THE FINDINGS OBTAINED DURING THE OECD/NEA BSAF ACTIVITY WITH THE EMPLOYMENT OF THE SAMPSON CODE

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ABSTRACT

The Organization for Economic Cooperation and Development, Nuclear Energy Agency (OECD/NEA) launched the Benchmark Study for Analysis of Fukushima Daiichi Nuclear Power Station (BSAF) project after the accident occurrence. The Institute of Applied Energy (IAE) has worked as the technical supporting organization for the operating agent, Japan Atomic Energy Agency (JAEA), of the BSAF project. The IAE has investigated the operative conditions of the plant equipment during the accident in cooperation with TEPCO and gave it to the BSAF member organizations as common understanding. This paper describes some events and phenomena that occurred which have been deemed specific to the Fukushima Daiichi Nuclear Power Plant: (1) direct steam release from the reactor pressure vessels (RPVs) to the drywells, (2) corium release from penetrating pipes at the RPV lower heads onto the pedestal floors, (3) Alternative water injection by the fire pumps, and (4) off-design operation for the reactor core isolation cooling system (RCIC) and the high pressure coolant injection system (HPCI), and calculation results of the accident progressions at Units 1, 2, and 3 at present using the SAMPSON code with new modellings for the above phenomena.

(1) Unit 1: 77% of the core materials had already melted down onto the pedestal floor at 15 hours after scram when the alternative water injection started and almost 100% at 100 hour after scram.
(2) Unit 2: 25% of the core materials had been relocated in the lower plenum at 105 h after scram when the core was re-flooded by the alternative water injection.
(3) Unit 3: Finally 80-100% of the core materials had melted down onto the pedestal floor depending on the HPCI performance and the alternative water mass flow rate into the core by the fire truck.

KEYWORDS
Fukushima Daiichi, OECD/NEA BSAF, SAMPSON, Severe Accident

1. INTRODUCTION

The Fukushima Daiichi Nuclear Power Plant (NPP) Units 1-4 suffered serious damage from the huge tsunami which followed the Great East Japan earthquake of magnitude 9.0 at 14:46 on March 11, 2011. The reactors shut down safely upon receiving the scram signal triggered by the earthquake. The loss of off-site power occurred immediately since power line pylons collapsed due to the landslip. After that the emergency diesel generators immediately started to supply AC power. At about 50 minutes after the occurrence of the earthquake, the station blackout (SBO) occurred after emergency power supply equipment facilities (AC power facilities) were submerged by tsunami sea water. These were the common events among the Units. At Units 1 and 2, not only AC powers but also DC power facilities were inundated and lost the function by the tsunami. At Unit 3, the DC power facility was survived.

At Unit 1, there was no intentional decay heat removal after the tsunami until the alternative water injection by the fire truck became effective. At Unit 2, the reactor core isolation cooling system (RCIC) had worked for 66 hours and then there was a time delay of 11 hours after the RCIC stop until the alternative water injection by the fire pump became effective. At Unit 3, the high pressure coolant injection system (HPCI) was switched off at 36 hours after the reactor scram. However, the HPCI water flow rate was considered to be insufficient for the decay heat removal since its work pressure was almost at the design lower limit, and there was a time delay of 6.7 hours after the HPCI switch-off until the alternative water injection by the fire truck became effective. Thus, core meltdowns occurred at all Units 1-3. The core meltdowns continued even after the alternative water injections by the fire trucks because the water injection mass flow rates into the cores were too small to remove decay heat and they often interrupted.

The Organization for Economic Cooperation and Development, Nuclear Energy Agency (OECD/NEA) launched the Benchmark Study for Analysis of Fukushima Daiichi Nuclear Power Station (BSAF) project on November, 2011. The Institute of Applied Energy (IAE) has worked as the technical supporting organization for the operating agent, Japan Atomic Energy Agency (JAEA), of the BSAF project. The IAE has investigated the detailed plant specifications and operative conditions of the plant equipment during the accident in cooperation with TEPCO and gave them to the BSAF member organizations as common understanding for the analyses of the accident progressions with their own severe accident analysis codes. The high priority objective of the BSAF project was to find locations and amounts of the corium through accident analyses. The time span for the analyses was 6 days (or, 144 hours) after scram.

This paper describes some events and phenomena that occurred which have been deemed specific to the Fukushima Daiichi NPP and calculation results of the accident progressions at Units-1, 2, and 3 using the SAMPSON code with new modellings for the above phenomena. The work described in this paper is based on the previous study (Ref. [1], [2]) and especially, discussions are focused on the phenomena in the RPV (so called in-vessel phenomena).

2. SPECIFIC PHENOMENA TO THE FUKUSHIMA DAIICHI NPP

There were some events and phenomena that occurred which have been deemed specific to the Fukushima Daiichi NPP. Such events and phenomena should be common conditions for accident progression analyses by the BSAF members. Having the common conditions, the BSAF members had developed their own modellings for analyses.

This chapter describes such specific events and phenomena, and also modellings developed by the IAE.

2.1. Direct Steam leakage from the RPVs to the Drywells

2.1.1. Damage to SRV gasket [1]

The safety relief valve (SRV) had intermittently worked keeping the pressure in the reactor pressure vessel (RPV) around 7.5 MPa for a long time until the RPV pressure decreased lower than the SRV activation pressure. The steam generated in the RPV was released into the suppression pool (S/P) through the SRV during the intermittent SRV operation period as shown in Figure 1 (red solid line), since the main steam isolation valve (MSIV) had closed as reactor shut down. Then, the water level in the RPV decreased due to the steam release from the SRV and the core would be exposed resulting in high core temperatures and consequently high-temperature steam. The SRV gasket, which had a maximum design temperature of 723 K, could withstand such high temperatures, and underwent deformation resulting in leakage through the junction. In particular the gasket would deform and leakage could start from the RPV into the drywell, even when the SRVs had closed as shown in Figure 1 (red dashed line). Thus, the steam leakage from the gasket into the drywell was assumed to start when the steam temperature reached 723 K. The opening of the gasket due to deformation was assumed to be along the circumferentially 60° in the analysis with the SAMPSON code. These were just the assumptions without any evidence.

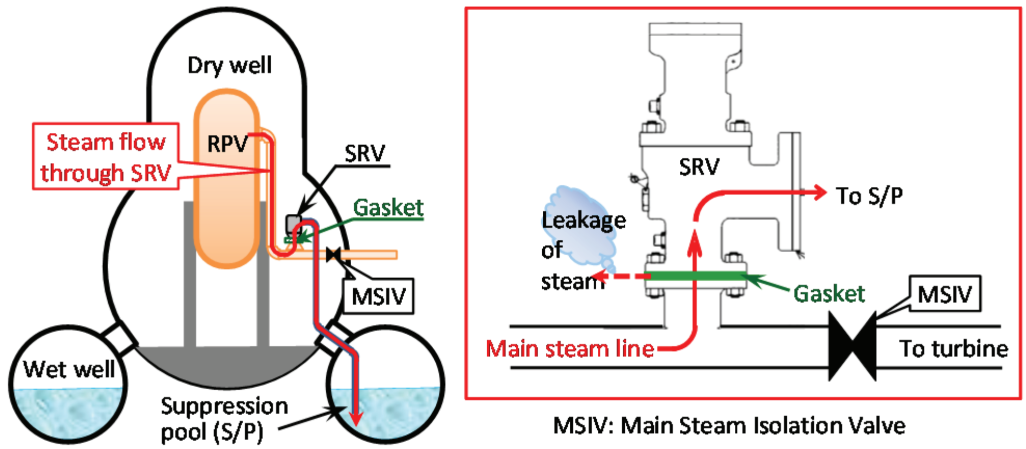

Figure 1 Leakage from SRV gasket

2.1.2. Buckling of SRM/IRM guide tubes [1]

Many tubes which penetrate the RPV bottom wall; control rod guide tubes (CRGTs), and in-core monitor guide tubes such as source range monitors (SRMs), intermediate range monitors (IRMs), traversing in-core probes (TIPs), and local power range monitors (LPRMs). The pressure boundary of the SRM and IRM guide tubes is inside the core, since their bottom ends are open to the drywell, while the pressure boundary of the other guide tubes is outside the RPV, as shown in Figure 2. The SRM/IRM guide tubes are the dry tubes, in which there is no water at all under the normal operating conditions, whereas the other in-core monitor guide tubes and the CRGTs are the wet tubes. The SRM/IRM guide tubes rarely break under the normal operating conditions, even having a pressure difference of about 7 MPa between the tube outside and inside. However, under the severe accident conditions they might buckle when their temperatures rise, even before reaching their melting point (about 1,700 K). The occurrence of the buckling would result in a direct release of steam into the drywell. In the analysis, the buckling condition was given by the simple Von Mises relation [3]. The details of the modelling were described in reference [1].

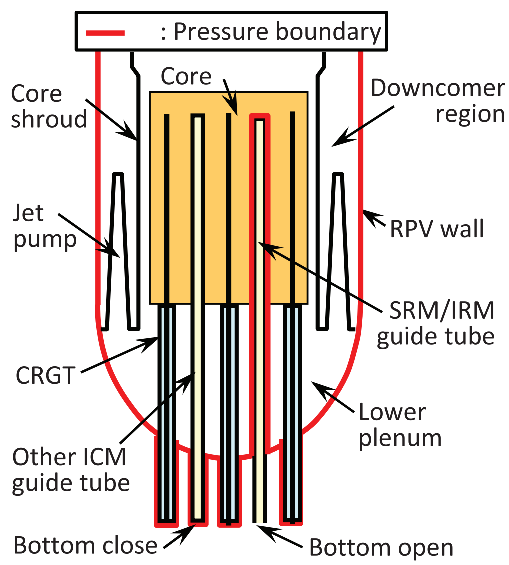

Figure 2 Pressure boundary

2.2. Direct Corium leakage from RPV to Pedestal

After the core melt occurrence, the corium falls down into the RPV lower plenum, where a lot of guide tubes are installed. Since the thickness of the in-core monitor guide tubes is much thinner than the CRGTs, they would melt earlier than the CRGTs. When they melt in the lower plenum, the corium would be released into the drywell from the melt portions of the SRM/IRM guide tubes, since the bottom ends of the SRM/IRM guide tubes are open to the drywell. The melt behaviour of the SRM/IRM guide tubes in the lower plenum was analyzed by solving the one dimensional heat conduction equation. This was newly programmed in the code after the completion of the previous calculations [1]. In the previous calculations, the guide tube was supposed to be a single-pipe type, but it was actually a double-pipe type and the gap between the inner and the outer tubes was plugged at the bottom end. Newly knowing this fact, the leakage of fluid from only the inner tube was considered in the present analysis, resulting in the smaller leak flow area than one in the previous calculations.

2.3. Melt Path [4]

Figure 3 shows three melt pathways, which were newly incorporated into the SAMPSON code.

(1) Some melts in the control rod region fall down inside the control rod guide tube. Since the flow area at the velocity limiter becomes narrow, some melts may solidify there.
(2) Channel box melts and some fuel melts fall down onto the core plate and accumulate on it. When the core plate breaks due to melting by direct contact with high temperature melts, the accumulated melts fall down into the lower plenum.
(3) Melts of fuels and the constituents inside the channel box fall down through the inlet orifice into the lower plenum.

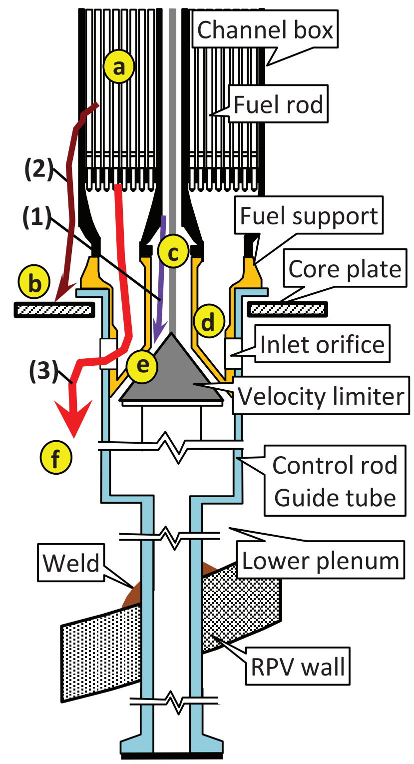

Figure 3 Schematic of CRGT

The previous SAMPSON model adopted the TMI behavior path. In the TMI accident, since water was present in the lower core region (wet core condition), generation of large amounts of steam would result in formation of a dense crust. In the Fukushima Daiichi NPP accident case, the melt relocation was considered to occur mainly after the water level decreased below the bottom of the active fuel. Under such dry core conditions, steam generation was considered to be insufficient to make a dense crust. And the melts were considered not to be kept above the core plate but to fall through the continuous drainage paths (1) and (3) as shown in Figure 3. The existence of such continuous drainage paths under BWR dry core conditions was confirmed by the XR-2 test at Sandia National Laboratory (SNL) [5].

Table 1 XR-2 test analysis

		Test	Analysis
Above core plate	a+b	9%	7%
	c	11%	15%
	d	11%	15%
Below core plate	e	37%	28%
	f	37%	43%

Table 1 shows comparison of analysis and XR-2 test result. In the SNL XR-2 test, metallic melts of stainless steel, B_4C, and zircaloy were used, and no oxidic melts such as UO_2. Since the temperature of the oxidic melts is much higher than one of the metallic melts, the melt behavior in the Fukushima Daiichi NPP cores might be different from ones obtained by the XR-2 test. However, the current test analysis gives a certain level of justification of this modelling.

2.4. Alternative Water Injection [4]

When the fire engine started alternative water injection, all the water it discharged could not reach the core, since there were some branch lines in the piping between the fire truck and the RPV. According to TEPCO findings, there were 10 branch lines in Unit 1, and 4 branch lines in Units 2 and 3 respectively [6]. The openings of the branch lines were at the turbine building where the pressure was almost atmospheric, whereas the RPV pressure must be higher than the atmospheric one and the location of the intake nozzle at the RPV was much higher than that at the fire truck. Therefore, it was quite possible that most of the water discharged from the fire pump leaked out through the branch lines. In the present analysis, such leak flows were calculated as functions of the RPV pressure transients by assuming friction losses in the pipe lines. The relation of the pump head and the discharged mass flow rate had to be also given for the calculation of the leak flow. The fire pump used was the A2 class [7]. Since the pump performance, or so-called "Q-H curve", was unclear, it was also assumed that the pump had specifications typical of the A2 class pump. The result for Unit 1 is shown in Figure 4.

At Unit 1, the alternative water injection started at 15 h after the scram. At the early phase from 15 h to 40 h after scram, the water mass flow rate into the core was only 0.07 – 0.075 kg/s because of the 10 branch lines with their outlets at atmospheric pressure and the considerably high RPV pressure. In order to remove decay heat generation of 8.3 MW at 15 h after scram by only evaporation of water, the water injection of about 3.16 kg/s was required, assuming an injection water temperature of 288 K. If all the water discharged from the pump could reach the core without any leakage, it had enough amount to remove the decay heat. However, especially at the early stage of water injection, the water mass flow rate into the core was quite insufficient for decay heat removal. And the water injection was often interrupted. Thus, even after the alternative water injection, the core meltdown continued.

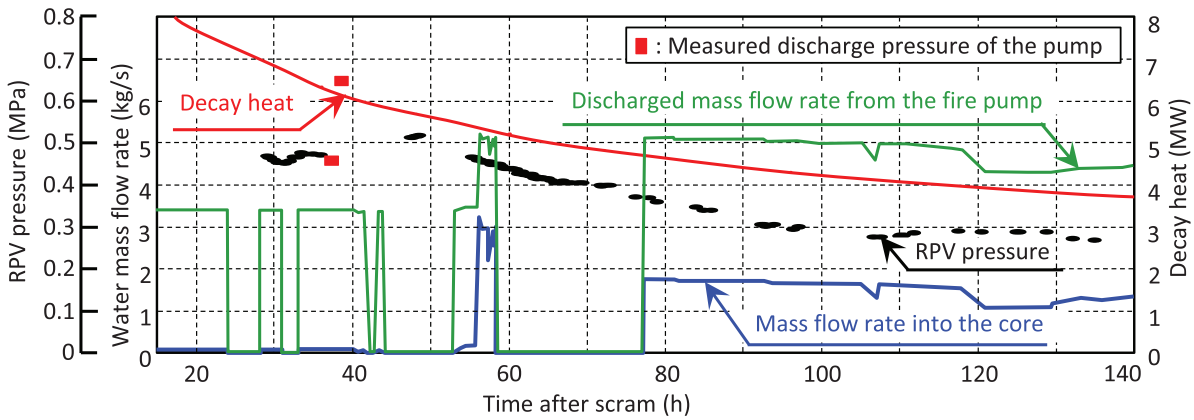

Figure 4 Mass flow rate of alternative water injection

2.5. Off-design Operation of RCIC and HPCI [1, 6]

For Unit 2, the RCIC was the only decay heat removal system available during the period after the reactor scram until the alternative water injection by the fire truck. The RCIC was originally designed to automatically start on receiving a signal of low water level in the core region and to automatically stop on receiving a high water level signal (so called L-8 level). The RCIC was initially activated by manual operation just after the reactor scram at Unit 2, but it automatically stopped with the L-8 signal. Then this on-off operation was repeated until the SBO. The Unit 2 DC power supply system was also submerged by the sea water and its function was lost at the SBO occurrence. Since the RCIC had been working at the SBO occurrence, the RCIC valve might have been kept open as it was at the total loss of power, and it continued to work for about 66 hours (until 9:00 on March 14) even after the total loss of power, without receiving the L-8 signal to stop it.

When the RCIC continued to work without L-8 signal, the water level in the RPV must continue to rise to the steam extraction line. Thus, steam to drive the RCIC turbine must include some water. The deterioration of the RCIC turbine performance under such two-phase flow condition was assumed to be proportional to the energy of inlet fluid to the turbine in the present analysis. Since the energy of inlet fluid, or two-phase flow, must be much less than one under the design condition, steam single-phase flow, the water mass flow rate injected into the core by the RCIC pump became very smaller than one under the design condition [8].

At Unit 3, the RCIC and the HPCI had worked under control by the DC power supply system. The RCIC had been first working under the design condition, or under the steam single-phase flow, and then the HPCI was automatically activated after the termination of the RCIC. Since the HPCI had very larger capacity to remove decay heat, the RPV pressure decreased rapidly after its activation to almost the lowest design limit, around 1 MPa. Such low pressure continued until a reactor personnel switched off the HPCI (see Figure 13 in the section 5.3). The HPCI turbine performance under such low pressure condition was calculated based on the energy balance similar to the assumptions applied for the Unit 2 RCIC.

3. THE SAMPSON CODE

SAMPSON was designed as a large-scale simulation system of inter-connected hierarchical modules covering a wide spectrum of scenarios ranging from normal operation to severe accident conditions [9]. Figure 5 shows the running sequence of SAMPSON modules.

The analysis control module (ACM) does not include physical models, but calls and terminates analysis modules as appropriate with respect to time in the event and physical location. The outline of SAMPSON is described in Ref. [1]. The major features of SAMPSON are as follows.

(1) Minimum use of empirical correlations to eliminate tuning parameters as much as possible.
(2) Maximum use of mechanistic models and theoretically-based equations.
(3) Validation by a wide range of analyses with separate effect and integral tests, mainly through participation in OECD/NEA projects.

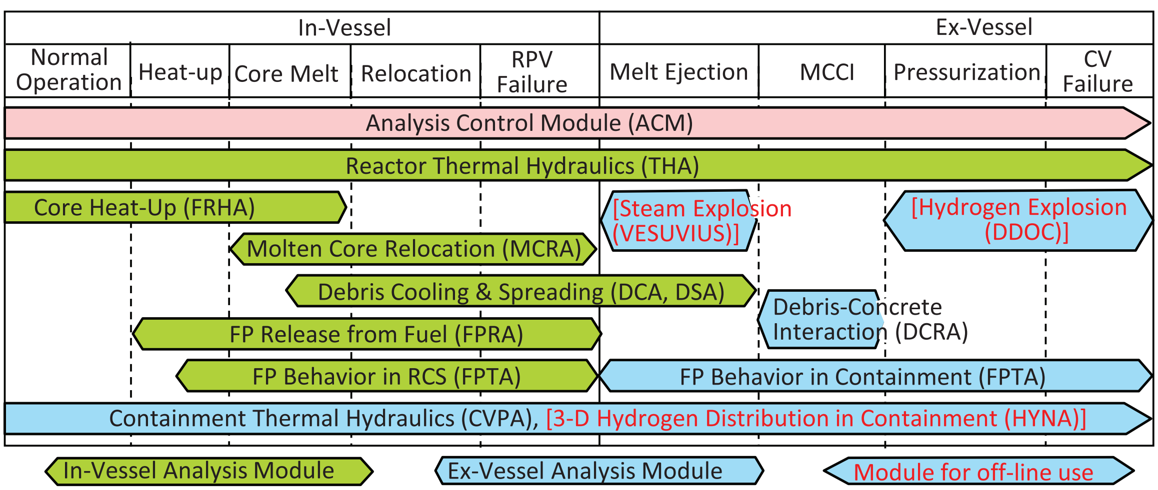

Figure 5 Running Sequence of SAMPSON modules

4. ANALUYSIS CONDITIONS

The core was divided into 8 channels and 13 axial nodes with r-z 2-D coordinates; 4 channels for fuel region and 4 channels for control rod/bypass region for every Unit, as shown in Figure 6. Ten of the 13 axial nodes were allocated for the fuel region. The exact dimensions of the reactor geometry [10] were reflected in the input data. The decay heat generation and its spatial distribution in the core were given by TEPCO [10].

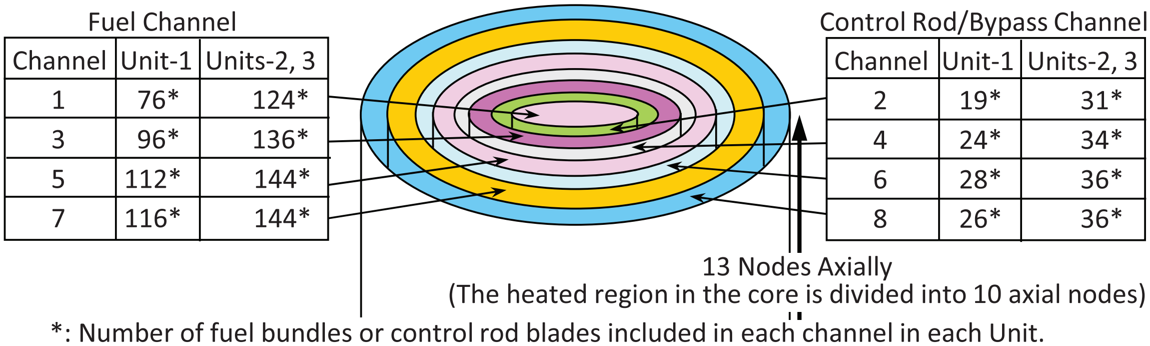

Figure 6 Node division

The melt of the constituent materials was assessed for its own melting point in the analysis. Since the eutectic reactions were also considered, some materials were considered to melt at lower temperature than the melting point of the pure materials. The melting temperatures defined in SAMPSON as default values are summarized in Table 2. The fuel cladding burst was calculated based on its temperature (or, the strength of the cladding material) and the difference between the cladding stresses inside and outside. The detailed temperature distribution in the RPV wall in the lower plenum region was three-dimensionally calculated. The break of the RPV bottom was evaluated by the melt of the SRM/IRM guide tubes in the bottom of the lower plenum defined in the section 2.2, or by the RPV wall break defined by the Larson-Miller creep rupture criterion, or by the melt of the RPV wall itself, whichever occurred first.

Table 2 Melting temperatures

3,113 K	Melting temperature of UO_2
2,980 K	Melting temperature of zirconium oxide
2,473 K	Eutectic temperature of UO_2+Zr
2,106 K	Melting temperature of Zircalloy
1,839 K	Melting temperature of iron oxide
1,671 K	Melting temperature of steel
1,500 K	Eutectic temperature of B_4C+steel

5. ANALYSIS RESULTS

5.1 Unit 1

The calculated time transients of the RPV pressure, the RPV water level, and the peak fuel temperature in the channel-1 (the most central channel in the core) are shown in Figures 7, 8, and 9 respectively. The peak fuel temperature shown in Figure 9 means the peak cladding temperature when the fuel cladding in the calculation cell stays there as the solid phase, or the peak surface temperature of fuel pellets after the fuel cladding melts and flowed out from the calculation cell. In the current modelling of SAMPSON, the thinner UO_2 pellets still stay there in the calculation cell even after the consumption of all zircaloy for the eutectic reaction with UO_2 and for Zr-H2O reaction, and the temperature of thinned UO_2 pellets can increase until they reach to the melting temperature of UO_2 itself. In other words, the breakdown of UO_2 pellets themselves is not considered.

The RPV pressure was continuously recorded on a chart until the total loss of power. After that, the measurements were manually and intermittently made by reactor personnel using portable devices powered by batteries. The first measurement was 7.0 MPa at 20:07 on March 11 and the second was 0.9 MPa at 02:45 on March 12. The exact timing of the RPV depressurization was not measured, but the RPV depressurization occurred sometime between them.

<u>Initial phase until the tsunami damage:</u>
Since the IC had enough capacity to remove decay heat, the RPV pressure decreased during the IC

operation periods. Once the IC stopped, the RPV pressure increased because of no decay heat removal. This pressure fluctuation was repeated during the intermittent IC operation period. Since the maximum RPV pressure was lower than the SRV activation pressure, the SRV did not open resulting in no coolant release. Thus, the collapsed water level in the RPV was kept constant during the intermittent IC operation periods. The analysis reproduced this pressure fluctuation until the total loss of power due to the tsunami.

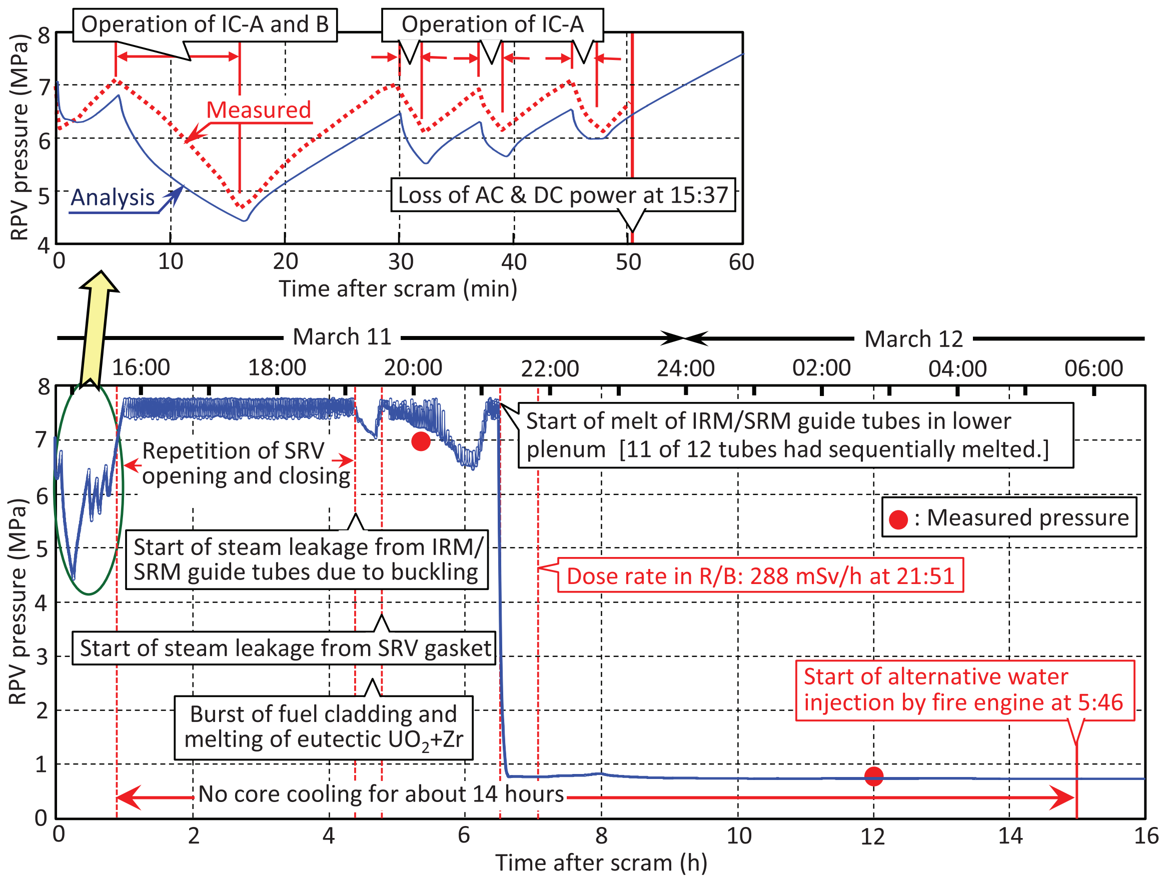

Figure 7 RPV pressure transient (Unit-1)

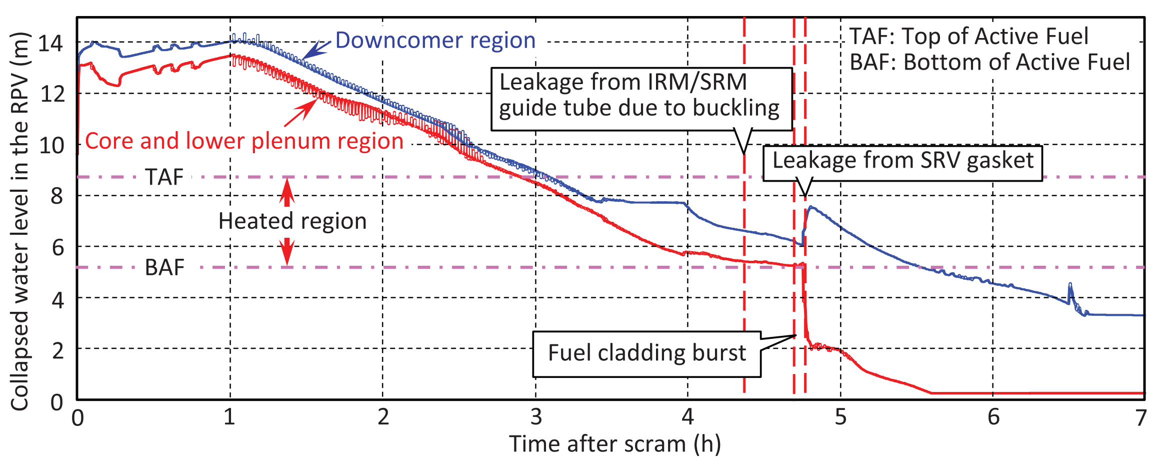

Figure 8 Collapsed water level in the RPV (Unit-1)

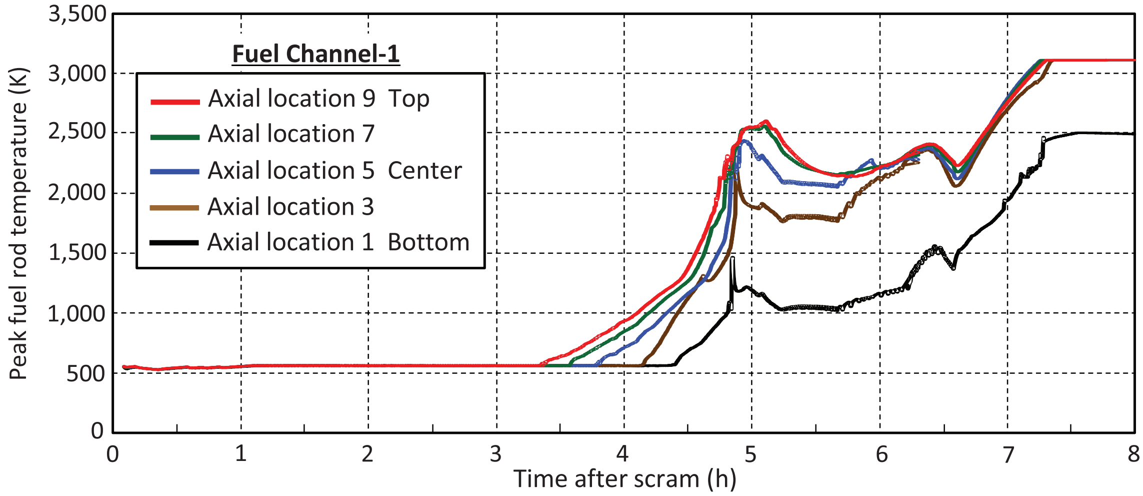

Figure 9 Transient of peak temperature at fuel channel-1 (Unit-1)

Repetition of SRV opening and closing:
After the total loss of power, the ICs did not work at all resulting in increase of the RPV pressure. When the RPV pressure reached the SRV activation pressure, the SRV automatically opened resulting in release of steam into the suppression pool, which then made the RPV pressure decrease, followed by automatic closure of the SRV. Thus the SRV had repeated its opening and closing for about 3.5 hours. During this repetition, the collapsed water level in the RPV went on to decrease by steam release from the SRV.

Core uncovery and heat-up:
After the collapsed water level reached the top of active fuel, the fuel went on to overheat. The fuel temperature increasing rate became faster with higher temperature because of the Zr-H_2O exothermic reaction. During this overheating process, the SRV had intermittently worked, and the RPV pressure stayed around 7.5 MPa due to steam generation by decay heat. At 4.4 h after scram, the steam leakage from the buckling portion of the SRM/IRM guide tubes started. And soon after that additional steam leakage from the SRV gasket started at 4.7 h after scram. Then, the analysis showed gradual depressurization by steam leakage from both the buckling portion of the IRM/SRM guide tubes and from the SRV gasket.

Core melt:
The eutectic B_4C (control rod material) and steel reacted, resulting in the initiation of melting at about 4.5 h after scram when the collapsed water level was getting closer to the bottom of active fuel (BAF). The fuel cladding burst occurred at 4.7 h after scram (19:28 on March 11), and soon after that melting of the eutectic compound of UO2 and Zr started, resulting in the release of large amount of fission products through the RPV leakage path to the primary containment vessel (PCV). Since there might be gas leakage from the PCV to the reactor building (RB), the dose rate in the RB was thought to increase. This was consistent with the high dose rate measurement at 7.1 h after scram (21:51 on March 11). Since the water still remained in the lower plenum when the eutectic UO2 and Zr started to melt at the temperature of 2,473 K and to fall into the lower plenum, a violent steam generation occurred there resulting in a temporal RPV pressure increase, which was indicated in Figure 7.

Damage to RPV bottom wall:
After all the water in the RPV was vaporized, the thin and dry guide tubes of SRMs/IRMs started melting since the molten eutectic compound of UO2 and Zr with the temperature of 2,473 K had continuously fallen down and accumulated in the lower plenum, while the RPV pressure was kept at around 7.5 MPa. Finally 11 of 12 SRM/IRM guide tubes had sequentially melted at their bases of the RPV bottom wall at

about 6.5 h after scram. The melt materials in the lower plenum were pushed out onto the pedestal floor through the melt portions of the SRM/IRM guide tubes, resulting in rapid RPV pressure decrease. The core melt and its fall down onto the pedestal floor started considerably earlier than the alternative water injection by the fire truck at 15 h after scram.

Amount of melts:
At the time of alternative water injection by the fire truck, 15 h after scram, 77% of the original total mass in the core region and 69% of the original total mass of UO_2 had already melted; all the corium fell down onto the pedestal floor in the drywell through the melt portions of IRM/SRM guide tubes in the lower plenum. The core meltdown continued gradually even after the alternative water injection by the fire truck, since the water mass flow rate into the core was too small to remove decay heat due to the branch flow. Finally at 100 h after scram, all the UO_2 had melted and fell down onto the pedestal floor.

5.2 Unit 2

Figure 10 shows the calculated RPV pressure transient at Unit 2. Before the tsunami hit, the RCIC intermittently worked at the rated conditions. Since the RCIC had enough capacity for decay heat removal, the RPV pressure decreased during its operational period. Two minutes before the tsunami hit, the RCIC had re-started and the valves on the RCIC line might have been kept open at the total loss of power; the RCIC continued to work for about 66 hours after scram. After the total loss of power, no high water level signal was sent to stop the RCIC. Therefore, the water level in the RPV increased by the RCIC water injection into the core resulting in inflow of two-phase fluid into the RCIC steam turbine. Even with this deteriorated turbine performance, the RCIC pump continued to inject fresh water into the core. Thus the water level stayed high and decay heat was removed well until the termination of the RCIC. After the termination of the RCIC, there was no intentional decay heat removal for a while, resulting in a pressure increase. The pressure was kept around 7 MPa by discharging steam through the intermittent work of SRVs. Due to this steam discharge, the collapsed water level in the RPV gradually decreased. In order to enable alternative water injection by a fire truck, reactor personnel activated a SRV to open at 75.3 h after scram (18:02 on March 14). Then the RPV pressure rapidly decreased. Even after this depressurization, the RPV pressure showed small peaks. The reactor personnel might consider that these pressure increases were due to the SRV closure. Therefore the reactor personnel tried to activate another SRV to open several times. This action by the reactor personnel was archived on the operators records [7]. These repetitions of SRV opening and closing were applied as the boundary conditions of the present analysis.

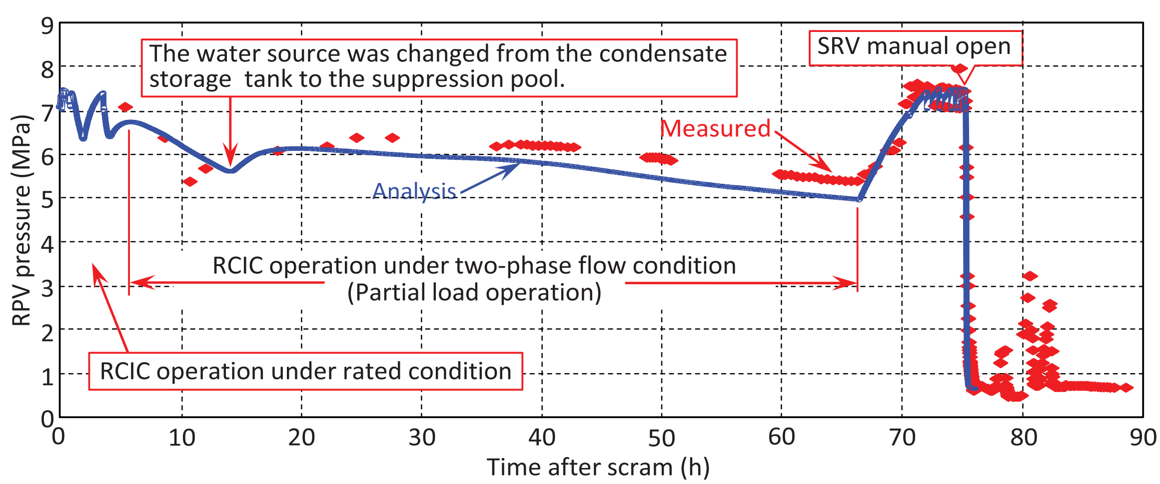

Figure 10 RPV pressure transient (Unit 2)

The collapsed water level rapidly decreased after the first SRV manual opening for depressurization, as shown in Figure 11. The site manager decided to manually open the SRVs for depressurization to enable alternative water injection by a fire truck. Since the water level had already started to decrease gradually during the SRVs on-off repetitions, the water level dropped to the TAF soon after the manual opening of the SRV, and it reached the BAF at 19:10 on March 14. The sea water injection by the fire truck started at 77.1 h after scrum (19:54 on March 14), when the decay heat generation was 7.7 MW. At Unit 2, the alternative water from the fire truck was injected into the bypass region where the jet pumps were located. After the bypass region was first filled up by the water, the lower plenum was filled up and then the core was re-flooded from the bottom. During this time delay, the fuel overheated and melted.

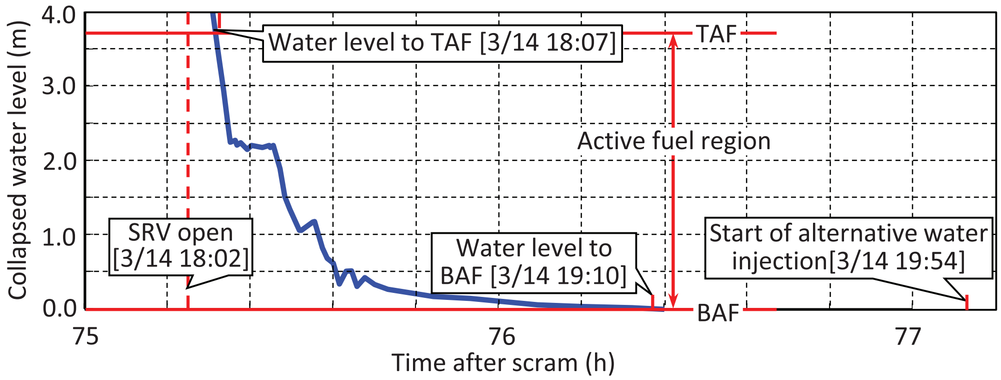

Figure 11 Collapsed water level in the RPV (Unit 2)

The deterioration of the SRV gasket and buckling of the SRM/IRM guide tubes at Unit 2 had little influence on the accident progression, since the core water level and the RPV pressure decreased rapidly due to manual opening of the SRV.

Figure 12 shows the transient of the peak fuel temperature in the channel-3 (the second central channel in the core).

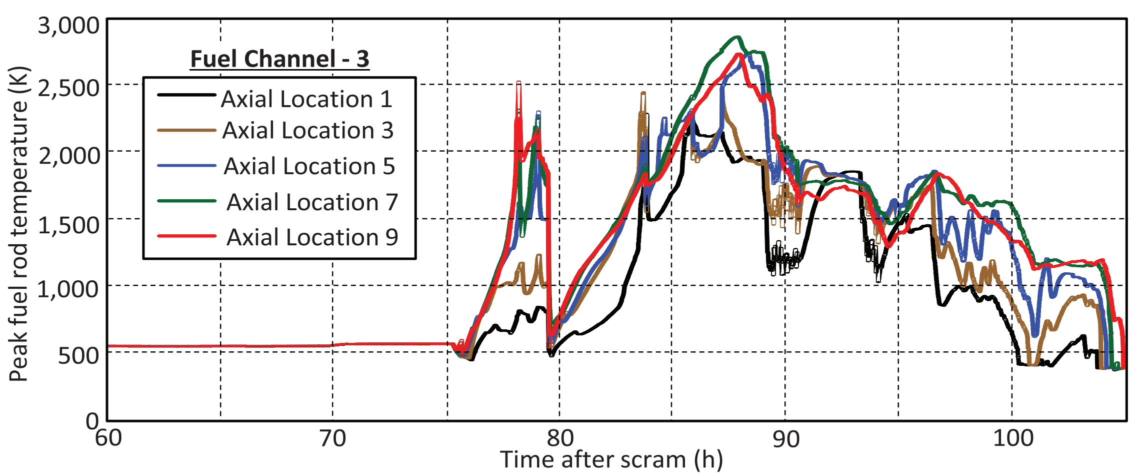

Figure 12 Transient of peak temperature at fuel channel-1 (Unit 2)

After the first SRV manual opening, the core temperature rapidly increased because of water level decrease. A part of the fuels reached higher temperature than 2,500 K, resulting in generation of molten corium of eutectic compound of UO_2+Zr. The generated high temperature molten corium fell down into the lower plenum where water still remained, resulting in violent steam generation. Due to this steam generation, the fuel was temporally and rapidly cooled down. This was considered to be the cause of rapid temperature decrease at just before 80 h after scram.

Finally at 105 h after scram when the core was re-flooded by the alternative water injection, 25% of the original total mass in the core region had melted; all the corium remained in the RPV lower plenum. The alternative water injection prevented RPV bottom break and further core meltdown.

In the recent TEPCO investigation, it was said that there was possibility that the SRV was kept open after its manual opening and the small RPV pressure peaks were not due to the SRV closure but due to violent steam generation in the lower plenum where molten corium fell down into the residual water. It was not confirmed yet and it must affect the accident progression in the analysis. This is one of future subjects.

5.3 Unit 3

The calculated time transients of the RPV pressure, the RPV water level, and the peak fuel temperature in the channel-3 (the second central channel in the core) are shown in Figures 13, 14, and 15 respectively. Since the DC power facility was survived, the RCIC and then the HPCI were well operated. The RPV pressure was kept around 7 MPa during RCIC operation. After the RCIC lost its function to inject water at 20.8 h after scrum (11:36 on March 12), the HPCI automatically started water injection at 21.8 h after scram (12:35 on March 12). Since the HPCI had more cooling capacity than the RCIC, after the automatic activation of the HPCI, the RPV pressure decreased rapidly to around 1 MPa which was almost the design lower limit. And since steam energy at around 1 MPa, which flowed into the HPCI turbine, would be considerably lower than that at around 7 MPa, the HPCI performance must deteriorate resulting in decrease of water injection flow rate into the core. Therefore, the water level decreased at latter half of the HPCI operation under around 1 MPa. After the manual switch-off of the HPCI by reactor personnel at 36 h after scrum (02:42 on March 13), the RPV pressure increased resulting in the repetition of the SRV opening and closing. Then the rapid depressurization occurred due to the unintentional activation of the automatic depressurization system (ADS) [7]. The fire truck was smoothly activated at 17 minutes just after the depressurization, when the decay heat generation was 9.9 MW, the largest among the Units. There was a branch flow of the discharged water from the fire truck and moreover the water was injected not directly into the core, but into the downcomer region. Therefore there was considerable time delay for effective core cooling by the bottom re-flooding of the alternative water injection by the fire truck.

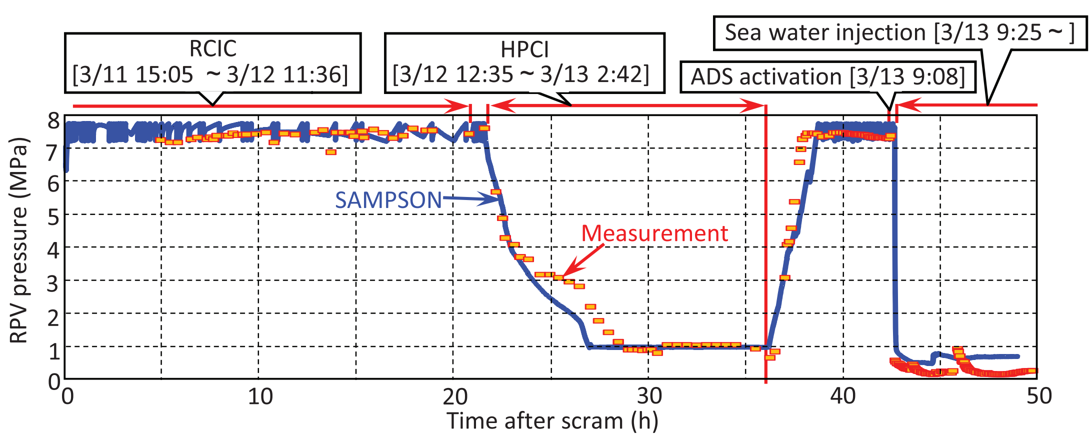

Figure 13 RPV pressure transient (Unit 3)

Similarly to Unit 2, the steam leakage from the SRV gasket and from the bucking portion of the SRM/IRM guide tubes at Unit 3 had little influence on the accident progression, since the core water level and the RPV pressure decreased rapidly due to ADS activation.

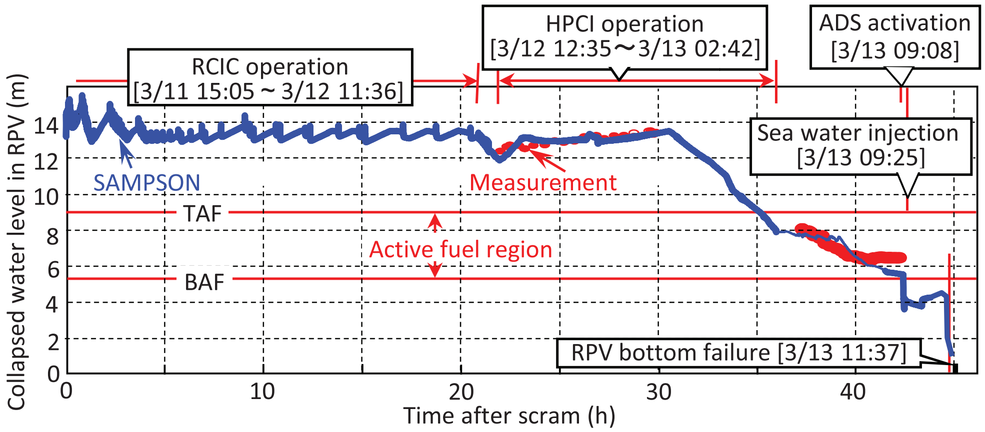

Figure 14 Collapsed water level in the RPV (Unit 3)

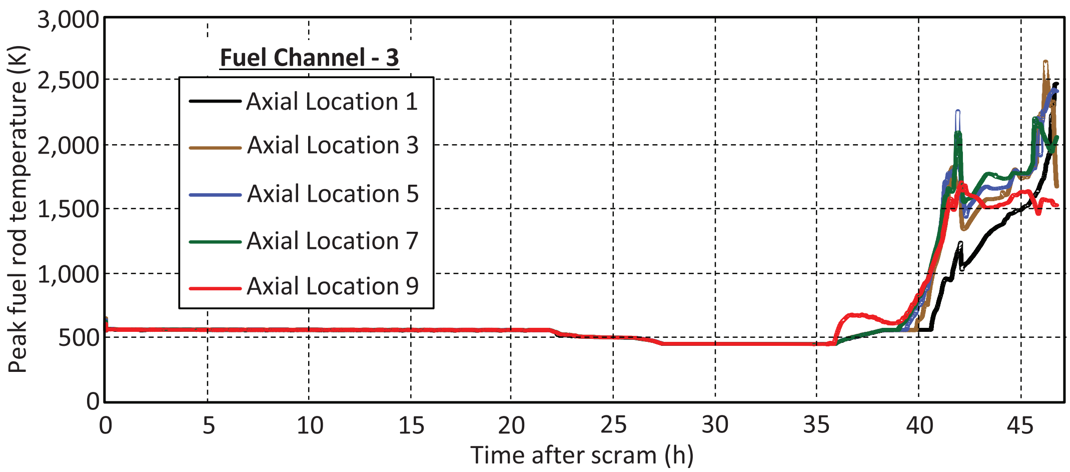

Figure 15 Transient of peak temperature at fuel channel-1 (Unit 3)

Since the water level had already reached BAF at the time of the ADS activation, the core meltdown started earlier than the ADS activation. Finally at 47 h after scram, 80% of the original total mass in the core region had been relocated; all the corium fell down onto the pedestal floor in the drywell through the melt portions of IRM/SRM guide tubes in the lower plenum.

There was another possibility that the HPCI had lost its function to inject water when the RPV pressure decreased to its design lower limit, earlier than the time of its switch-off by reactor personnel. In this case, 100% of the original total mass in the core region had melted and fell down onto the pedestal floor. This is still uncertain and future subject.

6. CONCLUSIONS

The accident progression of the Fukushima Daiichi NPP was analyzed by the severe accident analysis code SAMPSON. The original SAMPSON was improved by adding new modellings for the phenomena specific to this plant: (1) deterioration of SRV gaskets and buckling of in-core-monitor guide tubes, which caused the early steam leakage from the core into the drywell; (2) melting of the in-core-monitor guide tubes at the RPV bottom wall; (3) branch flow of the alternative water injection by the fire truck; (4) consideration of the continuous drainage pathways of the debris; and (5) performance degradation of RCIC and HPCI. The analysis results showed the following.

(1) Unit 1

At 15 h after scram when the water injection by the fire pump started, 77% of the original total mass in the core region had already melted; all the corium fell down onto the pedestal floor in the drywell. The core meltdown continued gradually even after the alternative water injection by the fire truck because of insufficient water mass flow rate into the core due to the branch flow. Finally at 100 h after scram, all the UO_2 had melted and fell down onto the pedestal floor.

(2) Unit 2

At 105 h after scram when the core was re-flooded by the alternative water injection, 25% of the original total mass in the core region had melted; all the corium remained in the RPV lower plenum. The alternative water injection prevented RPV bottom break and further core meltdown.

(3) Unit 3

At 47 h after scram, 80% of the original total mass in the core region had melted; all the corium fell down onto the pedestal floor in the drywell. However, when supposing earlier loss of function of water injection by HPCI than the manual switch-off, the melt amount would become 100%. This is still uncertain.

There may still be uncertainties in the analyzed accident progressions of Units 2, 3; those are uncertainties of SRV working at Unit 2 and of timing of water injection termination by the HPCI at Unit 3.

The total accident progression analyses of Fukushima Daiichi NPP Units 1-3 including in- and ex-vessel phenomena, and also source term evaluations are the future issues.

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