment	0.6	0.8	2.4	6.1	0.0	0.95	2.1	9.1	1.6

Table 5
Distribution of Iodine in unit 1 at the end of the calculation (% of i.i.).

	VTT	NRA	CIEMAT	KAERI	IRSN	JAEA	SNL	IAE	IBRAE
Fuel debris	2.2	0.0	10.0	0.4	2.2	0.58	0.0	0.0	0.0
Reactor	0.4	0.6	0.0	0.0	0.0	17.61	2.6	2.6	18.6
Steam line	0.1	N/A	–	N/A	0.0	–	0.0	0.0	N/A
D/W	0.2	18.9	9.8	7.3	74.5	13.93	69.4	69.4	64.7
S/C	62.1	37.4	59.6	69.8	22.6	59.94	16.8	16.8	12.6
R/B	27.4	41.3	15.9	6.3	0.2	1.46	1.7	1.7	0.0
Environment	7.3	2.5	4.7	16.2	0.4	6.48	9.6	9.6	3.7

less and it is expected that whenever more data on corium location and composition become available the number of scenario candidates might reduce even further. In addition, there are aspects on which a reasonable agreement has been reached among modelers:

- Core degradation temperatures were reached around 5 h after scram, at high system pressure.
- Large amounts of H_2 were produced in vessel, most probably between 600 kg and 1000 kg, and leaked from PCV to R/B through the D/W head flange.
- Core degradation was massive and most of it (over 120 tons) relocated out of RPV at times in between 10 and 15 h.

- Failure of PCV upper head flange has been generally accepted by all modeling teams and other failure types might have happened too (most probably at 50 h).
- Gas production from the interaction of corium with concrete kept PCV at high pressure (around 0.7 MPa), despite the PCV flange leaking, until venting at around 24 h.
- Water injection in Unit 1 was probably not effective until more than 11 days after reactor scram.
- Releases of NG, Cs and I from the fuel were most probably over 80% of the initial inventory.
- There was high retention of fission products in the suppression pool, but the overall efficiency trapping was highly affected by the set leak paths bypassing the suppression chamber.
- Around 1% of Cs and 4–5% of I were released to the environment from Unit 1 until the H_2 explosion in the R/B. Note that these numbers are somewhat higher than what environmental tools seem to indicate.

However, some big uncertainties remain, particularly those related to:

- The type of leakage/failure of RPV (and size).
- Given the high parametrization of core degradation and relocation models and the lack of data concerning BWR technology core degradation, event timings during the in-vessel phase of the accident are spread. In other words, core degradation and relocation is one of the areas with higher uncertainty.
- The presence of corium out of the pedestal, which is an aspect highly dependent on modeling assumptions.
- The continuous interaction between corium and concrete, as predicted by most codes, is controversial with the observations made. Corium spreading and concrete attack termination is other area in which large uncertainties remain.
- The additional failure of D/W to the one at the top head flange has not been soundly identified.
- Remobilization of deposits was estimated in some calculations and it might have affected long term releases from the Unit 1. However, some remobilization mechanisms are not modeled in the codes used.
- Given the a-priori known large uncertainties existing on iodine chemistry, few calculations included this aspect in their analyses.

The impact of core degradation uncertainties highlighted above is not just related to modeling options to be activated/deactivated by modelers. Current research conducted under the auspices of OECD/NEA (TCOFF project) indicates that complex material interactions probably taking place in BWR cores, have not been accounted for so far in severe accident codes. By implementing these considerations key aspects for decision making, like RPV failure timing or in-PCV corium composition, could change substantially; it is worth noting that the latter might have an effect on decommissioning.

There are two aspects discussed in previous sections which better modeling would substantially enhance severe accident codes capabilities: core degradation especially in BWR (online with the previous discussion) and fission product remobilization by a variety of mechanisms, which effect is even strengthen by accident management measures like containment venting and water injection. Further modeling developments are expected in both regards in the coming years.

Finally, the understanding of the accident unfolding in Unit 1 has noticeably grown in BSAF Phase 2 and the forensic analyses conducted have even helped to assess the effect, if any, of the operator actions. In addition, through the benchmark some needs of the analytical tools presently being used have been identified. This investigation will be continued further in the frame of the OECD/NEA ARC-F project.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

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References

M. Pellegrini, et al., “Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF), Phase 1, Final Report, (2015).

OECD/NEA, “Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF Project) – Phase 1, Summary Report, March 2015.” NEA/CSNI/R(2015) 18 (2016).

Tokyo Electric Power Company, Inc., Evaluation of the situation of cores and containment vessels of Fukushima Daiichi Nuclear Power Station Units1 to 3 and examination into unsolved issues in the accident progression, Progress Report No. 2, Attachment 1-5, “Evaluation into the amounts of water injected to Unit 1 by fire engines” August 6, 2014.

Tokyo Electric Power Company, Inc., “Reactor imaging technology for fuel debris detection by cosmic ray muon”, March 19, 2015.

Tokyo Electric Power Company, Inc. Development of a technology to investigate inside the Reactor Primary Containment Vessel (PCV), Site test “Investigation B1” on grating around the pedestal inside Unit 1 PCV, April 10th 2015. http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_150410_01-e.pdf.

Tokyo Electric Power Company, Inc. Development of a technology to investigate inside the Reactor Primary Containment Vessel (PCV), Site test “Investigation B1” on grating around the pedestal inside Unit 1 PCV, April 15th 2015. http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_150416_01-e.pdf.

Tokyo Electric Power Company, Inc., Development of a technology to investigate inside the Reactor Primary Containment Vessel (PCV), Site test “Investigation B1” on grating around the pedestal inside Unit 1 PCV, April 16th 2015. http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_150417_01-e.pdf.

Tokyo Electric Power Company, Inc., Development of a technology to investigate inside the Reactor Primary Containment Vessel (PCV), Results of follow up check after the site test, April 18th 19th. http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2015/images/handouts_150420_01-e.pdf.

Tokyo Electric Power Company, Inc., Unit 1 Primary Containment Vessel (PCV) Investigation at Fukushima Daiichi Nuclear Power Station, March 9th 2017. http://www.tepco.co.jp/en/nu/fukushima-np/handouts/2017/images/handouts_170309_01-e.pdf.

Tokyo Electric Power Company, Inc., “Results of Investigation on Upper Parts of S/C (Suppression Chamber) of Unit 1 at Fukushima Daiichi NPS”, May 27th, 2014.

Tokyo Electric Power Company, Inc., “Results of Investigation around Lower Parts of Unit 1 Vent Pipes at Fukushima Daiichi NPS (Second Day), November 14th, 2013.

Tokyo Electric Power Company, Inc., Evaluation of the situation of cores and containment vessels of Fukushima Daiichi Nuclear Power Station Units1 to 3 and examination into unsolved issues in the accident progression, Progress Report No. 4, Attachment 1-9, “Estimation of causes of high contamination of RCW piping at Unit-1”, December17th, 2015.