# FIRST RESULTS OF THE SIMULATIONS OF FUKUSHIMA-DAIICHI UNIT 3 ACCIDENT FOR AN ASSESSMENT OF THE APPLICABILITY AND THE CAPABILITY OF THE CODE ATHLET-CD

**Christoph Bratfisch, Mathias Hoffmann and Marco K. Koch**
Ruhr-University Bochum (RUB)
Chair of Energy Systems and Energy Economics (LEE)
Reactor Simulation and Safety Group
Universitätsstraße 150
44801 Bochum, Germany
christoph.bratfisch@lee.rub.de; hoffmann@lee.rub.de; koch@lee.rub.de

## ABSTRACT

In order to understand the course of severe accidents and to improve the appropriate accident management measures for Light Water Reactors numerical analyses and simulations are necessary and useful tools.
To assess the applicability and the capability of ATHLET-CD (Analysis of Thermal-hydraulics of Leaks and Transients - Core Degradation) as a severe accident code, events of the Fukushima-Daiichi accident, focusing on Unit 3, a Boiling Water Reactor by General Electric (GE BWR4), are simulated. For the simulations the latest code version 3.0A of ATHLET-CD is used. Regarding the first simulation results, an input deck based on a generic boiling water reactor (SWR-69) by Kraftwerk Union AG (KWU) concerning the cooling circuits only was used as a starting point and later on modified.
For an improved simulation of the events, the decay heat -given by TEPCO- was adjusted. Furthermore, mass flow rates of the High-Pressure Coolant Injection (HPCI) and the Reactor Core Isolation Cooling (RCIC) systems were adapted. To enhance the simulated course of the pressure in the RPV and the measured water level, the operating time of the RCIC was modified. Concerning the simulation, the alternative water injection given by a fire pump during the accident, was taken into account.
Later investigations had shown that the HPCI injection into the reactor pressure vessel stopped earlier than expected. Sensitivity analyses were performed to analyze the impact of the shortened HPCI injection regarding the thermohydraulic behaviour as well as the core degradation phenomena.

**KEYWORDS**
ATHLET-CD, Fukushima-Daiichi, severe accidents, BWR

## 1. INTRODUCTION

On March 11, 2011, an earthquake, measuring 9.0 on the moment magnitude scale, occurred in Japan. This earthquake led to a loss of the off-site electrical power source to the site. Initiated by the earthquake, a tsunami hit the coast 50 min later. The tsunami flooded the coastal areas and hit the nuclear power plant Fukushima-Daiichi. This event led to a Station Blackout (SBO) in four of the six units. As a consequence a failure of the emergency cooling system occurred in three units of the power plant, resulting in meltdown accidents in these reactors during the course of the severe accident.
In the frame of the research project SUBA as a part of the BMBF sponsored collaborative project WASA-BOSS the severe accident in the nuclear power plant Fukushima-Daiichi is considered.

For further application and analyses of the program ATHLET-CD (Analysis of Thermal-hydraulics of Leaks and Transients – Core Degradation), the course of the accident in Unit 3 of Fukushima-Daiichi is simulated.
In the following an overview of the sequences of the course of the accident as well as a description of the relevant emergency cooling systems will be given. Furthermore the modification of an input deck, based on a generic boiling water reactor (SWR-69), to Unit 3 is shown. To assess the applicability and the capability of the severe accident code ATHLET-CD the input deck has been adapted to the boundary conditions describing the accident. The in vessel simulation results are compared to the measured data of Unit 3 as well as to simulation runs given in literature performed with the codes MELCOR and MAAP5 by Sandia National Laboratories, Electric Power Research Institute and Japan Nuclear Energy Safety Organization. The focus is on the thermohydraulic behaviour since measured data describing the core behaviour (e.g. hydrogen release, molten material, etc.) are not available.
Further the impact of the HPCI injection regarding the core degradation is analyzed. Therefore two simulation runs are performed with ATHLET-CD. The two runs differentiate in the shut down time of the HPCI referring to assumptions made by TEPCO. In order to investigate the core degradation as a result of the different HPCI injection time the simulated hydrogen mass as well as the molten mass are compared.

## 2. THE ACCIDENT IN FUKUSHIMA-DAIICHI UNIT 3

The initiated event, a tsunami caused by an earthquake, led to the core meltdown accident in the considered Unit 3 of the nuclear power plant Fukushima-Daiichi, occurred on March 11 at 14:46 UTC+09:00. At this time Unit 3 was running at full capacity. In consequence of the intense quake, enormous damages occurred in Japan. These affected the electricity grid, too, leading to a loss of off-site power to the Fukushima-Daiichi power plant. With the loss of the power supply, the reactor emergency shutdown SCRAM was triggered immediately. Simultaneously, the emergency diesels started to supply power to the activated safety systems. In order to react to a dropping reactor water level, the Reactor Core Isolation Cooling System (RCIC) was started manually.
This emergency cooling system consists of a steam driven turbine linked to a pump. The pump injects water into the reactor pressure vessel (RPV) to maintain a specified water level. In addition the reactor pressure was controlled by the Safety and Relief Valves (SRV).
The RCIC stopped automatically due to a high water level in the RPV, 20 minutes after it started for the first time. At 15:27 the first tsunami hit the site followed by a second wave approximately eight minutes later. The tsunamis caused a flooding of the buildings leading to a loss of the emergency diesel generators which resulted in a Station Blackout (SBO) with no AC power left available. The two emergency cooling systems RCIC and HPCI (High Pressure Coolant Injection) remained operable due to the fact that AC power is not needed for these systems. Due to continued steam generation, caused by the decay heat, the water level dropped again. As a consequence, the RCIC was restarted 25 minutes after SBO and kept operating for the next twenty hours before an unsuspected shutdown occurred.
The shutdown of the RCIC was caused by an increasing turbine exhaust pressure in the suppression pool due to the failure of its cooling systems, in consequence of the SBO. After an electric trip was triggered, leading to the shutdown, the RCIC remained inoperative. With a lowering of the reactor water level, the HPCI started one hour after RCIC shutdown. Concerning the mode of operation HPCI is alike RCIC but is able to provide a much higher coolant injection flow rate. To ensure the high coolant injection, the HPCI draws larger amounts of steam from the reactor. In addition to prevent a shutdown of the system in consequence of a high water level in the RPV, a test line was activated. With the test line activated, the HPCI circulates coolant without injecting it into the RPV, remaining operational.
While the HPCI kept in service, the reactor pressure dropped significantly as a consequence of steam taken from the RPV. In the next seven hours the pressure decreased from 7.3 MPa to 0.8 MPa, whereby the design condition of 1.0 MPa was undershot at 08:00 p.m., on March 12. Later investigations show that no water was injected into the RPV anymore when design condition was reached. Alike to the shutdown of the RCIC, the HPCI cutoff was caused by a triggered trip due to low pressure in the RPV to protect the

HPCI turbine. Unaware of the fact that no water was injected into the RPV, the operators kept the HPCI in service. With the test line activated, steam was continuously taken out of the RPV to keep the HPCI running. Approximately 36 hours after SBO, the HPCI was shut down manually to prevent a sudden cutoff due to the low pressure and the declining amount of steam which could be extracted out of the RPV. Additionally the operators' intention was to start an alternative water injection with fire pumps provided. After deactivation of the HPCI, no steam was taken from RPV anymore, resulting in an immediate increase of reactor pressure. The water injection with fire pumps failed in consequence of the rapidly rising pressure. With the DC batteries depleted, depressurization of the RPV was not possible while pressure further increased. The pressure reached about 7.0 MPa two hours after HPCI shutdown. As a consequence of missing water injection into the RPV, the reactor water level started to decrease. While the pressure remained constant for the next 5 hours, the water level dropped about two meters below top of active fuel. The partially uncovered core started to heat up and zirconium-steam reactions proceeded leading to core heat-up and degradation.
Approximately 42 hours after SBO, enough batteries were provided to open the motor driven valves, leading to a successful depressurization of the RPV. At the same time the water injection with fire pumps could be started. The pressure dropped immediately from about 7.0 MPa to about 0.6 MPa with decreasing trend to 0.2 MPa. At 09:00 a.m. on March 13, the pressure increased to 0.9 MPa only for a short period, decreasing instantly to 0.2 MPa. It is assumed that molten fuel dropped into the lower plenum, causing massive steam generation. [1]; [2]; [3]; [4]; [5]
With the occurrence of core degradation, the considered period of time in the following analyses will end 50 hours after earthquake.

## 3. ASSESSMENT CRITERIA FOR ATHLET-CD

To simulate severe accidents in a reasonable way it is a necessary requirement of code to enable the modeling of parameters and boundary conditions referring to a given accident. This is essential to simulate automated accident measures as well as actions by operators during an accident. In contrast to analyzing experiments the boundary conditions as well as the initial parameters are not precisely defined for the accident analyzed in this work. Since there are less specific measuring sensors in power plants than in experimental set-ups, less parameters are recorded during the course of an accident. For the accident in Fukushima-Daiichi Unit 3 no data for hydrogen generation, molten mass as well as relocated mass to the lower plenum are available. The course of the RPV water level as well as the RPV pressure were recorded during the accident and can be compared to ATHLET-CD simulation results which means the focus is on the thermohydraulic behavior.
Concerning the assessment of this behavior sufficient capability is given if no inconsistencies regarding the ATHLET-CD simulations occur and the measured data are reproduced in a good agreement taking into account the modeled boundary conditions. Since the measured data can be afflicted with unknown uncertainties the ATHLET-CD simulation results are additionally compared to simulation runs performed with the codes MAAP5 and MELCOR depicted from literature for comparative assessment to show plausibility where no measured data are given.

## 4. ATHLET-CD MODELING

The severe accident analysis code ATHLET-CD 3.0A (Analysis of Thermal-hydraulics of Leaks and Transients – Core Degradation) is developed by the German TSO Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH in cooperation with the university of Stuttgart (Institute of Nuclear Technology and Energy Systems).
For the simulation of the accident in Unit 3 of Fukushima-Daiichi an input deck, based on a SWR-69 generic boiling water reactor was adapted. The input deck was modified to represent a General Electric BWR 4. The containment behavior is considered as boundary conditions. In the input deck the RPV

height is assumed with 21 m while the height of the original RPV is 21.9 m. The diameter is modeled with 5.61 m while the original diameter is 5.57 m. These deviations are small and can be neglected for the analysis. Further the input deck is limited to the primary coolant circuit. The modeled RPV consists of downcomer, lower plenum, core region, water separator, steam dryer, steam dome and main steam lines. Figure 1 shows the nodalization of the modeled RPV. For better representation and unterstanding only so-called thermo fluid dynamic objects (TFO) are shown in Figure 1 while the names of the TFOs are explained in Table I. While the TFOs represent the flow paths the objects representing the structures (e.g. walls) are not shown.

The core consists of six core sections, modeled as concentric rings. The core sections are axially subdivided in 23 nodes for a better representation of core degradation. A defined number of fuel rods and control blades are modeled for each of these core sections:

- 1st core section: 2341.8 fuel rods and 8 control blades (ROD1)
- 2nd core section: 4683.7 fuel rods and 16 control blades (ROD2)
- 3rd core section: 5269.1 fuel rods and 18 control blades (ROD3)
- 4th core section: 7903.7 fuel rods and 27 control blades (ROD4)
- 5th core section: 8781.9 fuel rods and 30 control blades (ROD5)
- 6th core section: 11123.8 fuel rods and 38 control blades (ROD6)

Thereby the 1st core section is the innermost, the 6th section is the outermost. In total, 41104 fuel rods and 137 control blades are modeled in the input deck. Based on the fact that one control blade is designed to control four fuel assemblies, 548 fuel assemblies are considered. These parameters correspond to Unit 3 of the Fukushima-Daiichi power plant [6].

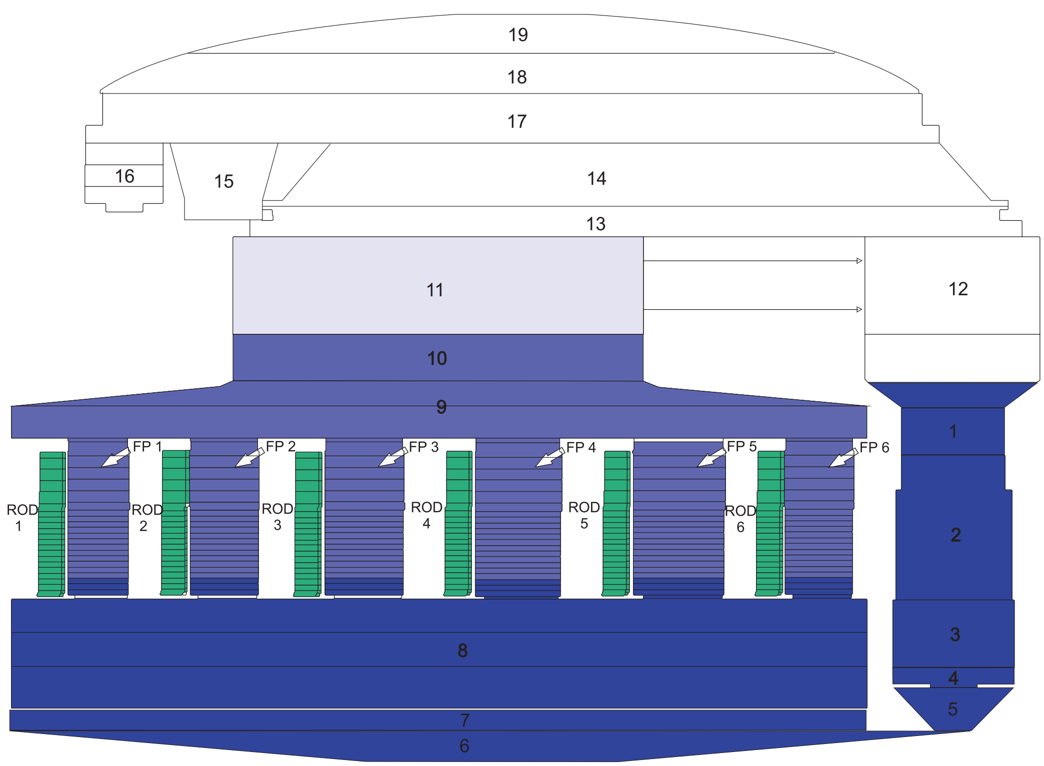


**Figure 1. Nodalization of the RPV in the input deck**

The total power is assumed with 2381 $MW_{th}$ , referring to the thermal capacity of Unit 3 in nominal operation. The steam flow rate was adapted to Unit 3 and is considered with 1233.33 kg/s [6]. To simulate a steady state condition, the feedwater flow rate has to be consistent with the steam flow rate. Further the feedwater injection as well as the injection of the emergency cooling systems is modeled as fill.

**Table I. Modeled TFOs**

| No. of TFO | Description |
|---|---|
| 1-3; 5 | Downcomer |
| 4 | Recirculation pump |
| 6-8 | Lower plenum |
| ROD 1-6 | Core section |
| FP 1-6 | Flow paths coupled to the core sections |
| 9-12 | Steam seperator |
| 13-16 | Steam dryer |
| 17-19 | Steam dome |

The core degradation module ECORE is applied to simulate important phenomena related to core degradation. To simulate the oxidation of zircaloy claddings the correlations “Cathcart” and “Urbanic/Heidrick” are selected [8]. These correlations are valid in the low (<1800 K) and high (>1800 K) temperature ranges. The temperature defining start of melting of metallic core material is set to 2250 K. The melting temperature for fuel and oxides ($UO_2$-$ZrO_2$) is set to 2500 K [9]. Further the boron carbide of the absorber structures starts to melt at 1520 K, whereby the oxidation of the absorber blades was validated with the CORA-16 and CORA-17 experiments [10].
The simulation starts with a 900 s lasting steady-state phase to achieve convergence so that all physical parameters (e.g. pressure, temperature) are well-balanced. To simulate the start of the accident, SCRAM is modeled at t = 900 s, right after the pre-calculation phase. In the simulation the trend of the decay heat is given by a table. The decay heat is considered with 130.97 $MW_{th}$ immediately after SCRAM [7]. After SCRAM, the shutdown of the recirculation pumps, the stop of the feedwater injection as well as the closure of the main isolation valves is implemented. Referring to Unit 3, eight safety relief valves are modeled to simulate the pressure control during the accident. As boundary conditions the set points of the safety relief valves was taken into account. Table II shows the SRV set points.

**Table II. SRV set points [2]**

| | Pressure [MPa] | |
|---|---|---|
| Number of valves | opens | closes |
| 1 | 7.545 | 7.319 |
| 3 | 7.613 | 7.385 |
| 4 | 7.682 | 7.452 |

For the depressurization of the RPV at 42.37 hours, the opening of one valve is modeled to simulate the activation of the ADS (Automatic Depressurization System).
Further the times of start-up and the shutdown of the RCIC/HPCI injection are implemented, according to the times during the course of the accident, in order to simulate the operating emergency cooling systems.

The maximum flow rates of these systems are set to 26.94 kg/s (RCIC) and to 268.0 kg/s (HPCI) [2]. For the alternative water injection several sensitivity analyses are performed, considering a range of mass flow from 5 kg/s to 36 kg/s due to the fact that the total amount of water injected by fire pumps is not exactly known. Best results, concerning the measured data, are achieved, expecting a mass flow rate up to 10 kg/s. In order to simulate an adequate performance of the emergency cooling systems, the extracted steam has to be taken into account. For the simulation of the operating RCIC, it is interpolated by the following Table III. Steam, extracted to operate the HPCI, is calculated as a function of the injected coolant into the RPV. Based on a minimum flow rate of 5.5 kg/s (consideration of activated test line), the steam flow rate can increase up to 7.46 kg/s [2]. The time frame in the simulations ends at 50 hours, since relevant events leading to core degradation phenomena occurred in this time frame.

**Table III. Extracted RCIC steam mass flow rates**

| Differential pressure [MPa] | Steam mass flow rate [kg/s] |
|---|---|
| 0.0 | 0.78 |
| 1.063 | 0.78 |
| 7.998 | 2.94 |

With the modeled boundary conditions two simulation runs were performed. These two runs differ in the time modeled when the HPCI injection stops. In one simulation the HPCI injection stops after 36 hours basing on first investigations of TEPCO while in the second run HPCI stops after 29 hours taking into account a new assumption made by TEPCO [5].

## 5. RESULTS

Since the focus of these analyses is on the capability of the code ATHLET-CD to reproduce the thermohydraulic behavior during severe accidents, only selected parameters are discussed in the following part. Hereafter two simulation runs will be compared to measured data of Fukushima-Daiichi Unit 3. The two shown simulations differ in the assumption of the point of time when water injection into the RPV, provided by the HPCI, was stopped. The results of the simulation in which the coolant injection stops after 36 hours is named run A and is colored red. The black colored line represents the simulation in which the injection stops after 29 hours, regarding to the assumption made by TEPCO [5]. This simulation run is named run B. Further the modeled steam-output is stopped in both ATHLET-CD simulations after 36 hours.

Three additional simulations based on the stated literature are depicted for comparative assessment. The purple line represents the results of a simulation performed by the Electric Power Research Institute (EPRI) in 2013 using the code MAAP5 (Modular Accident Analysis Program Version 5) [10]. The blue line represents results of a simulation performed by the Japan Nuclear Energy Safety Organization in 2012 using MELCOR 1.8.5 [11]. The green line represents a MELCOR 2.1 simulation performed by Sandia National Laboratories in 2012 [3]. For the MAAP5 and the MELCOR simulations a containment is modeled contrary to the ATHLET-CD simulations.

Figure 2 shows the measured and the simulated pressure in the RPV during the course of the accident. Until 42.5 hours after simulations start, measured data are reproduced in a quite good accordance by all simulations. During the first 22 hours, a good agreement of the results to the measured data is achieved by the descripted pressure depending control of the SRVs in the ATHLET-CD simulations. For the MAAP5 and MELCOR simulations similar boundary conditions regarding the SRVs were implemented and can be found in the stated literature. The rapid pressure decrease at 22 hours, caused by the high mass flow rate of coolant, injected by the HPCI during the accident, is simulated without any mayor discrepancies by run A, run B and MAAP5. Until approximately 30 hours the pressure decreases to 1.0 MPa and remain constant for the next six hours, represented by the ATHLET-CD and the MAAP5 simulations. In the MELCORa simulation the pressure drops to 1.0 MPa at 32 hours and is calculated similar to the measured data and the ATHLET-CD and MAAP5 simulations. Until 30 hours the results of run A and run B are identical with regard to the different assumptions modeled for these two simulation runs. From 30-36 hours a slightly lower pressure, compared to run B and the measured data, is calculated in run A. At 36 hours there is a minor pressure drop simulated by run A while a constant course of pressure is calculated in run B. The decrease of the pressure at this point is indicated by the measured data as well. From this point the pressure increases rapidly caused by the cutoff of the steam supply. The course of increasing pressure is simulated in all simulation runs in quite good accordance to the measured data. The increase of pressure is simulated in a better accordance to the measured data by run B.

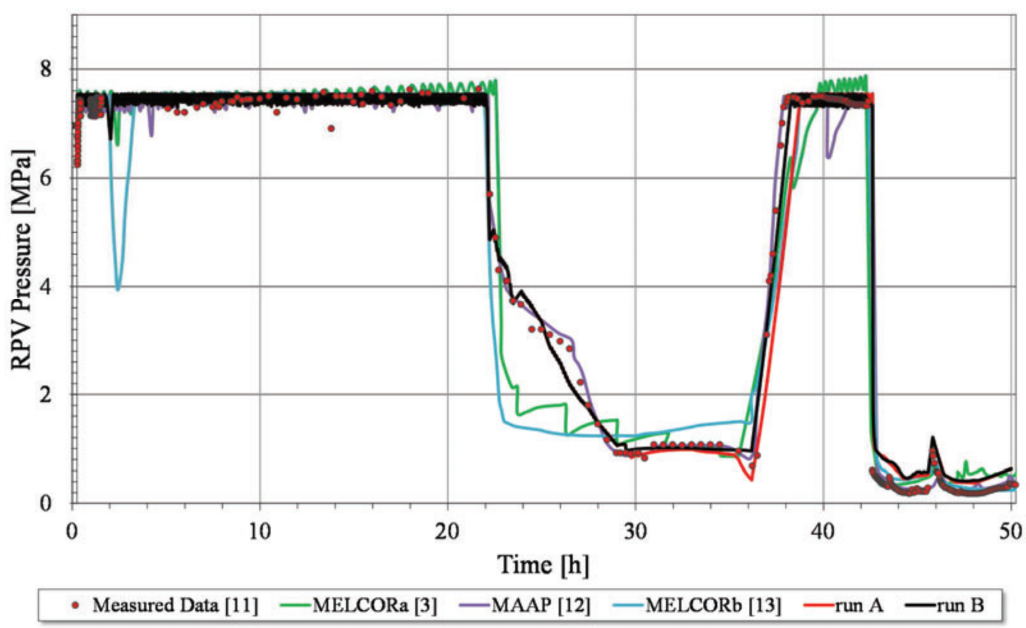


**Figure 2. Measured and simulated RPV pressure**

Before depressurization the pressure is simulated nearly identical by run A and run B. The depressurization at 42.5 hours is represented by all simulations as a rapid pressure drop. Since there are no measured data available during depressurization of the RPV it can be seen that the depressurization is simulated in good accordance by all codes. Further it can be noted that the pressure is calculated in good agreement to the next available measured data. The ATHLET-CD simulations are overestimating the pressure by 0.6 at 42.5 hours. After 44 hours the pressure is overestimated by the ATHLET-CD simulations while the measured data are reproduced in good agreement by the MAAP5 and the MELCORb simulations.

Figure 3 shows the measured and the simulated liquid level in the RPV as well as the mass flow rate of the coolant injection for the ATHLET-CD simulations. Thereby the results concerning the liquid level are oriented to the primary axis and the curves of the mass flow rate are oriented to the secondary axis of the diagram. The mass flow rates are only shown for the ATHLET-CD simulations since no homogenous representation of the mass flow rate for the other simulations is available. For the first two hours at the beginning of the simulations, the liquid level is underestimated by all shown simulations. At this time the modeled fill in ATHLET-CD, representing the RCIC is only activated for 20 min referring to the RCIC operations during the accident. After two hours there is a continuous injection simulated in both

ATHLET-CD runs, resulting in a constant representation of the liquid level for nearly 20 hours. At approximately 21 hours the injection stops, taken into account that the RCIC tripped at this time. Without any activated coolant injection the water level drops in accordance to the measured data which is represented by all simulations. With the activation of the modeled fill, representing the HPCI, one hour after RCIC injection stopped, the liquid level is reproduced with a rising trend by run A as well as by run B.

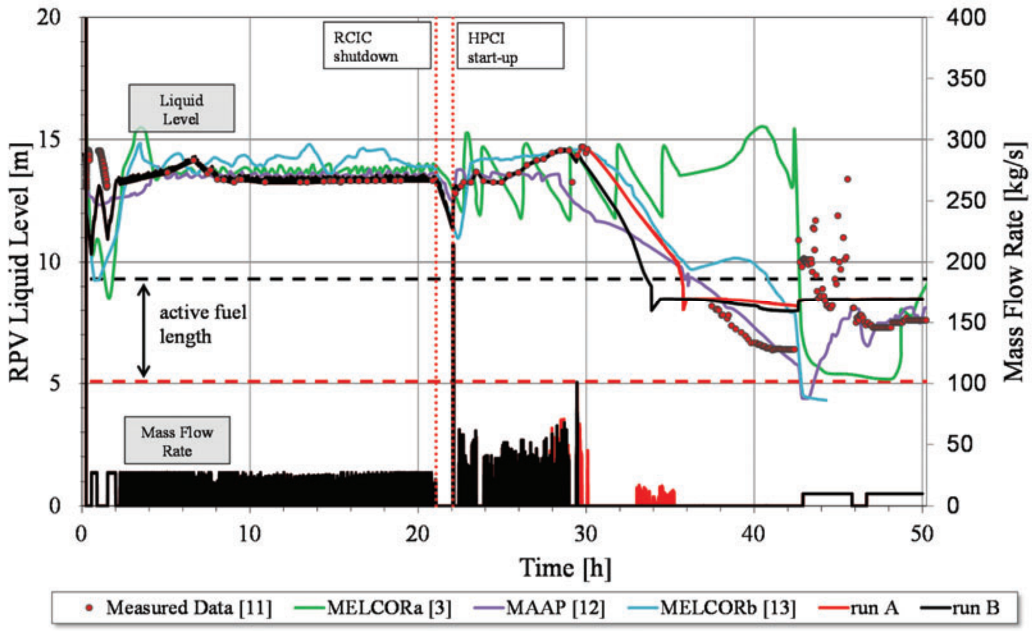


**Figure 3. Measured and simulated liquid level and mass flow rate**

Form 30 hours to 42 hours the behavior of the calculated water level by the different codes differ. At 29 hours after initiating event, the stop of coolant injection into the RPV is assumed. For this assumption the results are represented by run B. The injection stops ultimately at approximately 36 hours in the other simulations. At this time the HPCI injection is reduced due to low pressure in the RPV. As a result the water level decreases. This is captured by all simulations. Since the water injection is stopped at 29 hours in run B the water level starts to fall 0.5 hours earlier than in run B. After 34 hours the top of active fuel is uncovered in run B. The uncovering of fuel in run A starts two hours later nearly simultaneously with the MAAP5 simulation. For the time the fuel is uncovering there are measured data available. The water level below the top of active fuel was recorded approximately 38 hours after SCRAM. The measured water level is overestimated up to two meters in the ATHLET-CD simulations. At 42.5 hours an alternative water injection is started in the ATHLET-CD simulations, as a consequence the liquid level rises. The water injection results in a slight increase of the simulated water level in run A as well in run B. For the rest of the simulation time, both curves remain constant, overestimating the measured water level by one meter while the water level is reproduced in a good agreement by the MAAP5 simulation.
While partially uncovered, the core starts to heat up and oxidation of zirconium in steam atmosphere begins. As a result of these reactions, hydrogen is released and the core heat-up proceeds due to the exothermal reaction leading to core meltdown when not cooled on time. Figure 4 shows the released hydrogen mass calculated with the severe accident codes. The occurrence of larger amounts of hydrogen is taken as evidence for rising core temperature. For this parameter there are no measured data for Unit 3 of the Fukushima-Daiichi power plant available in the open literature yet.

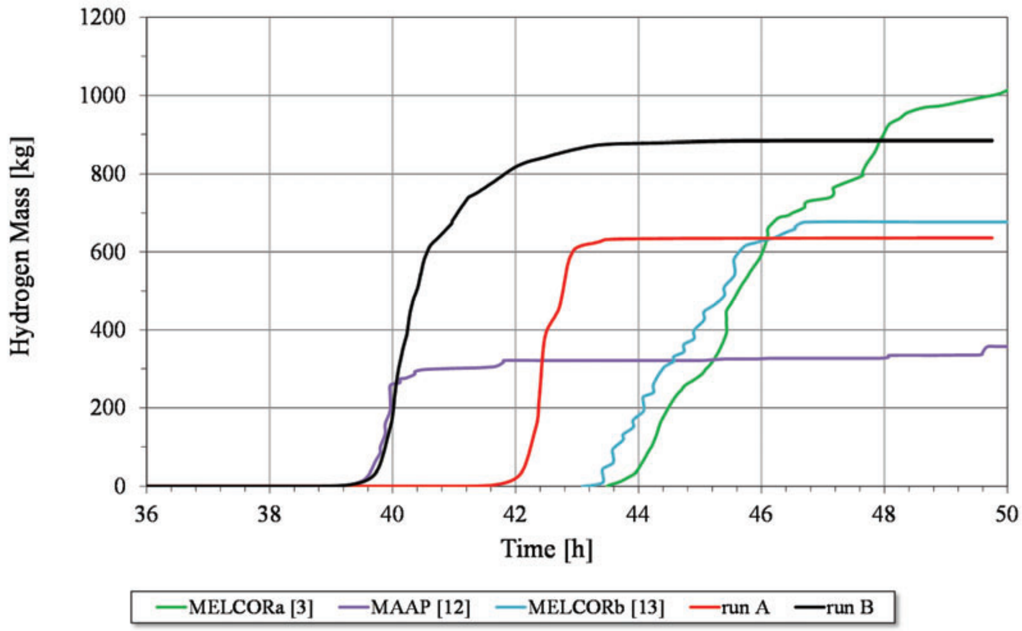


**Figure 4. Hydrogen release simulated by ATHLET-CD**

At 34 hours after the calculated water level starts to fall below the top of active fuel, as seen in Figure 3, the core begins to heat up in run B. Five hours later, at approximately 39.5 hours, first hydrogen release due to oxidation is simulated. In the MAAP5 simulation first hydrogen release is calculated simultaneously with run B. With a steep gradient of released hydrogen a total hydrogen mass of nearly 880 kg is calculated in run B at 42.5 hours while only 320 kg of $H_2$ are calculated in the MAAP5 simulation. In run A first hydrogen release is calculated at approximately 41.5 hours although the core started to uncover nearly at the same time compared to the MAAP5 simulation. Considering run A first hydrogen is released two hours later than in run B due to the HPCI water injection until 36 hours. In this run a total hydrogen mass of 620 kg is calculated. In the ATHLET-CD simulations the hydrogen generation seems to be stopped with the starting alternative water injection at 42.5 hours. Hydrogen release is calculated in the MELCOR simulations nearly two hours later compared to run A and four hours later compared to run B. While a total hydrogen mass of about 1000 kg are calculated in the MELCORa simulation, a total hydrogen mass of nearly 670 kg is reproduced by the MECLORb simulation.

Figure 5 shows the mass of molten material in the core for the ATHLET-CD simulations. The simulated molten mass is only shown for the ATHLET-CD simulations since no homogenous representation of the mass is available for the other simulations. Further there is no data of molten mass in Unit 3 available. As seen in Figure 5, the start of core meltdown is simulated nearly 0.5 hours after first hydrogen release by both ATHLED-CD runs. The gradient of the molten mass corresponds with the gradient of the hydrogen release. After two hours a completely molten core is simulated in run B. The total amount of molten mass in this simulation is approximately 150 t. Since no melt relocation into the lower plenum is considered most of the core is molten despite the fact that the water level remains constant.

Start of core meltdown is simulated at nearly 42 hours in run A, as a consequence of the later stopped coolant injection as well as the delayed start of oxidation of the fuel rods. In this simulation only 20 t of molten mass are calculated by ATHLET-CD before the alternative water injection starts. Due to the cooling ability of the injected water, the core meltdown stops in run A and the degraded mass remains constant, while the core is completely molten in run B before water is injected.

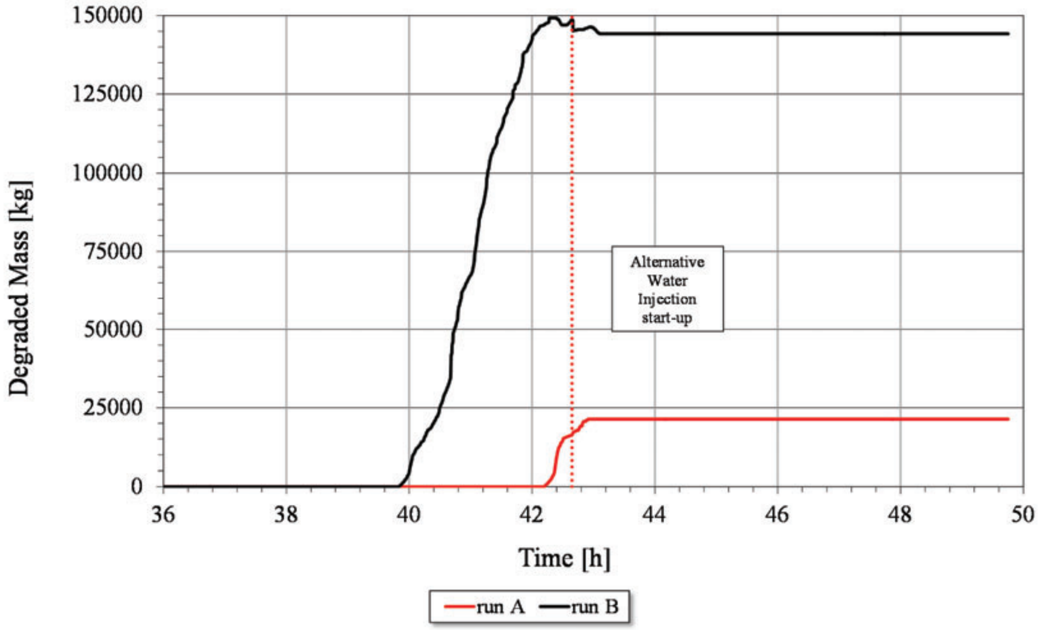


**Figure 5. Degraded mass simulated by ATHLET-CD**

Several stages of core degradation from heat up to meltdown are shown in Figure 6 for calculation runs with ATHLET.CD. The visualization represents the inner core section ROD 1 plus the canister wall and the control blade for selected times. The temperature is indicated by the color, while blue colored objects are about 300 K, the bright yellow colored objects are about 2000 K. The comparison of the detailed figures reveals the time difference in the two simulation runs. In the figure, the illustrations belonging to run A are presented in the left column, the illustrations belonging to run B are shown in the right column for four times, at 39 hours, at 40 hours, at 42 hours and at 50 hours, respectively.

At 39 h run A shows that an adequate axial temperature distributions of the fuel rods, the canister wall and the control blade are simulated for this condition, since the core is left without any cooling for nearly three hours. While the rod and the BWR structures in run B indicate significantly higher temperatures, assuming no core cooling for almost seven hours as a consequence of the earlier HPCI injection stop. One hour later at 40 h, the next two illustrations show just a slight temperature rise in run A for the rod and BWR structures. In comparison the rod and the canister wall still heat up in run B, reaching a temperature of 2000 K in several nodes, while the control blade already has failed in middle section. At 42 h, right before the alternative water injections starts, a temperature of nearly 1000 K is simulated for the rod and structures in run A. The figure of run B shows a complete meltdown of the rod and failure of the structures excepting the upper node of the rod and the lower node of the control blade.

The last seen illustrations show the condition of the inner part of the core at the end of the simulations. Considering run A, the canister wall plus the control blade are nearly molten completely, while most of the fuel in the rod remains solid due to higher liquidus temperature of the fuel and to the fact alternative water injection has started. As a consequence of the completely molten core in run B at 42 h, the considered figure at 50 h shows no considerable deviations, except for the impact of the coolant injection, resulting in a temperature decrease in the remaining upper node of the rod and in the lower node of the control blade.

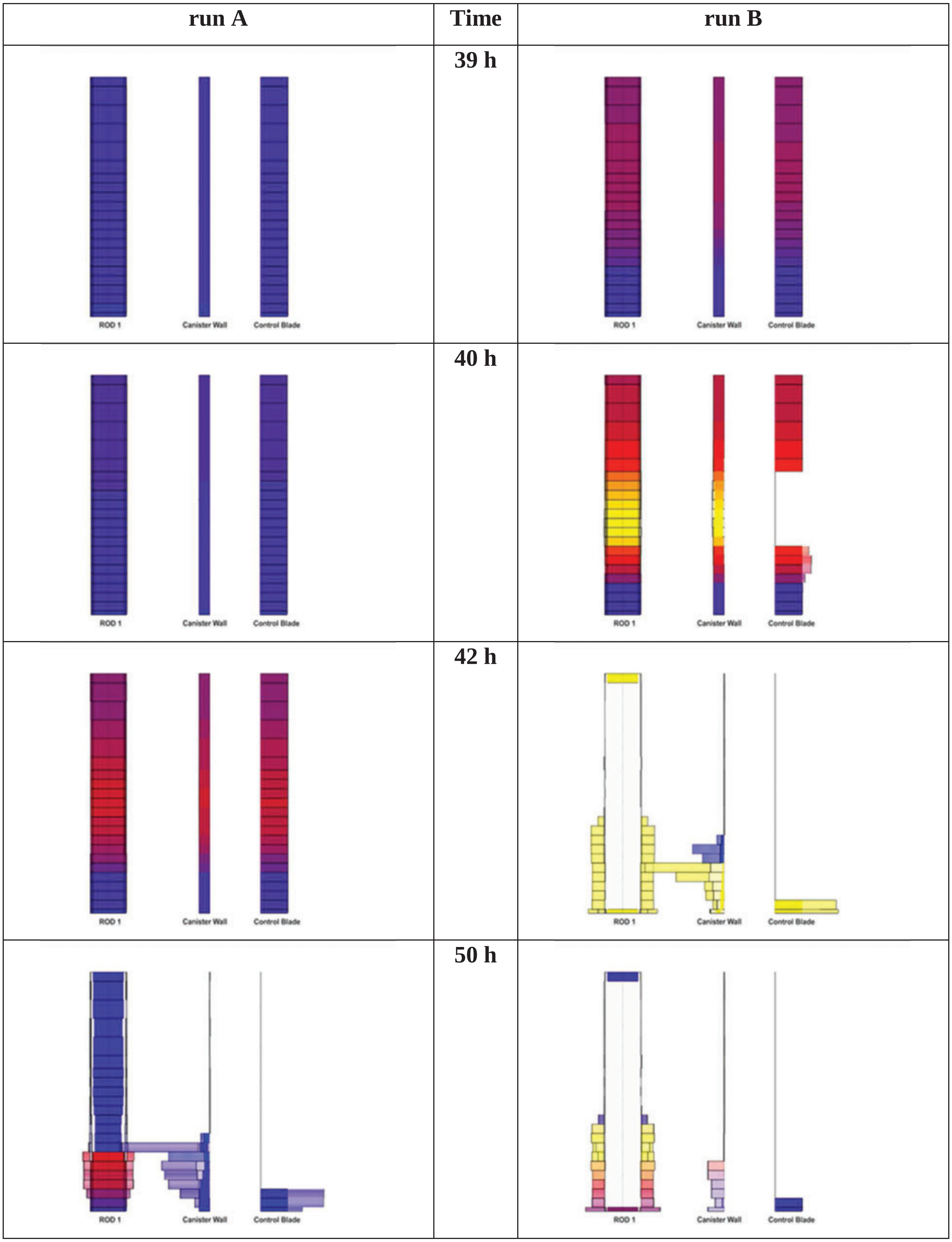


**Figure 6. Stages of core degradation**

## 6. CONCLUSIONS

Since the aim of the research activities described in this paper is the assessment of the applicability and the capability of ATHLET-CD to simulate accident progression in a BWR it has to be noted that the method for this is based on a comparison of different simulation runs given in literature as well as a comparison to the measured data of Unit 3 of the Fukushima-Daiichi power plant as far as possible. Although these data can be afflicted with uncertainties due to sensor failures caused by the accident.
The opportunity to adapt the input deck to the plant specific data as well as to the boundary conditions of the accident in Unit 3 enabled the simulation of events of the accident. In the first hours the pressure behaviour is simulated in dependence of the modeled SRV set points. Further the automated RCIC and the HPCI emergency coolant injection into the RPV is implemented resulting in an adequate modeling. Concerning the evaluated results, ATHLET-CD reproduces the thermohydraulic behaviour of the plant during the accident in Unit 3 in good agreement to the measured data and is reasonable comparative to other codes. From discussed RPV pressure and the reactor water level it can be stated that no inconsistencies occurred in the simulation runs. It can be concluded that ATHLET-CD shows an adequate applicability and capability making it possible to simulate complex sequences of accidents taking into account the related boundary conditions.
Further the behaviour of the core regarding the hydrogen release and the meltdown are reproduced in a reasonable way referring to the time of first hydrogen release and the beginning of the core meltdown. The delayed stop of coolant injection, assumed in run A, has only a minor impact on the simulated RPV pressure, compared to run B and the measured data. As expected the simulated water level in run A drops later than the water level in run B, due to a longer lasting coolant injection. As a consequence of the HPCI injection stop at 29 hours in run B, the uncovering of the core, hydrogen release and core meltdown occurs earlier in this simulation. A complete meltdown is calculated in run B, even before first hydrogen is simulated in run A. Due to a later stop of coolant injection, while the point of start-up of alternative water injection is assumed constant, the time without any coolant injected into the RPV is shorter. While in run B the core is completely molten, only 20 t of molten mass are calculated by the code in run A.
In addition the comparison of the two simulated ATHLET-CD runs demonstrates the impact of the operating time of the HPCI, concerning core degradation phenomena. Regarding the assumptions made by TEPCO referring the HPCI shut down times it has to be stated that, based on the ATHLET-CD simulations, severe core degradation might have occurred in the course of the accident in Unit 3.
Until now the working package of the collaborative research project WASA-BOSS, involving this work, is not finally completed so that further ATHLET-CD simulations regarding Unit 3 will be performed. Within this working package additional adaption of the input deck will be done as well as analyses of the late phase modelling will be performed. With the next version of ATHLET-CD a new model for melt relocation will be implemented. Simulations regarding melt relocation will be performed with the new version analysing the melt behaviour in the lower plenum and as well as the probability of potential lower head failure. Further it is planned to perform code-to-code comparisons with MELCOR. As a starting point, independent calculations are done by Ruhr University Bochum with ATHLET-CD and the Karlsruhe Institute of Technology with MELCOR using the same boundary conditions and assumptions. The code-to-code comparison will provide further insights in order to assess the capabilities of ATHLET-CD regarding the modelling of late-phase core degradation phenomena

## ACKNOWLEDGMENTS

This work is sponsored by the German Federal Ministry of Education and Research (BMBF) in the frame of the collaborative research project WASA-BOSS (02NUK028G).
The ATHLET-CD simulations were performed with the code version ATHLET-CD3.0A, developed by GRS.

SPONSORED BY THE
Federal Ministry of Education and Research

## NOMENCLATURE

| | |
|---|---|
| ATHLET-CD | Analysis of Thermal-hydraulics of Leaks and Transients – Core Degradation |
| BMBF | German Federal Ministry of Education and Research |
| GRS | Gesellschaft für Anlagen- und Reaktorsicherheit mbH, German TSO |
| HPCI | High Pressure Coolant Injection |
| RCIC | Reactor Core Isolation Cooling |
| RPV | Reactor Pressure Vessel |
| SBO | Station Blackout |
| SWR-69 | German boiling water reactor type 69 |
| TSO | Technical Safety Organization |

## REFERENCES


1. The Institute of Nuclear Power Operations, "*Special Report on the Nuclear Accident at the Fukushima Daiichi Nuclear Power Station*, pp. 28-32, Institute of Nuclear Power Operations, Atlanta, Georgia, USA (2011).
2. Gauntt, R.; et al., "*Fukushima Daiichi Accident Study (Status as of April 2012)*", Sandia National Laboratories, Albuquerque, New Mexico, USA (2012).
3. Band, S.; et al., „Fuk*ushima Daiichi 11. März 2011- Unfallablauf | Radiologische Folgen"*, pp. 22-24, Media Cologne Kommunikationsmedien GmbH, Cologne Germany (2013).
4. Ahn, J.; et al., "*Reflections on the Fukushima Daiichi Nuclear Accident: Toward Social-Scientific Literacy and Engineering Resilience"*, pp.41-43, Springer International Publishing (2013).
5. Tokyo Electric Power Company, Inc. "*Report on the Investigation and Study of Unconfirmed/Unclear Matters In the Fukushima Nuclear Accident Progress Report No.2*" (2014).
6. Tokyo Electric Power Company, Inc., "https://fdada.info/docdata/accident_analysis/PS-Unit3-01.pdf" (2012).
7. Tokyo Electric Power Company, Inc., "https://fdada.info/docdata/accident_analysis/ES-Unit123-05.pdf" (2012).
8. G. Lerchl, H.; Austregesilo, P.; Schöffel, D.; et al.: "*ATHLET-CD Mod- 3.0 Cycle A – User's Manual*," Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH (2012).
9. Hoffmann, M.; Koch, M.K.: "SIMULATION OF THE TMI-2 ACCIDENT WITH ATHLET-CD AND ANALYSIS OF THE LATE-PHASE MODELING", *Proceedings of the 2012 20th International Conference on Nuclear Engineering collocated with the ASME 2012 Power Conference*, Anaheim, California, USA, July 30-August 3, pp. 1-9 (2012).
10. Di Marcello, V.; Imke, U.; Sanchez, V.: " *Qualification of ATHLET-CD for BWR based on the KIT Bundle Tests CORA-16 and CORA-17*", Technical Report to the BMBF Project WASA-BOSS, KIT – University of the State of Baden-Wuerttemberg and National Research Center of the Helmholtz Association (2014).
11. Idaho National Laboratory: Fukushima Daiichi Accident Study Information Portal "https://fukushima.inl.gov/", (2013).
12. Electric Power Research Institute (EPRI): "*Fukushima Technical Evaluation, Phase 1 – MAAP5 Analysis"* Electric Power Research Institute, Palo Alto, California, USA (2013).
13. Hoshi, H.; Kawabe, R.: "*Accident Sequence Analysis of Unit 1 to 3 Using MELCOR Code"* Japan Nuclear Energy Safety Organization (JNES) (2012).