The OECD-NEA Benchmark Study of the accident at the Fukushima Daiichi Nuclear Power Station (BSAF) Project – Phase 2

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ABSTRACT

The OECD-NEA BSAF project is aimed to evaluate and analyse the likely end-state of the reactor core after the accidents at the Fukushima Daiichi nuclear power stations. The project will help the TEPCO to plan the removal of components from the reactor containment and the final decontamination. The Phase 1 of the project concentrated in the accident progression for 6 days after SCRAM and it has concluded in November 2014.

A follow-on project, designated as Phase 2, is ongoing. For the Phase 2, the scope of the analysis is extended to include the **hydrogen generation** and potential for combustion as well as the **source term analysis** and comparison with the measured activities and dose rates at relevant locations at the plant and in the plant vicinity. In addition, the duration of the analysed sequence will be extended to 20 days from the accident initiation, a task which may prove very challenging to the severe accident analysis tools.

The signatories countries from Phase 1 will continue in Phase 2: France, Germany, Japan, Korea, Russia, Spain, Switzerland and United States. Additionally 3 new signatories have joined the project: China, Canada and Finland. The operating agent of the project is The Japan Atomic Energy Agency (JAEA). The project started 2015 and is planned to end in 2018.

PSI has successfully participated in the BSAF project Phase 1, represented Switzerland in the project meetings and has contributed to the final and summary BSAF Phase 1 report. During Phase 2, MELCOR 2.1 is being used as the main tool for the system level simulation of the sequence. However, the use in the future of other tools which provide a more detailed treatment of hydrogen distribution and fission product behaviour is not excluded. During the present year, the main possible failures of RPV and PCV were identified; therefore the MELCOR 2.1 input has been modified to include such failures. Additionally a proper representation of the fission product release has been included. After the changes a similar sequence to the one from BSAF Phase 1 was obtained. Plausible failures of RPV and PCV were preliminary evaluated.

Project goals

The Project OECD-NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) [1] Phase 2 is intended to extend the scope of the analysis performed in Phase 1 to include the **hydrogen generation** and potential for combustion as well as the **source term analysis** and comparison with the measured activities and dose rates at relevant locations at the plant and in the plant vicinity. The following main objectives will be addressed:

- To extend the analysis time span from the *6 days* in Phase 1 to until the end of March, 2011, or to approximately *20 days* from accident initiation (the earthquake).
- To extend the scope of the accident analyses of Fukushima Daiichi units 1–3 to include the amount of *hydrogen generated*.
- To extend the analysis to include the *fission product release* from the core, the retention in and transport through the units (reactor system, containment, reactor auxiliary buildings) and release to the environment, and hence provide guidance on the level of contamination likely to be encountered during ongoing operations at and in the vicinity of the station. For this, the necessary models / nodalization have to be developed;
- To extend the analysis to consider the buildings adjacent to the containments, e.g., the reactor building, to evaluate the *hydrogen effects and the source term transport*. For this, it is necessary to improve / develop nodalization for the plant to include the adjacent buildings;
- To improve the methods and models of computer codes in use by each participating country and organisation, to reduce uncertainties in Severe Accidents (SA) analysis and validate SA codes using actual data available from Fukushima Daiichi NPS units 1–3.

The analysis at Paul Scherrer Institute (PSI) will concentrate on Unit 3, as during BSAF Phase 1. During the present period, the main tool for the system level simulation of the sequence used was MELCOR 2.1. Use of other tools which provide a more detailed treatment of hydrogen distribution and fission product behaviour is not excluded but were not used in the present period.

Work carried out and results obtained

1. Overall performed work

PSI has participated in the BSAF project Phase 1, represented Switzerland in the project meetings and has contributed to the final and summary BSAF phase 1 report which was finalized beginning of 2015. The main outcome of BSAF Phase 1 analysis performed in PSI is summarized in [2].
The following activities were performed in 2015 at PSI:

- The BSAF Phase 1 final report was reviewed and finalised between end of 2014/beginning 2015.
- The results of the BSAF Phase 1 project were analysed to determine the *most likely accident progression*, possible Reactor Pressure Vessel (RPV) and Pressure Containment Vessel (PCV) failures and the most relevant phenomena to be included in the BSAF Phase 2 analysis.
- All the relevant technical data were and will be continuously reviewed as they become available on the project internet site. This includes any new plant design data, boundary conditions, accident data for the extended time frame, and activity measurements at and in the vicinity of Fukushima Daiichi.
- The existing model of the Unit 3 of the Fukushima Daiichi nuclear power plant for MELCOR 2.1 was modified to *include the source term* and all *the likely release pathways*.
- The *impacts of changes in the boundary conditions* on the accident progression were evaluated, for example, vessel breach and containment leakage.

2. Plausible RPV and PCV failures

One of the main tasks during the present period was to evaluate the available information concerning possible RPV and PCV failures and its influence in the fission products release. The transport path of fission products from the reactor pressure vessel to the primary containment and from the primary containment to the adjacent buildings or to the environment determines to a large extent the release of radioactivity to the environment.
The primary containments in the Fukushima Daiichi power plant had a possibility for depressurization through the suppression pool leading to a possibility of retaining a fraction of the activity in the water in the suppression pool. It is generally accepted that this lead to a considerable decrease of the release of the activity during the accidents.

One of the major open questions concerns the extent to which the radioactive compounds were transported through the suppression pools in the three reactors. The release path of fission products may be through a water pool or mainly dry/steam atmosphere. These two release paths may result in very different concentrations of activity being released to the reactor building and/or to the environment. In the present study, only the possible release pathways of Fukushima Unit 3 have been considered. After reviewing the different RPV and PCV failure assumptions for U3 from [2–9], two main groups have been identified. The possible failures are represented in figure 1:

a) A leakage from the primary containment vessel through, e.g., the upper head, to the reactor building (in red).
b) Normal venting through the stack or a leakage of the vent line from the suppression pool to the stack releasing gas to the reactor building (in yellow).

For group a) the release would follow the path direct to the reactor building in the case Drywell (DW) head flange failure occurs. The retention will depend on whether the RPV pressure boundaries have failed or not (e.g. Main Steam Line (MSL) failure, Safety Release Valves (SRV) gaskets failure, Lower Head Failure). If the release path is mainly dry, or a steam atmosphere (no water pools), the fission products will be depositing on the walls and floors of the transport path by mainly mechanisms governed by aerosol physics (settling, impaction, diffusion, turbulent deposition, condensation, etc.) or chemistry (reactions of the gas phase compounds, absorption, etc.).

For the second group b), even in the case with releases from the RPV to the DW (e.g. MSL failure, Lower head Failure, SRV gaskets) the gases would firstly follow the path through the water in the suppression pool. The fission product would be released to the environment via the venting line, either direct to the stack or through the building in the case the venting line failed. The retention of both aerosols and gas phase iodine increase significantly as compared to the dry transport path.

The successful venting through the stack or the bypass through the building may not make a significant difference in the retention of the aerosols. It is also worth noting that in Fukushima, the suppression pool was partly under saturated conditions during the gas injection, and this affects the fission product retention. Steam condensation in the water pools is known to increase aerosol retention. In addition, boiling together with high gas injection rates may generate droplets on the pool surface, and some activity may be transported away from the pool in these droplets.

3. Calculations with MELCOR

A. Modification to the input

Some modifications were made to the MELCOR input from BSAF Phase in order to better represent the geometry of Unit 3. The modifications included:

a) Correction of the SRV's flow area
b) Opening of six valves instead of 2 during depressurisation, in order to better capture the pressure drop during depressurisation.
c) Modification of initial radionuclides inventories to discard the use of CsMo and replace it by only CsOH.
d) The assumption that the SRV's are operating at a lower set point after ca. 36 h, in order to reproduce the observed pressure data.

In addition, the input was modified to add the possibility to study different release paths depending on the failures from RPV and PCV and release as shown in figure 2.

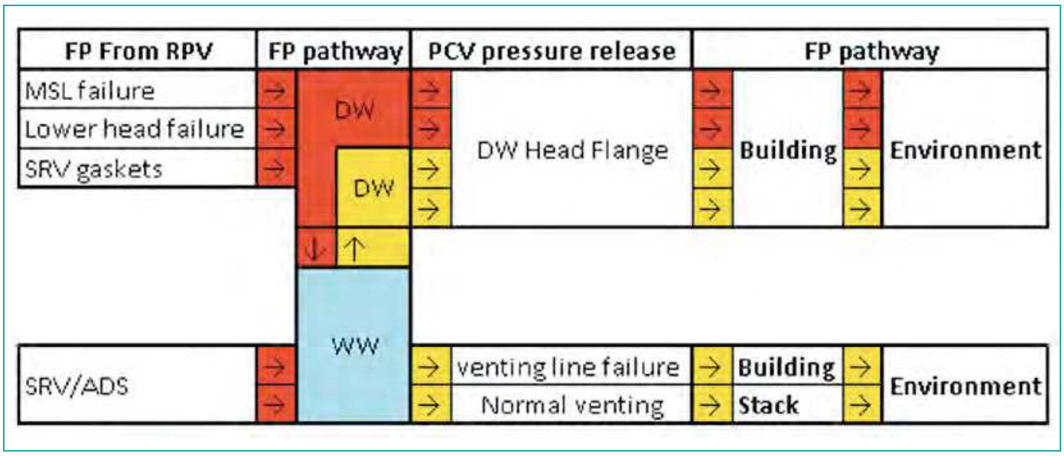
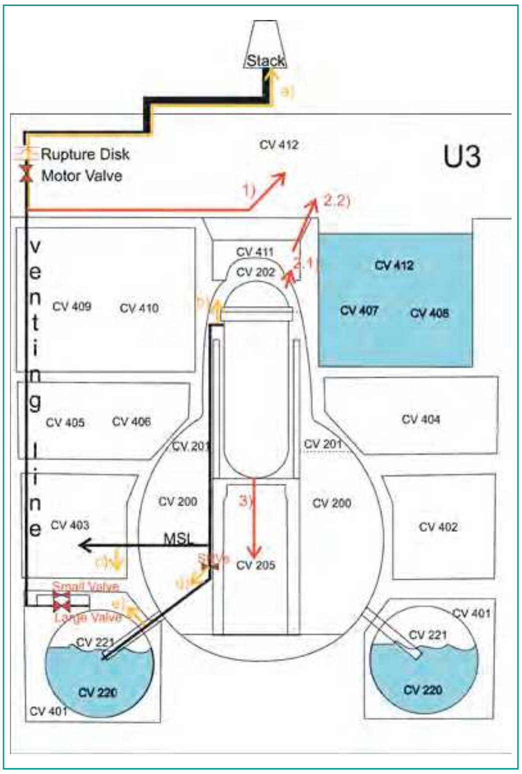


Figure 1:
Plausible venting paths

Figure 2): *Containment Nodalisation*



The existent FP release paths from BSAF Phase 1 included (see figure 2 in red):

1) Venting bypass
2) DW head flange failure
3) Lower head penetration failure

The following release paths were added (see figure 2 in orange):

a) PCV venting through the stack
b) Main steam line (MSL) leakage
c) Leakage from the DW to the building through the MSL penetration
d) SRV's leakage to the Drywell
e) Failure from vacuum breakers to the building

It was intended to obtain an input deck that can reproduce the main results from BSAF Phase 1, bearing in mind that MELCOR is extremely sensitive to small changes in the input such as different time step, changes in the nodalization or changes in the modelling. Therefore, it was expected that the included changes in the input will have an impact in the calculation results. Therefore it was necessary to slightly modify some of the boundary conditions in order to obtain similar results. The modifications included:

a) Adjustment of the water injection magnitudes during Reactor Core Isolation (RCIC) sytem operation.
b) Adjustment of the water injection depletion time during High Pressure Coolant Injection (HPCI) system operation.
c) Adjustments on the Alternative Water Injection (AWI) and venting areas.

The added release paths were not used at the same time in the input but it was important to have a sequence where they could be modified on restart. In this way the uncertainties due to the difference in the calculation (i.e calculated time steps or cycles) may be reduced. The calculations that were performed during the present period as well as their main release paths/failure assumption are shown in table 1.

B. Reproduction of BSAF-I calculation results

The ability to reproduce the main signatures is crucial to have meaningful calculations for the analysis of the FP releases. Figure 3 and 4 show the pressure signature for the RPV and the PCV respectively. The BSAF Phase 2 (BSAF-II-eBE, in green) calculation results, as expected, didn't give exactly the same results as those for BSAF Phase 1 (BSAF-I-BE, in red). Nevertheless, a calculation which reproduce very similar pressure signatures and that support the main conclusions of the BSAF-I was obtained. During the period between 0–42 h after SCRAM, the time when the RCIC and HPCI are operating, the main differences are during repressurisation and depressurisation due to the modifications in the input previously described.

Table 1: *Calculation performed during the present period*

	Vent to Building	Vent to Stack	MSL leakage	DW upper head failure	Penetration Failure	MCCI
BSAF-I BE	✔			✔	✔	
BSAF-II eBE	✔			✔	✔	
BSAF-II vent to stack		✔		✔	✔	
BSAF-II_MSL fail		✔	✔	✔	✔	
BSAF-II_MSL Fail_Ex-vessel		✔	✔	✔	✔	✔

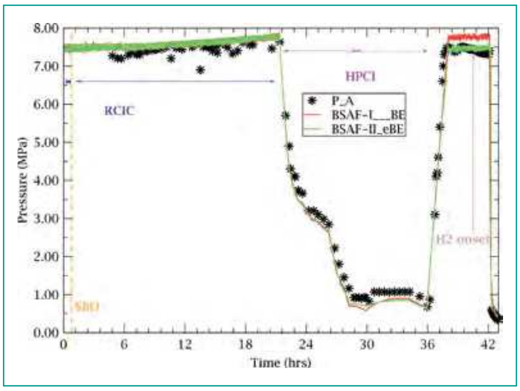


Figure 3:
RPV Pressure

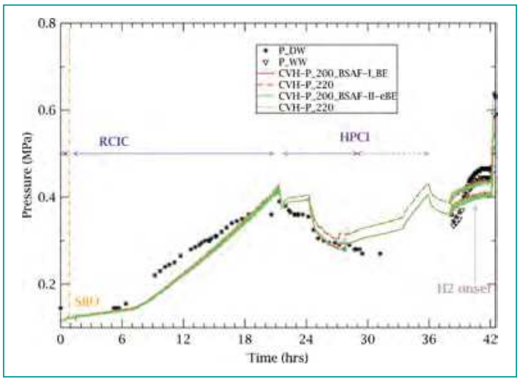


Figure 4:
PCV Pressure

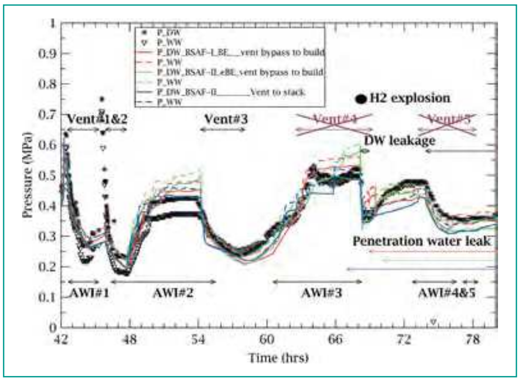


Figure 5:
PCV Pressure

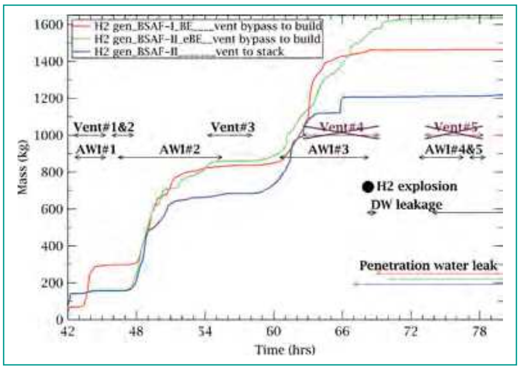


Figure 6:
H_2 Generation

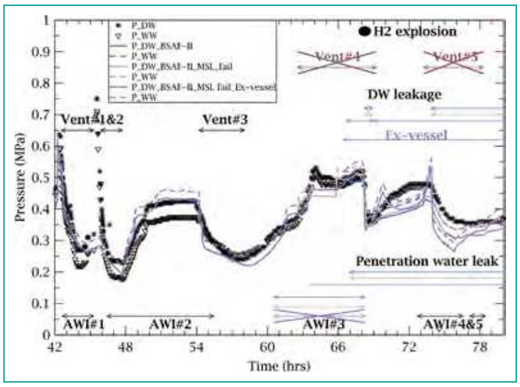
Figures 5 and 6 show the Primary Containment Vessel (PCV) pressure and the hydrogen generation for the period between 42–80 h. A calculation, where the PCV was directly vented to the stack (in blue), was performed. The times when the alternative water injection (AWI), and PCV venting (Vent) are taking place are indicated with black arrows. The times when Drywell (DW) leakage and H_2 explosion take place are as well indicated. The calculated water leak by penetration failure is indicated in the color of the specific calculation. The calculations predict similar pressures trends in the PCV. The hydrogen generation is slightly different, as expected due to sensitivity of the calculation to minor modifications, but in overall similar trends are obtained.

C. Preliminary study of fission product release

A preliminary study to obtain plausible release paths was performed. Different assumptions of RPV or PCV failures were made. Figure 7 and 8 shows the pressure in the PCV and the hydrogen generation respectively. In all the presented cases the venting was made through the stack. The base calculation (in blue) has no further failures and those described in the previous section, the relocated debris formed during the degradation process remain in the lower head of the RPV. Additionally, the assumption of Main Steam Line (MSL) leakage starting at ca. 67.5 h (calculations in brown and purple) was made. In order to obtain a calculation were the debris are expelled from the RPV to the pedestal a calculation was made (in purple) with the assumption that no AWI is reaching the RPV between ca. 60–68 h. The calculations predicted similar pressure trends, meaning that any of the scenarios presented in this report are plausible. In order to be able to predict the pressure in the PCV for the ex-vessel case (in purple), it was necessary to assume as well DW leakage. However the leakage needed to start earlier and should be of a higher magnitude than the case where the debris remained in the lower head of the RPV (in brown).

Figure 9 shows the total hydrogen releases to the stack and to the upper part of the building. All proposed cases have the assumption of DW leakage and they would give comparable amounts of hydrogen being released to the upper part of the building, either generated in-vessel (debris staying in the lower head) or ex-vessel (debris going out of the RPV and producing hydrogen by MCCI).

Figure 10 shows the temperature in the upper part of the drywell (DW). The calculations where MSL failure was assumed (brown and purple) calculated a higher temperature, in the time before the hydrogen explosion, than the one without MSL failure (in blue) The assumed MSL leakage is in the order of 0.5 kg/s. this leakage would correspond to a small hole in the MSL which may be due to different causes (e.g. a broken seal, a small hole in the MSL, etc.). The increase of temperature in the upper part of the containment may have contributed to further weaken the bolts of the DW upper



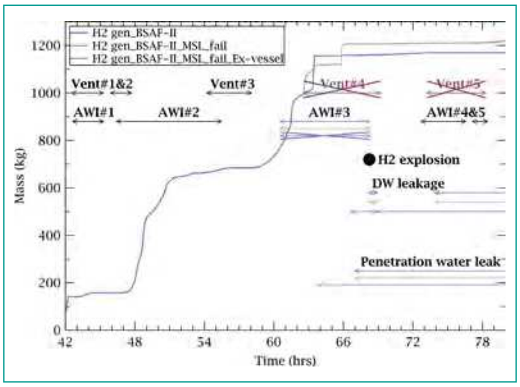
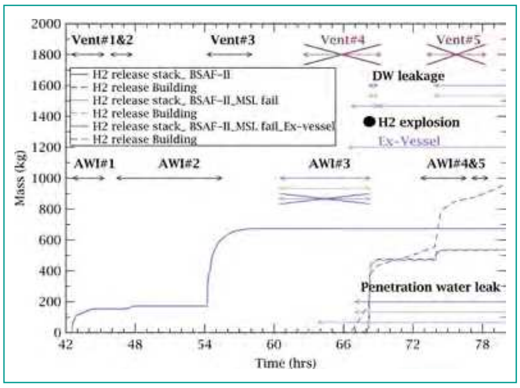


Figure 7 (left): *PCV Pressure*

Figure 8 (right): *H_2 Generation*



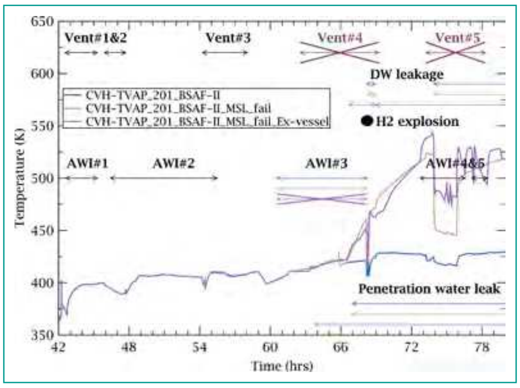


Figure 9 (left): *H_2 releases*

Figure 10 (right): *Temperatures*

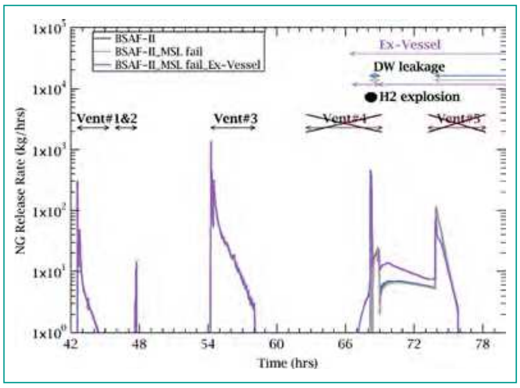


Figure 11: *NG rel. to atmosphere*

head flange. These results would reinforce the assumption of DW leakage before the H_2 explosion made during the BSAF phase 1.

Figure 11 shows the calculated NG release rates (kg/h) to the atmosphere. The calculated releases follow either a venting action or leakage from the DW. A detailed fission product release analysis and possible retention in the PCV and building will be performed in the future. The calculations presented in the present period will serve as a departure point. The future study will serve to quantify the amount of aerosols which may have been released with the proposed release paths.

The comparison of future calculations with publicly available measurements and future measurements obtained from the project would give strong indications which were the most likely release path(s) that took place during the accident.

National Cooperation

None.

International Cooperation

The project is coordinated by the OECD Nuclear Energy Agency (NEA). The Operating Agent (OA) is Japan Atomic Energy Agency (JAEA) who is technically supported by the Japan Institute of Applied Energy (IAE). The participants from Japan, Canada, China, Finland, France, Germany, Korea, Russia, Spain, USA, and Switzerland, each cooperate formally with NEA and the OA. There is informal cooperation between the participants.

Assessment 2015 and Perspectives for 2016

Progress is going as planned as the first modification for the input was made and a similar sequence was obtained after the implementation of the several changes mentioned in section 3A.

It is expected to have some delays in the project as the new confidential data and the new boundary conditions have not yet been released.

Nevertheless, in PSI we continue to progress in our individual analysis. The following tasks are expected to take place in the future:

1. Modification of the model and / or input description as appropriate based on the new data that become available. Expert judgement and consultations with other project participants will be used to decide on the extent of the input model modifications.
2. Perform an additional calculation where the Cs will be represented with CsMo (standard MELCOR 2.1 way) instead of CsOH (standard MELCOR 1.8.6 way). The comparison may give an indication which is the best way to represent the Cs releases for Fukushima U3.
3. Evaluation of the uncertainties in the data. First, simulations with MELCOR for 6 days accident duration as well as comparison of the calculated activity/dose rate with different RPV and PCV failure assumptions with the measured plant data and activity/dose rate data during the accident will be performed.
4. The calculation time migth be extended to 20 days after SCRAM, as defined in the BSAF Phase 2 work plan. This is the time after there is thought to have been little further release from the core.
5. Examination of the impact of changes in the boundary conditions on the accident progression. For example, vessel breach, main steam line leakage and containment leakage.

Publications

- *Fernandez-Moguel, L. and Birchley, J.* Analysis of the accident in the Fukushima Daiichi nuclear power station Unit 3 with MELCOR_2.1 Annals of Nuclear Energy. Volume 83, 2015, Pages 193–215
- *L. Fernandez-Moguel, A. Rydl, B. Jaeckel*: Analysis OF Fukushima Daiichi NPP Unit 3 WITH MELCOR_2.1. NURETH 16: Proceedings of the 16th International Topical Meeting on Nuclear Reactor Thermalhydraulics, 6401 (2015).

References

[1] *OECD/NEA/CSNI* 2014. Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station. https://www.oecd-nea.org/jointproj/bsaf.html

[2] *Fernandez-Moguel, L. and Birchley, J.* Analysis of the accident in the Fukushima Daiichi nuclear power station Unit 3 with MELCOR_2.1. Annals of Nuclear Energy. 2015(83): 193–215

[3] Bo*nneville H., Luciani A.* Simulation of the core degradation phase of the Fukushima accidents usig the ASTEC code. Nuclear Engineering and Design 272(2014) 261–272.

[4] EPRI, Fukushima Technical Evaluation: Phase 1—MAAP5 Analysis. Palo Alto, CA: 2013. 1025750.

[5] *Cardoni, J., Gauntt, R., Kalinich, D. and Phillips, J.* 2014. MELCOR simulations of the severe accident at Fukushima Daiichi Unit 3. Nuclear Technology Vol. 186.

[6] *Gauntt, R., Kalinich, D., Cardoni, J., Phillips, J., Goldmann, A., Pickering, S., Francis, M., Robb, K. Ott, L., Wang, D., Smith, C., St. Germain, S., Schwieder, D. and Phelan, Ch.* 2012. FUKUSHIMA DAIICHI: ANS Committee Report, Fukushima Daiichi Accident Study (Status as of April 2012). SANDIA REPORT SAND2012-6173. Sandia National Laboratories, Albuquerque, New Mexico 87185 and Livermore, California 94550

[7] R*obb, K.R., Francis, M.W. and Ott, L.J.* 2014. Insight from Fukushima Daiichi Unit 3 investigations using MELCOR. Nuclear Technology Vol. 186.

[8] *Sevón, T.* 2015. A MELCOR model of Fukushima Daiichi Unit 3 accident. Nuclear Engineering and Design. 284 80–90.

[9] *Pellegrini, M., Suzuki, H., Mizouchi, H. and Naitoh M.* 2014. Early phase accident progression analysis of Fukushima Daiichi unit 3 by the SAMPSON Code. Nuclear Technology vol. 186.