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SEVERE ACCIDENT ISSUES RAISED BY THE FUKUSHIMA ACCIDENT AND IMPROVEMENTS SUGGESTED

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This paper revisits the Fukushima accident to draw lessons in the aspect of nuclear safety considering the fact that the Fukushima accident resulted in core damage for three nuclear power plants simultaneously and that there is a high possibility of a failure of the integrity of reactor vessel and primary containment vessel.

A brief review on the accident progression at Fukushima nuclear power plants is discussed to highlight the nature and characteristic of the event. As the severe accident management measures at the Fukushima Daiich nuclear power plants seem to be not fully effective, limitations of current severe accident management strategy are discussed to identify the areas for the potential improvements including core cooling strategy, containment venting, hydrogen control, depressurization of primary system, and proper indication of event progression. The gap between the Fukushima accident event progression and current understanding of severe accident phenomenology including the core damage, reactor vessel failure, containment failure, and hydrogen explosion are discussed.

Adequacy of current safety goals are also discussed in view of the socio-economic impact of the Fukushima accident. As a conclusion, it is suggested that an investigation on a coherent integrated safety principle for the severe accident and development of innovative mitigation features is necessary for robust and resilient nuclear power system.

KEYWORDS : Fukushima Accident, Severe Accident, Severe Accident Management, Severe Accident Phenomenology, Boling Water Reactor

1. NATURE OF THE FUKUSHIMA ACCIDENT

The accidents at Fukushima Daiichi nuclear power plants are striking as they not only resulted in simultaneous core damage in multiple units, but also there was a high possibility of failure of the reactor vessels and primary containment vessels in all three reactors. Though the radiological release is estimated to be about 10% of the Chernobyl accident [1, 2], the severity of the accident in terms of scale and number of units involved is unprecedented. The accident was classified as International Nuclear Events Scale (INES) level 7 accident [1].

The accident progression, including the cause of the accident, the response of the reactor and safety system, recovery actions, and core damage progression leading to a release of radioactive material, were investigated and reported by the Japanese Government [1], TEPCO [2] and international experts [3]. However, the status of the damaged reactor vessel, and damage to the primary containment vessel are still under investigation.

Though the occurrence of severe accidents were evidenced in the Three Mile Island (TMI) and Chernobyl accident, the measures for the prevention and mitigation of a severe accident were not strictly regulated. In most countries, severe accident prevention and mitigation measures were recommended only for new builds, as voluntary actions to enhance the safety, while provision of severe accident management guidelines were recommended for operating reactors. It is stated in reference 1 that "While the Japanese National Government recognized that further safety regulations were unnecessary as the safety of nuclear power plant in Japan was fully ensured by the present safety measures, it recommended that electric utilities should perform self-disciplined safety efforts in order to reduce a risk of accident and to further enhance safety."

The severe accidents at Fukushima Daiichi nuclear power plants happened unexpectedly. The event occurred due to a combination of an earthquake and tsunami in an unprecedented scale. The question is "Was it possible to predict and be prepared for this kind of accident?" Certainly, it seems to not be.

Though external events are considered in the design of the nuclear power plant, as recommended in safety guide IAEA-NS-G-1.5 [4], the Fukushima accident suggests that the very low probability of extreme external events can be overlooked, which can lead to catastrophic consequences. Different perspectives of external hazards, such as an impact of simultaneous occurrence of external events, the need for the provision of a long term electricity backup capacity, and a potential impact of terror, have to be investigated. The Fukushima accident is like a "Black Swan" [5], as it lay outside the realm of regular expectations. It came with catastrophic consequences and we were able to explain it only after the fact. Therefore, it might be wise to focus on how we can be prepared for this kind of severe accident in the future, rather than focus on the reasons for this particular accident happening.

The nuclear industry tended to be confident that nuclear power plants were safe, and there was very little chance of severe accidents like TMI or Chernobyl. This overconfidence could be one of the reasons why we were not able to predict the Fukushima accident, and why the defense-in-depth implemented, including the support system and emergency preparedness in the plant, was not robust enough to avoid the substantial release of radioactive material.

The nature of a low probability severe accident has often led to quite different views for the implementation of preventive and mitigation features among countries. The range of views has been wide, between pessimistic and optimistic. The gap should be narrowed to be properly prepared for the highly improbable event of a severe accident .

2. ACCIDENT PROGRESSION AT FUKUSHIMA DAIICH NUCLEAR POWER PLANTS

This section provides an overview of the chronology from the occurrence of the accident to the emergency measures taken at Fukushima Daiichi Nuclear Power station. In addition, highlights of the event progression, including plant response, operator recovery and unresolved issues, are discussed.

2.1 Plant Configuration and Event Progression for Each Unit

The event progression and plant specification discussed here are excerpts taken from Reference 1, 2 and 3. Major design parameters for the Fukushima Daiichi Units 1 through 4 are summarized in Table 1.

The earthquake which occurred at 14:46 on March 11, 2011 brought Fukushima Daiichi Units 1 through 3, which were in operation, to a reactor trip, due to the high earthquake acceleration. Unit 4 was under outage for periodic inspection when the earthquake occurred. All fuel had been removed from the reactor and transferred to the Spent Fuel Pool (SFP). Units 5 and 6 were under outage for periodic inspection, with all fuel in the reactors and all control rods inserted.

After the automatic shutdown of the reactors, the station power supply was switched to offsite power. However, the power plants were unable to receive electricity from the offsite power transmission lines, because some of the transmission towers had collapsed due to the earthquake. For this reason, the emergency Diesel Generators (DGs) for each Unit were automatically started to maintain cooling of the reactors and the spent fuel pools.

Later, all the emergency DGs, except at Unit 6, stopped, because their seawater cooling systems and metal-clad switchgears were submerged due to the tsunami that followed the earthquake. The result was that all AC power

Table 1. Major Design Parameters of Fukushima Daiichi Units 1 through 4

	Unit 1	Unit 2	Unit 3	Unit 4
Commercial Operation	1971	1974	1976	1978
Reactor Design	BWR-3	BWR-4	BWR-4	BWR-4
Rated Power (MWe)	460	784	784	784
Thermal Power (MWt)	1,380	2,381	2,381	2,381
Isolation Cooling System	IC	RCIC	RCIC	RCIC
ECCS Configuration	HPCI (1) ADS CS (4)	HPCI (1) ADS CS (2) LPCI (2)	HPCI (1) ADS CS (2) LPCI (2)	HPCI (1) ADS CS (2) LPCI (2)
Primary Containment Vessel	Mark-I	Mark-I	Mark-I	Mark-I

supply was lost at Units 1 to 4 . This led to Station Black Out (SBO) scenarios for these Units, which had been considered a hypothetical event [6], while emergency power to Units 5 and 6 was recovered. It has to be noted that following the earthquake, all the safety systems, including on-site emergency electrical power, operated properly as designed. It was the subsequent tsunami that caused the major damage.

At 15:42 on March 11, TEPCO determined that the plant condition fell under the category of specific initial events defined in Article 10 of the Act on Special Measures Concerning Nuclear Emergency Preparedness, and notified the national government, local governments, and other parties concerned.

At 16:36 on the same day, TEPCO found they were unable to monitor the water level in the reactors of Units 1 and 2, and determined that the conditions of these Units were "unable to inject water by the emergency core cooling system" as defined in Article 15 of the Nuclear Emergency Preparedness Act.

TEPCO opened the valve of the Isolation Condenser (IC) of Unit 1, and in an effort to maintain the functions of the IC, injected fresh water into its shell side. Later it was reported that the operator stopped operation of IC due to a rapid cool down of the reactor coolant system, according to the procedure, but was not able to activate the IC again [2].

Immediately after the tsunami, TEPCO could not confirm the operation of the Reactor Core Isolation Cooling (RCIC) system of Unit 2, but at 03:00 on March 12, they confirmed that it was operating properly. Unit 3 was cooled using its RCIC system, and as a result, the Primary Containment Vessel (PCV) pressure and water levels remained stable.

In order to recover the power supply, TEPCO took emergency measures, such as making arrangements for power supply vehicles, but the efforts were not effective due to damages caused by the earthquake and tsunami.

It was confirmed around 23:00 on March 11 that the radiation level in the turbine building of Unit 1 was increasing. In addition, at 00:49 on March 12, TEPCO confirmed that there was a possibility that the PCV pressure of Unit 1 had exceeded the maximum operating pressure. For this reason, the Minister of Economy, Trade and Industry ordered TEPCO to reduce the PCV pressure of Units 1 and 2. A brief summary of the events progression of each unit is summarized in Table 2.

At 05:46 on March 12, the company began alternative water injection (fresh water) into the reactor Pressure Vessel (RPV) for Unit 1, using fire engines. In addition, TEPCO began preparations for PCV venting because the PCV pressure was high, but the work ran into trouble because the radiation level in the reactor building was already high. It was around 14:30 on the same day that a

Table 2. A Brief Summary of Event Progression

	Unit 1	Unit 2	Unit 3
March 11th	14:46 Earthquake 15:42 SBO 16:36 Inability of water injection of the Emergency Core Cooling System (HPCI was not working)	14:46 Earthquake 15:42 SBO 16:36 Inability of water injection of the Emergency Core Cooling System (RCIC was working)	14:46 Earthquake 15:42 SBO
March 12th	01:20 Increase of PCV pressure 10:17 Start to vent 15:36 Sound of Explosion 20:20 Start to inject sea water water		(RCIC and HPCI was working)
March 13th		11:00 started to vent	05:10 Inability of water injection of the Emergency Core Cooling System 08:41 Start to vent 13:12 Seawater Injection to RPV
March 14th		13:25 Loss of reactor cooling function 16:34 Start Seawater Injection 22:50 Increase of PCV pressure	05:20 Started to vent 07:44 Increase of PCV pressure 11:01 Sound of Explosion
March 15th		00:02 Start to vent 06:10 Sound of Explosion 06:20 Possible damage of SC	

decrease in the PCV pressure level was actually confirmed. Subsequently, at 15:36 on the same day, what is thought to be a hydrogen explosion occurred in the upper part of the Unit 1 reactor building.

Meanwhile, the RCIC system of Unit 3 stopped at 11:36 on March 12, but later, the HPCI system was automatically activated, which continued to maintain the water level in the reactor at a certain level. It was confirmed at 02:42 on March 13 that the HPCI system had stopped. After the HPCI system stopped, TEPCO performed wet venting to decrease the PCV pressure, and fire engines began alternative water injection (fresh water) into the reactor around 09:25 on March 13.

As the PCV pressure increased, PCV venting was performed several times. As a result, the PCV pressure decreased. Subsequently, at 11:01 on March 14, a hydrogen explosion occurred in the upper part of the reactor building.

At 13:25 on March 14, TEPCO determined that the RCIC system of Unit 2 had stopped because the reactor water level was decreasing, and began to reduce the RPV pressure and inject seawater into the reactor using fire-extinguishing system lines. The wet venting line configuration had been completed by 11:00 on March 13, but the PCV pressure exceeded the maximum operating pressure. At 06:00 on March 15, an impulsive sound was reported near the Suppression Chamber (SC) and the SC pressure decreased sharply. Later, it was corrected that the reported explosion sound could have been confused with the explosion at Unit 4 [2].

2.2 Observations on the Event Progression

The first observation is about the possibility of core damage in these three units, which can be indicated either by an increase in the pressure of PCV or a hydrogen explosion.

As hydrogen is generated by the reaction between the zirconium cladding and steam during core damage, a hydrogen explosion could be a direct indication of core damage.

The pressure increase in the PCV could be also an indication of core damage. Significant amounts of steam coming from the reactor vessel during the boil off process in the core creates a saturated condition within the SC, and the resulting pressurization causes the steam to be discharged to the PCV. Also, when a significant amount of hydrogen is generated from the damaged core, the hydrogen gas would accumulate in the SC. It would result in a pressurization of SC and hydrogen gas flow to the PCV.

The PCV pressure increased well above the design pressure in Unit 1, while the possibility of the SRVs opening was suspected. If the SRVs were not opened enough, there would be a possible leak in the reactor pressure vessel. The relocation of molten core to the bottom of the reactor vessel could result in a loss of integrity of the reactor vessel. The direct path from the reactor vessel to the PCVs would result in a pressurization of the PCV.

The second observation is about the reason for the difference in the core damage progression among three units, which was due to a difference in their core cooling function.

For Unit 1, the Isolation Condenser (IC) was used for core cooling. Configuration for the Isolation Condenser in Unit 1 is shown in Fig. 1. The steam generated from the core goes to the isolation condenser and the condensed liquid comes back to the reactor vessel, so it is a closed loop operation. While the steam is condensing, the water in the shell side of the condenser is heated up. The water should be refilled to maintain the cooling capability in the IC. The water inventory can last 8 hours before refill [1]. The IC can be operated without an external power supply, as it is a fully passive safety system driven by gravity. The battery power was necessary for the control of the system.

However, it was reported by TEPCO that the IC was not properly operated for Unit 1. It was suspected that the IC was stopped just after the tsunami, and was never reactivated [1, 2]. This fact led to an earlier core damage progression in Unit 1, compared to the other units.

For Units 2 and 3, an RCIC was used for core cooling. The typical arrangement for the RCIC for these units is shown in Fig. 2. The steam generated from the reactor vessel is fed to the RCIC turbine, which drives the RCIC pump. The steam exiting the RCIC turbine is discharged to the SC, while the suction of RCIC pump is taken from the condensate storage tank. The RCIC achieves two functions of inventory make up and decay heat removal until the cold water is depleted from the condensate storage tank. Then, the suction has to be switched to the SC. In the case of Unit 2, the suction was switched to the SC at 05:00 on March 12 (after 14.2 hrs) and the RCIC failed finally at 13:18 on March 14. When the water in the SC reaches a saturated condition and/or there is no battery power, the RCIC cooling function could be lost.

The RCIC functioned properly for Units 2 and 3 for a significant amount of time. The core cooling function was achieved by the operation of the RCIC. It is suggested that the RCIC functioned longer than the battery depletion time of 8 hours, due to a partial load operation with a two phase inflow [7].

In the case of Unit 3, both the HPCI and RCIC remained available, whereas the HPCI was not available for Units 1 and 2. The RCIC was operated until 11:36 on March 12 (after 20 hrs). It stopped unexpectedly. An hour after the RCIC tripped, the HPCI automatically started on a low-low reactor level signal and began to restore the reactor water level. Operation of the HPCI is quite similar to that of the RCIC, with an injection flow capacity about twice the size. The HPCI system tripped at 02:42 on March 13 (after 35.9 hrs). Thereafter, no cooling or injection was available, the reactor core was uncovered, and this led to core damage.

The third observation is about the hydrogen explosion that occurred after the venting operation. It is suspected

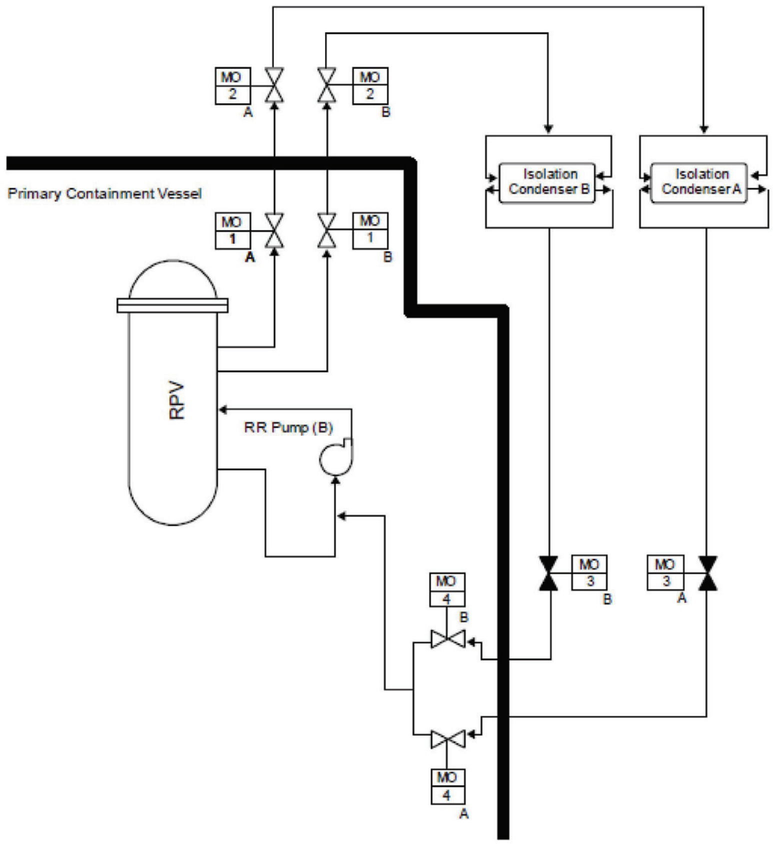

Fig. 1. Configuration of Isolation Condensers on Unit 1 [1]

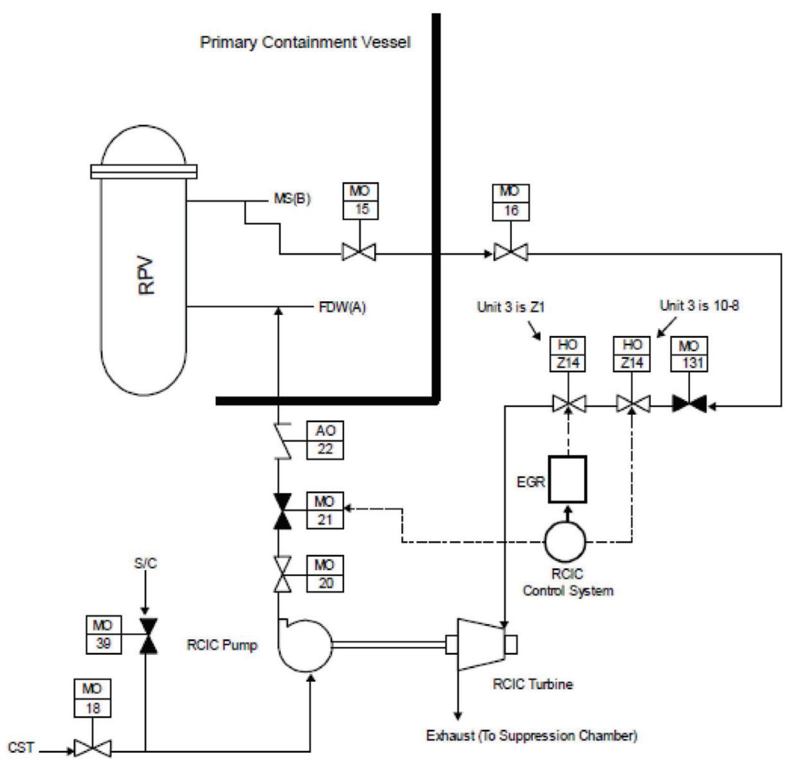

Fig. 2. Typical Arrangement for Reactor Core Isolation Cooling (RCIC) for Units 2 and 3 [1]

that the venting operation resulted in a significant leakage and subsequent accumulation of hydrogen in the reactor building. The hydrogen explosion occurred in the reactor building as it is under an air environment, as it would not be plausible in the wet-well, where an inert condition was maintained.

The leakage to the reactor building is suspected to have occurred either at the containment vessel flange and airlocks, or due to an opening of a flow path between the wet well and the reactor building.

The fourth observation is about the radiological releases. It was reported around 23:00 on March 11 that the radiation level in the turbine building of Unit 1 was increasing, which indicates that there was a leakage path from the reactor vessel to the outside. This could be due to a failure of the reactor vessel.

Recently, simulations of the accident progression were performed by JNES, TEPCO, and IAE as summarized in Fig. 3 [8]. Please note that except the MELCOR analysis by JNES in 2009, other analyses were made in 2011, after the Fukushima accident. Except for the MAAP calculation performed by TEPCO, it is estimated that the reactor vessel failure occurred early in the transient. This is very consistent with the abnormal pressure increase in the PCV, and it might have resulted in a significant release of hydrogen, steam, and fission product outside the reactor vessel.

The rapid increase of the dose rate on the operation floor of the reactor building at about 21:50 on March 11 is consistent with the failure of the reactor vessel, as this would certainly result in the release of volatile fission products.

The fifth observation is about the spent fuel pool. In Unit 4, all the fuel rods in the core had been moved to the Spent Fuel Pool (SFP). It was reported that at 04:08 on March 14, the cooling function of Unit 4's spent fuel pool was lost, and the water temperature rose to 84 ºC. At around 06:00 on March 15, a hydrogen explosion occurred in the reactor building, collapsing several floors, the western wall, and the wall along the stairs[1]. At first it was feared that the hydrogen explosion was due to the fuel cladding oxidation in the SFP and all the efforts were made to fill the SFP with water. If there was a leak in the spent fuel pool, there was a chance of dry out of the water inventory in the spent fuel pool, which could lead to a fuel failure and subsequent release of radioactive material.

However, it was confirmed on March 16 that the aforementioned pool was filled with water and that the fuel was not exposed. Therefore, it was not possible that hydrogen could have been produced by a zirconium-water reaction from the fuel in the SFP. An investigation into the cause of the explosion suggested that the vent flow, including hydrogen gas from Unit 3, could have flowed into Unit 4 through the stack junction [2]. Though there seems to be no significant damage to fuel in the spent fuel pool of Unit 4, proper indication of the spent fuel pool conditions including the level and temperature during accident conditions became a very important safety measure. Therefore, most regulators, including USNRC, require reliable indications of the water level in the spent fuel pool [9].

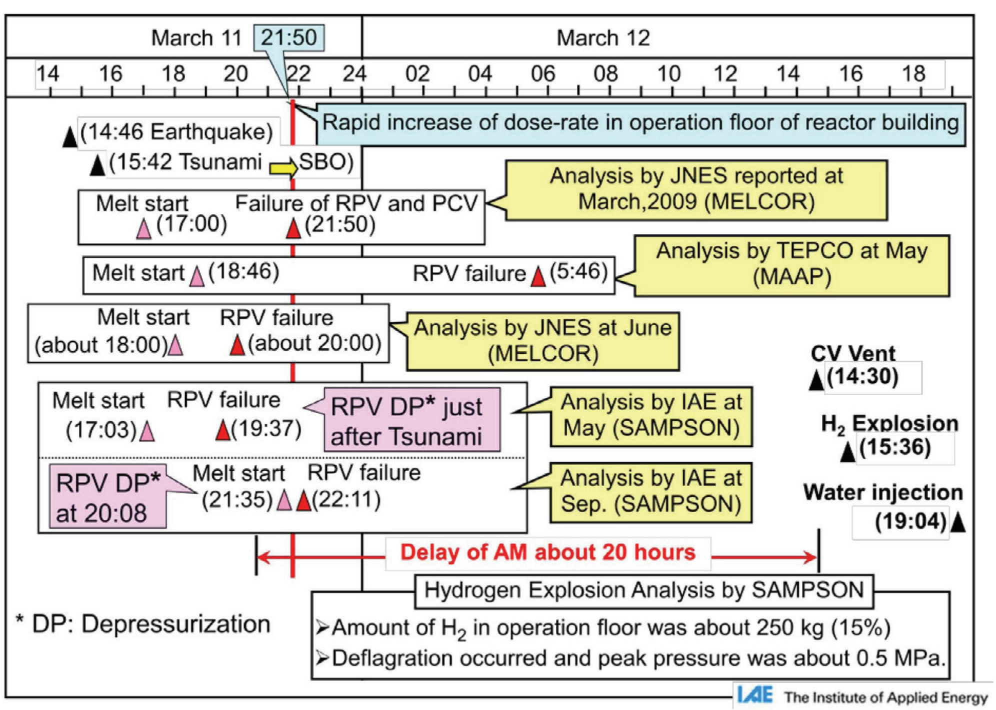

Fig. 3. Summary of Fukushima Daiichi Units 1 Event Progression [8]

3. SEVERE ACCIDENT PHENOMENOLOGY AND PHENOMENOLOGICAL UNCERTAINTIES

Though there are speculations on the event progression in the Fukushima nuclear power plant accident, there are no direct indications of the core melt progression, due to limited measurement data and very limited access to the inside of the RPV and PCV.

Here we would like to discuss some of the important phenomena, including the failure of the reactor vessel, the hydrogen explosion, and the potential for the re-criticality.

The fact that the reactor vessel did not fail in the case of the Three Mile Island (TMI) accident, where a significant amount of the core melted, gave us an expectation that a majority of severe accidents would not result in a failure of the reactor vessel. Investigations on TMI reported three major findings [10, 11]. (1) Failures of penetrations, such as in core instrumentation tubes, have been eliminated as potential failure mechanisms during the TMI-2 event. (2) Debris cooling occurred within the first 2 hours after debris relocation. (3) Enhanced debris cooling may have occurred via coolant traveling in channels within the debris and in channels between the debris and the vessel.

For a typical BWR, depressurization of the reactor pressure vessel is recommended in the severe accident management guideline [12, 13, 14] to enable alternate water injection into the core to arrest core damage progression. It would prevent reactor vessel failure. In the case of Unit 1, it is not clear whether the initial depressurization resulted from an operator action or due to a failure of the reactor vessel. Recent studies of the Fukushima accident [7] indicated that there was a chance of penetration failures, such source range monitors and intermediate range monitors, due to buckling at high temperature. Though the timing and reason of the reactor vessel failure is not clear, it is indicated there is a high possibility of RPV failure [1, 2].

Though the severe accident analysis computer code indicated a reactor vessel failure within 4 to 10 hours after the initiation of a severe accident for typical BWR and PWR [15, 16], the uncertainties in the modeling of the reactor vessel failures mechanism allowed a wide range of vessel failure time predictions, due to lack of full scale experimental evidence. It is recommended that the proper understanding and modeling failure of lower head penetrations during the severe accident be revisited.

It is also noted that the reactor vessel failure is an important issue during the phase of recovery and decontamination of a nuclear power plant. If there is no reactor vessel failure, like in the case of TMI, a closed loop cooling and decontamination of the damaged core would be possible. However, closed loop cooling and decontamination of the damaged core was not possible in the case of the Fukushima accident. The water injected into the reactor vessel leaked into the containment and resulted in a significant accumulation, and accidental release into the sea, of contaminated water, which became one of the difficult technical problems to solve during the recovery and decontamination efforts [17].

In a typical BWR, like the Fukushima Daiichi nuclear power plant Units 1, 2, and 3, hydrogen combustion and/or hydrogen explosion was not considered as a potential threat, because, as a preventive measure, the PCV is filled with nitrogen to suppress any hydrogen combustion. However, hydrogen combustion or explosion was observed in both Units1 and 3 unexpectedly. It is explained below.

A substantial amount of steam and hydrogen generation due to significant core damage resulted in the pressurization of the PCV. As the PCV failure could lead to a direct large release of fission product gases to the environment, the operators tried to vent the PCV in all three units. However, the manual venting operation was not successful due to the lack of power supply. Operators had a difficult time implementing the venting operation. After the venting operation, an unexpected leakage of hydrogen to the reactor building resulted in a hydrogen explosion in all three units with core damage. It was suggested that the gas mixture leaked though the PCV head flange could have been migrated to the reactor building could be the cause of the hydrogen explosion [18]. As the gas mixture, consisting of steam, combustible gases, and fission product, would be in contact with air in the reactor building, the steam condensation could make the mixture gas flammable.

A potential leakage from the PCV and the resultant migration of combustible gas into the reactor building had not been considered for the BWR. Therefore, proper mitigation measures need to be considered for the future. Either adding a hydrogen mitigation feature, such as passive autocatalytic re-combiner, or adding a preventive measure for the PCV leakage could be considered.

Since the failure of reactor vessel was suspected, it was further suspected that there could have been molten core concrete interaction. However, still there is no direct indication of this. The molten fuel debris relocated into the lower head filled with water could lead to a re-criticality if there is a source of neutrons [19]. Reference 19 investigated the possibility of re-criticality during the re-flood phase of a severe accident in a boiling water reactor (BWR). In the case of the Fukushima accident, the sea water had to be mixed with boron before injection into the reactor vessel to prevent re-criticality.

4. TOWARD A ROBUST AND RESILIENT SYSTEM

From the lessons we learned from the Fukushima accident, the plausible questions for the enhancement of nuclear reactor safety would be about how to set up coherent safety principles for a severe accident, and what kind of prevention and mitigation measures are necessary for a robust and resilient nuclear power system.

4.1 Severe Accident Management

Japan was prepared for a severe accident in terms of SAMG (Severe Accident Management Guideline). The Nuclear Safety Commission (NSC) of Japan issued a decision entitled "Accident Management (AM) as a Measure against Severe Accidents at Power Generating Light Water Reactors" in May 1992 [13]. The utilities completed implementation of AM to their NPPs by February 2002, and reported to the Nuclear and Industrial Safety Agency (NISA), which was the regulatory body of NPPs founded in Japan.

However, the SAMG for the Fukushima nuclear power plant turned out to be not able to stop the accident progression, such that the accident led to a substantial release of radioactive materials into the environment. It indicates that there are some weaknesses to be fixed in the current SAMG, which are discussed below.

Due to the total loss of AC and DC power supply, the main control room was blacked out, and critical safety parameters were not available for the operators in the early phase of the accident. In the current SAMG, it is assumed that indications for the important safety parameters are available for the operators. Operator decisions at the branching point are based on the observation of the trend of important safety parameters. However, these trends were hardly available, such that operators could not take proper SAMG actions in the case of the Fukushima accident. Therefore, the SAMGs should consider unavailability of important safety parameters.

As for the information on the critical safety parameters, such as reactor vessel water level, reactor coolant temperature, reactor pressure, and containment pressure, they were not available from the early phase of the event progression, due to a loss of all AC and DC power or they gave wrong indications because the sensor was not qualified for the severe accident environmental conditions. This fact suggests that the sensors, cables, and indicators for the critical safety parameters required for the SAMG actions should be qualified for the severe accident conditions, including temperature, pressure, and radiation. Will there be innovative techniques to improve the measurement? A candidate could be a remote power supply, though the transfer distance is yet to be increased for a practical application. Another option could be to develop fully analog, or hybrid analog /digital, measurements with low power consumption.

In the case of Unit 1, the isolation condenser, which was the last resort for decay heat removal, was not properly operated, while the RCIC functioned properly for a significant amount of time in the case of Units 2 and 3. It turned out that the operator was not fully trained for the operational characteristic of the isolation condenser, including the operational characteristics of the valves shown in Fig.1. The plant supervisor and operator did not notice the fact that the isolation condenser had stopped operation. This led to a delay in recovery actions, and subsequent core damage and reactor vessel failure much earlier than in Units 2 and 3. This suggests that the operator should have enough training for the operation of the major safety system during a beyond-design basis situation, like the one experienced in the Fukushima accident.

Depressurization of the reactor coolant system is a recommended SAMG action, as it not only prevents the damage due to the high pressure melt ejection and direct containment heating, but also enables injection into reactor coolant system at low pressure. However, it is very difficult to implement depressurization without a power supply, because the valves need to be operated at high pressure. Also, as the timely operation of depressurization is very critical for the timing of core damage and vessel failure [14, 16], the power supply to the SRVs should be secured. A mobile alternative AC power source or battery power could be prepared to enable activation of the depressurization, even in the event of SBO.

Because depressurization of the reactor vessel was not implemented in time, coolant injection into the reactor vessel was not successful during the early phase of the accident. There was also lack of available pumps and borated coolant supply. Sea water injection was implemented later with an addition of boron to prevent re-criticality during the late phase injection into RPV.

Containment venting had to be implemented because the primary containment pressure stayed well above the design pressure of the containment. Though, it had to be activated before substantial damage of the core, the venting was so late that there was a substantial accumulation of hydrogen in the suppression chamber and primary containment vessel. Even though the operator wanted to vent, it was a very difficult procedure to implement, because the he had to bring power to the vent valve which was in an inaccessible location [1]. If there was a provision for the manual operation driven by compressed air, or a compressed spring, it could have been much better. It is recommended that the mitigation features for a severe accident be fully passive and rely on natural forces for a reliable performance.

The opening of a direct release path from the suppression pool to the reactor building and to the environment, without filter, resulted in a substantial release of radioactive material to the site, and a high dose rate. Highly contaminated debris from the reactor building structure was spread widely due to a hydrogen explosion, and resulted in a contamination of the site in a wide range scale. This heavily slowed the recovery action.

Another important point is the leakage of highly contaminated water to the turbine building. This heavily hindered and delayed the operators' actions. There seemed to be a leak from the reactor vessel and primary containment, such that water injected into the reactor vessel was mixed with contaminated water and flowed out to the turbine building. The dose rate in the turbine building was so high that the operator was not able to stay there more than 15 minutes, which resulted in a heavily delayed recovery action.

The above situation is also true for the PWR. When water is flooded on top of the molten core relocated in the reactor cavity, it would certainly result in contamination of the containment. As it would result in a very limited access for operator action, this aspect should be properly handled in severe accident management procedures.

4.2 Safety Goals

The Japanese regulatory body, the Nuclear Safety Commission issued safety goal in terms of qualitative health objectives in 2003, such that individual fatality risks should not exceed 10^{-6}/year. Also, in 2006 the performance safety goals are specified as Core Damage Frequency (CDF) of 10-4/yr and Containment Failure Frequency (CFF) of 10-5/yr [20] for internal events.

For the effectiveness of the AM measures, Japanese electric utilities issued the results of Probabilistic Safety Assessments (PSAs) after the AM measures were established [13]. The results below clearly indicate that the performance safety goals set by the Japanese regulation are met as below.

CDF : 1.6 x 10^{-7} /yr and CFF : 1.2 x 10^{-8} /yr for BWR-4
CDF : 2.4 x 10^{-8} /yr and CFF : 5.5 x 10^{-9} /yr for BWR-5

The fact that no individuals suffered from prompt fatality because of the Fukushima accident is consistent with the performance safety goal. However, the radiological impacts to the people and environment, and property damage at the Fukushima accident exceed the tolerability of people and society. There could be two questions on the performance goal specified above: “Was the performance goal proper?” and “Why were those evaluations not able to predict the occurrence of the Fukushima accident?”

The first point could be the uncertainties associated with a PSA. A mean frequency number without uncertainty was estimated and taken as a criterion, though it has a big uncertainty. The importance of uncertainty could often be neglected and forgotten.

The second point could be that the external events, such as natural hazards, including earth quake and tsunami, fire, and terrorist attacks were treated as a probabilistic argument. If the probability was very low, it was not considered in the design basis for the nuclear power plant. For example, the effects of a tsunami of an unprecedented height and prolonged station blackout were not properly accounted for.

The values of a PSA, which can be summarized as a language for quantifying uncertainty, a structured view of plant dependencies and interactions, a rational integrated view of plant response in terms of consequences, their likelihood, and the responsible contributing factors, and a flexible tool for managing plant safety, are very useful and valid [21]. Then the question is how we can augment or develop an innovative PSA frame where we can incorporate unexpected events with high risk, such as those that occurred in the Fukushima accident. A deterministic safety analysis frame to prevent or mitigate the consequence of very low probability scenarios should be added to augment the PSA.

As the Fukushima accident resulted in a huge loss in property and wide spread contamination of the soil, air, and water, the above safety goals are no longer acceptable to society, especially for highly populated countries. The safety goals for limiting radiological releases need to be defined to complement current safety goals on quantitative health objectives, CDF and CFF.

Already in northern Europe, countries like Finland have regulations on the the radiological release during a severe accident, such that an atmospheric release of caesium-137 is limited to 100 TBq as specified in YVL 2.2 [22]. Recently, IAEA also proposed a specific safety requirement [23] for the design extension condition, which is stated as “The design shall be such that design extension conditions that could lead to significant radioactive releases are practically eliminated”. However, as there are big uncertainties in methodologies for evaluating the radiological consequences, further study is recommended before setting up a quantitative goal.

5. REMARK

The very low probability of a severe accident led to a wide gap between pessimistic views and optimistic views for the necessity of mitigation measures for the nuclear power plants. Each country has different philosophies and regulations for dealing with severe accidents. As a minimum, SAMGs were provided for most operating nuclear reactors in the world, to be prepared for a severe accident. However, the SAMG prepared for Fukushima was not able to stop the progression of the accident, such that it led to a substantial release of radioactive materials into the environment.

In the frame of the current SAMG, there is no analysis for the practical aspects of recovery actions in the worst case scenario. It was assumed that even during the severe accident progression there would be something available for a successful recovery action, the feasibility of which was not seriously investigated. There should be more reliable, tolerant, diverse measures based on the defense-in-depth concept, and implemented as a passive system. Also, diversity and redundancy for the indication of reactor status and availability of safety systems is required to assure that recovery actions are effective in the worst case scenarios.

As a summary, it is suggested that an investigation into the coherent integrated safety principles for a severe accident, and development of innovative mitigation features are necessary for a robust and resilient nuclear power system.

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NOMENCLATURE

ADS	Automatic Depressurization System
CDF	Core Damage Frequency
CFF	Containment Failure Frequency
CS	Core Spray System
ECCS	Emergency Core Cooling System
EDGs	Emergency Diesel Generators
HPCI	High Pressure Core Injection System
IC	Isolation Condenser,
INES	International Nuclear Events Scale
LPCI	Low Pressure Core Injection System
NCS	Nuclear Safety Commission
RCIC	Reactor Core Isolation Cooling System.
SAMG	Severe Accident Management Guideline
SC	Suppression Chamber
SFP	Spent Fuel Pool
SRV	Safety Relief Valves
TEPCO	Tokyo Electric Power Company
TMI	Three Mile Island
PCV	Primary Containment Vessel

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