Reinforced measures of severe accident prevention for restarting Japanese PWR plants after Fukushima accident

YOSHIKAWA Hidekazu[1]

1. Symbio Community Forum, C/o Research Institute of Applied Sciences, 49 Tanaka Ohi-Cho, Sakyo, Kyoto, 606-8201, Japan (yosikawa@kib.biglobe.ne.jp)

Abstract: In Japan, the national institution of nuclear safety regulation had been completely re-organized after Fukushima accident in March 2011. In this paper, the review on how the revision of nuclear safety standards of light water reactors (LWR) is firstly made after Fukushima accident by a newly established nuclear regulation authority. And then the process on what should be done by the nuclear power plant operator is introduced in order to pass the licensing examination to get the permission of restarting their plants. The essential efforts paid by an operators of pressurized water reactor (PWR) is to enhance the countermeasures against severe accident, for which they should strive for building up the ability to conduct on severe accident analysis more efficiently. In this respect, some ideas are proposed to improve the efficiency of the related works.

Keyword: severe accident prevention; Fukushima accident; safety regulation standard; severe accident analysis; PRA

1 Introduction

In Japan, the governmental regulation institutions of nuclear safety was completely reorganized in September 2012 as the result of Tokyo Electric Power Company's Fukushima Daiichi Nuclear Power Plant accident (hereafter Fukushima accident) which was caused by East Japan Great Earthquake in March 11, 2011, when as many as four units No.1 to 4 committed severe accidents with hydrogen explosion in the reactor buildings. As many as 200,000 people ca. 40km surrounding the plant site had to be evacuated and the land and sea much broader than 40 km circle were contaminated by extensive radioactive release.

In this paper, the author of this paper will introduce how the nuclear safety standards of light water reactors (LWRs) was revised after Fukushima accident by a newly established nuclear regulation authority, and what should be done by the nuclear power plant operator in order to pass the licensing examination to get the permission of restarting their plants. The essential efforts to be made by the PWR operators in Japan had been to strengthen the countermeasures against severe accident, for which they should consume a lot of "energy" (cost, manpower and time) to conduct on severe accident analysis. In this respect, the author also would like to propose some ideas to improve the efficiency of the related works.

The contents of this paper are organized as follows: First in Chapter 2, the overview of the revised safety standard will be introduced. In Chapter 3, the major point of the licensing examinations will be summarized. In Chapter 4, the process of the applicants' activities will be introduced on how to conduct on severe accident analysis in order to validate the effectiveness of their introduced severe accident countermeasures. In Chapter 5, the problems of the current activities by the applicants will be discussed and some proposals will be made to improve the conduction of severe accident analysis.

2 Safety standard for restarting LWRs and the examining process after Fukushima accident

2.1 What were at issues and altered from before

With regards to the governmental reform of nuclear safety regulation, many organizations with complicated responsibility allocation in the past had been unified into newly established Nuclear Regulatory Authority (NRA) in September 2012.

Received date: July 19, 2016
(Revised date: July 20, 2016)

This is an independent administrative committee in the Government to have the power to rule on nuclear safety issues independent from the Cabinet, and the secretarial works of this NRA are conducted by Nuclear Regulatory Agency (also abbreviated as NRA).

Afterwards the NRA had set to revise the nuclear safety standards for LWRs, and in July 2013 the NRA had officially announced the new safety licensing standard for LWRs together with the related national law amendments. The intention of this juridical action by NRA is to strengthen the abilities of LWR facilities against severe accidents caused by not only big natural disasters but also human-caused disasters. It is assumed that all the LWRs (both PWR and BWR) which had stopped operation after Fukushima accident should meet with the requirements set by the new regulation standard, if the operator would like to restart the plants.

The operators of both the PWR and BWR in Japan had set to prepare for applying the NRA for the restart of their plants, upon the announcement of the new regulation standard in July 2013. However, at the time of the announcement, there had not so detailed procedures and rules being established by both NRA and the applicants, on how the applicants prepare for the documents for NRA, while how to examine the document by NRA, *etc.*

Therefore although the existence of uncertainties in the procedural details, the examination of the licensee's documents had started in 2013 first for PWR operators, and later for BWRs. Lack of the enough number of officers for examining the applicants' document in NRA hindered the smooth and rapid process of licensing examination by NRA but any way in the middle of 2015 several PWRs had been approved to restart by NRA, and in December 2015, the first PWRs of Sendai Unit Nos. 1 and 2 of Kyushu Electric Power Company started commercial operation.

To sum up, Fig. 1 shows how the light water safety standard in Japan was strengthened by Fukushima accident.

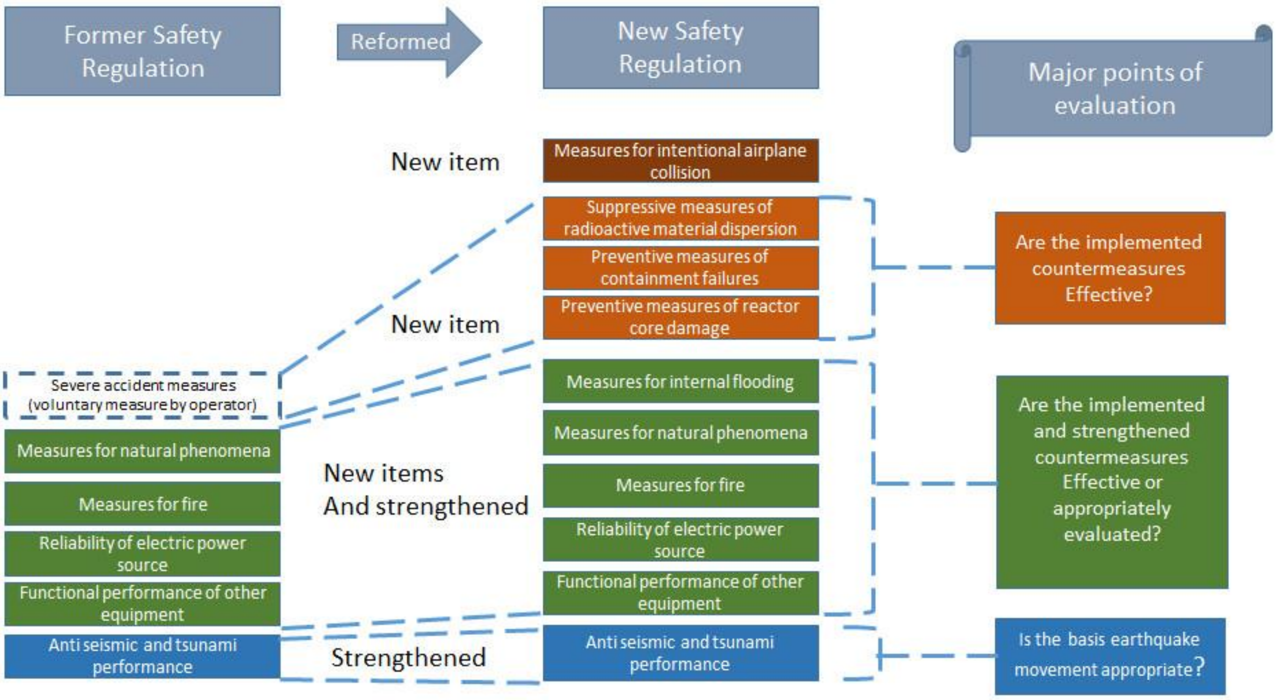

Fig. 1 Safety regulation prior to Fukushima accident and the new safety regulation standard with the major evaluation items after Fukushima accident in Japan.

Historically there had been many big earthquakes in Japan, and therefore the countermeasures against earthquake and tsunami had been included as the object of nuclear regulation. The governmental nuclear regulation had also ordered other restrictions to prevent from natural disaster originated things, fire protection, reliability maintenance of electric power, and other functional requirements for components. However, the measures against severe accident had been left to the operator's voluntary activities. This means that the severe accident has not been included in Design Basis Accident (DBA) of nuclear power plants.

However after Fukushima accident, both severe accident measures and anti-terrorist attacks were included in NRA's regulation and the new regulation standard has been overall more strengthened than before. The details of the regulatory requirements will be introduced in the subsequent sections.

2.2 New required functions after Fukushima accident and the examples of safety measures

There are three functions as below reinforced in the new regulation standard by NRA:

A. Anti-seismic and tsunami functions,
B. Functions maintained as design basis,
C. Necessary functions to cope with severe accident. (Severe accident measures).

There are 5 items for A, 6 items for B, and 20 items for C, so that as many as 31 items in total are requested. Those newly requested safety functions and the example countermeasures for A and B are shown in Tables 1 and 2, respectively.

Table 1 Anti-seismic and tsunami safety functions.

No.	New requested functions	Examples of countermeasures
A.1	Maintaining safety against attack of base tsunami	Establishing base tsunami. Install anti-tide banks and gates
A.2	Strengthening anti-seismic resistance for tsunami protection facilities	Maintaining strong seismic resistance for anti-tsunami banks and tsunami monitoring facilities.
A.3	Proving no existence of active fault layers until 400,000 years ago, if necessary.	Detailed field observation examination to prove no existence of active fault.
A.4	Three dimensional grasping of underground structure to check the base seismic motion.	Investigation by using driving vehicle to add seismic motion on the ground.
A.5	Safety-critical facilities should not be built on the ground that has apparent trace of active fault.	

Table 2. Design-basis safety functions.

No.	New requested functions	Examples of countermeasures
B.1	Plant safety should not be damaged by volcano eruption, tornado and external fires.	Conduct on related consequence evaluation of those effects, and if necessary, make repair works.
B.2	Plant safety should not be damaged by internal flooding.	Conduct on consequence evaluation of internal flooding, and if necessary, make repair works.
B.3	Plant safety should not be damaged by internal fire.	Conduct repair works to prevent from fires, to detect and extinguish fires, and to mitigate the consequence of fire.
B.4	Maintain high reliability of safety-important functions.	Multiplication of safety-important piping, *etc.*
B.5	Maintain high reliability of electrical systems.	Duplication of external power lines, switch yards, and emergency diesel generators. Maintain anti-seismic resistance of fuel tanks.
B.6	Physical protection of heat transport systems to the final heat sinks.	Physical protection of sea water pump.

On the other hand, severe accident countermeasures for C requested by NRA are so versatile and numerous, and they are classified into the following four ranges.
(1) Strengthen safety functions of reactor,
(2) Strengthen safety functions of containment vessel,(CV)
(3) Strengthen emergency support functions,
(4) Strengthen safety functions of site periphery.
Those four strengthened functions are listed in Table 3 to 6.

Table 3. Countermeasures against severe accident- Part (1) Enhanced safety functions in nuclear reactor.

No.	New requested functions	Examples of countermeasures
C.1	Reactor shutdown function.	Borated water injection facilities.
C.2	Cooling function of reactor coolant at high pressure state.	Preparation of batteries for valve operation necessary for starting reactor isolation cooling.
C.3	Depressurization function of pressure boundary of reactor coolant.	Preparation of batteries for valve operation for depressurization.
C.4	Cooling function of reactor coolant at low pressure state.	Preparation of stationary and portable water injection facilities.
C.5	Ultimate heat sink functions for preventing severe accident.	Car-loaded facility of heat sink capability.

Table 4. Countermeasures against severe accident-Part (2) Enhanced safety functions in reactor vessel.

No.	New requested functions	Examples of countermeasures
C.6	Cool, depressurize and diminish radioactivity in containment vessel.	Preparation of alternative water injection through containment spray system.
C.7	Prevent over-pressure rupture of containment vessel.	Installation of filter vent system from containment vessel. (PWR).
C.8	Cooling function of molten core dropped in lower part of containment vessel.	Water injection facility into lower part of containment vessel.
C.9	Prevention function of hydrogen explosion in containment	Preparation of hydrogen density control facility.
C.10	Prevention function of hydrogen explosion of reactor building. (BWR).	Preparation of hydrogen density control or exhausting facility and hydrogen density monitoring.
C.11	Cooling, shielding and maintaining sub-criticality of spent fuel pool.	Preparation of portable alternative water injection facility, Preparation of portable water spray facility.

Table 5. Countermeasures against severe accident-Part (3) Enhanced emergency safety support functions.

No.	New requested functions	Examples of countermeasures
C.12	Water support function.	Preparation of water sources. Transport root and transport machines.
C.13	Electricity support function.	Preparation of stationary and portable alternative current generators, Enhancement of stationary direct current generator. Preparation of portable direct current generator.
C.14	Control room function	Evaluation of radiation dose in reactor core damage condition.
C.15	Emergency response facility function.	Maintain anti-seismic and tsunami function. Evaluation of radiation dose. Preparation of necessary stocks and procurement.

Table 6. Countermeasures against severe accident-Part (4) Enhanced safety functions in site periphery.

No.	New requested functions	Examples of countermeasures
C.16	Instrumentation function	Preparation of estimating means of plant condition when the plant state exceeds the normal instrumentation systems.
C.17	Monitoring function.	Preparation of portable alternative monitoring facility.
C.18	Telecommunication and transmit function.	Preparation of telecommunication facility supplied by alternative power source.
C.19	Discharge restriction of radioactive materials outside of plant site.	Preparation of portable water discharge facility.
C.20	Water discharge function to extensively destroyed plant by large-scale natural disaster and by intentional attack of airplane by terrorists.	Distributed preparation of potable water injection facility power source, and water discharge facility, so that the effect by natural disaster and airplane attacks can be negated.

2.3 Items of regulatory examination

The safety of the applicant's plant will be first examined by NRA with respect to earthquake, ground structure and tsunami. As listed in Table 7, there are 7 items of earthquake issue, and 2 items for tsunami issue. Then, the accident prevention measures of the whole plant are examined from the versatile aspects as indicated by NRA in Table 8 for both design basis accident (DBA)and severe accident (SA).This means the applicants should prepare for the documents for all items listed in Table 8.

Table 7 Examining issues on earthquake, ground structure and tsunami.

Subjects	Items
Earthquake issues	Underground structures of site plant and the surrounding periphery
	Specified earthquake movement by identifying the origin of earthquake
	Selected earthquake movement without identifying the origin of earthquake
	Basis earthquake movement
	Anti-seismic design principle
	Geology of site and geological structure
	Stability of ground and slope
Tsunami issues	Basis tsunami
	Anti-tsunami design principle

Table 8 Accident prevention measures of nuclear power plant.

Subjects	Items
Design basis accident	External events and Internal flooding
	Fire, Tornado and Volcano
	Common equipment
	Single failure of passive equipment
	Protective power supply
	Human error prevention, Safety escape path and Safety protection circuit
	Reactor coolant pressure boundary
	Telecommunication equipment and monitoring equipment
Severe accident	Probabilistic risk assessment
	Accident sequence selection
	Effectiveness evaluation
	Analysis codes
	Control room
	Emergency response facility
	Filter vent facility
	Hydrogen explosion prevention

3 Document examination of applicant by NRA

The document examination of applicant's plant safety by NRA is conducted by the flow as shown in Fig.2.

The nuclear power plant which is applied for NRA had already built and had been operated for a long time but has been stopped operation after Fukushima accident. The NRA will examine whether or not the applied plant satisfies with the new standard set by NRA after Fukushima accident. The important point of NRA's examination to permit the restart of the plant can be summarized by the following A and B:

A. Is the plant system adequately improved to preclude the occurrence of Design Basis Accident (DBA)?

Regarding DBA, there are two major items to examine: (i) Items which are reinforced from the previous standards (*ex.* Single failure of passive equipment), and (ii) Consequence evaluation and countermeasures to the newly added external factors of natural phenomena such as internal flooding, volcano, tornado, fire, *etc*.

B. Are the implemented countermeasures adequate to minimize the consequence of Severe Accident (SA)?

Whether or not the newly implemented equipment and systems (such as. pump cars, high voltage generator car, filter vent system, on-site emergency response facility, *etc.*, and the related procedures for emergency response) will satisfy with the safety standard.

The whole flow of how to evaluate the effectiveness of introduced SA measures can be summarized as shown in Fig.3.

The blocks A, B, and C in Fig.3 are conducted by the applicant, while D, the part of judgment by NRA whether or not meets with the standard.

In A, the applicant will conduct PRAs where both internal and external events are taken as causes for bringing SA, by assuming no implementing SA measures. In B, by utilizing thus conducted PRA results, the applicant will reduce typical accident sequences which would lead to hazardous SA situations. And in C, for thus reduced accident sequences, the applicant will try to validate the effectiveness of implemented SA measures to manage SA. While in D, the NRA will confirm whether or not the implemented SA measures to the plant would satisfy with the new regulation standards.

The accident progression scenario as the target of evaluating the effectiveness of SA measures corresponds to the "Accident Sequence" in the block B of Fig.3. Concretely, the representative accident scenarios will be selected respectively by classifying the following four cases:

(A) Accident sequences in which the plant in operation state will lead to SA. There are seven accident sequences to be evaluated as listed below:

(A-1) Loss of both high and low pressure water injection functions.

(A-2) Loss of high pressure water injection and depressurization functions.

(A-3) Loss of all AC powers.

(A-4) Loss of decay heat removal function.

(A-5) Loss of reactor shutdown function.

(A-6) Loss of water injection function during LOCA.

(A-7) Containment bypass.

(B) Accident sequences in which the plant in shutdown state will lead to SA. There are four accident sequences to be evaluated as listed below:

(B-1) Loss of decay heat removal function.

(B-2) Loss of all AC powers.

(B-3) Loss of reactor coolan.t

(B-4) Mistaken reactivity insertion.

(C)Accident sequences that spent fuel pool will lead to SA. There are two type situations to be evaluated as listed below:

(C-1) Type 1 cases where cooling function and water injection are lost in the fuel pool.

(C-2) Type 2 cases where small water leakage with no water injection function occurs in the fuel pool.

(D)Different types of SA phenomena. There are six types of different SA phenomena as listed below:

(D-1) Containment failure due to over pressure and/or over temperature.

(D-2) Direct heating of containment atmosphere by discharge of high pressure molten materials.

(D-3) Molten fuel-coolant interaction outside of reactor pressure vessel.

(D-4) Direct contact to containment vessel (shell attack).

(D-5) Molten core- concrete interaction.

(D-6) Hydrogen burning.

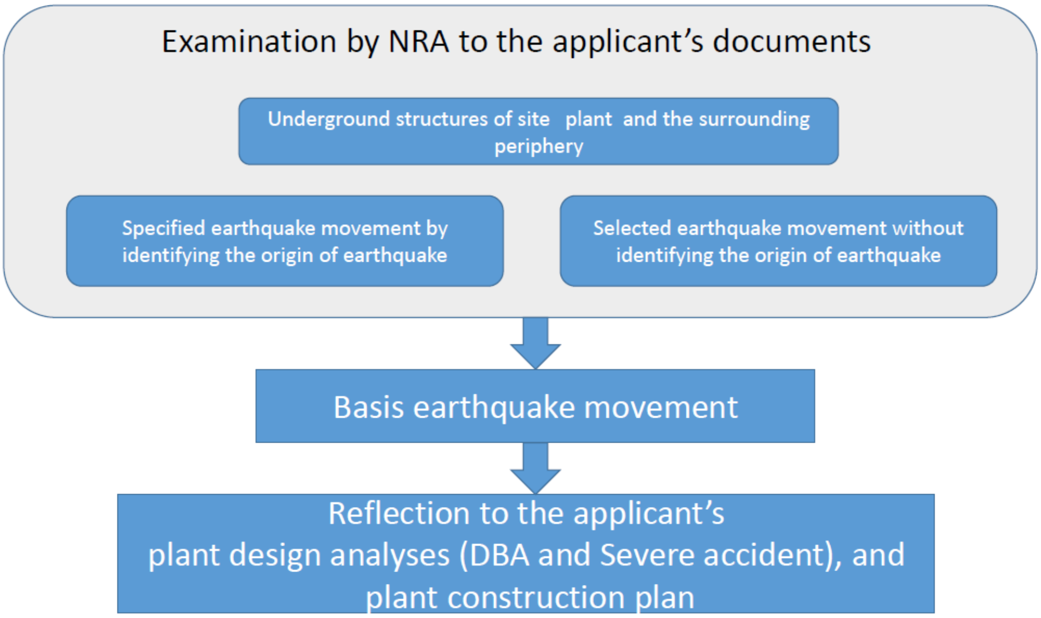

Fig.2 Flow chart of examination by NRA.

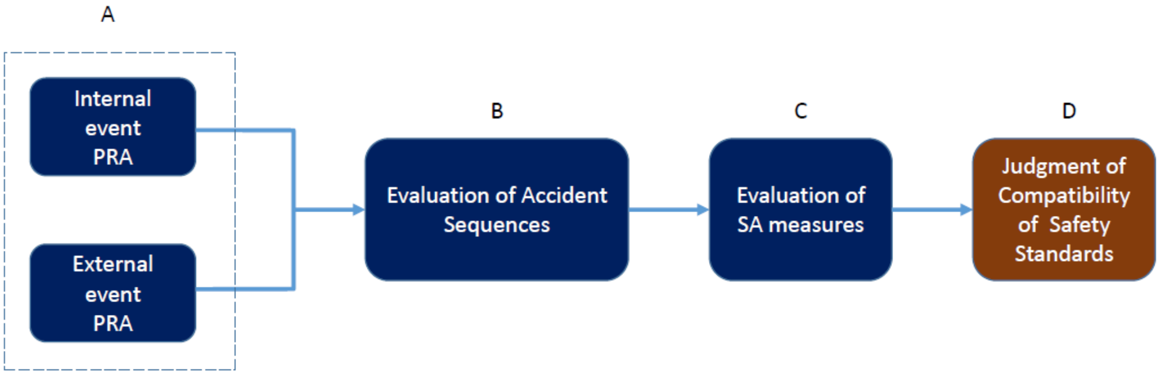

Fig. 3 Whole flow of examining the effectiveness of severe accident measures.

4 Application of severe accident analysis to reduce effective SA measures

4.1 Standard procedure of applicants for the restart of PWR

The general steps of how the NRA asks the applicants to conduct on evaluating the effectiveness of SA measures is already shown in Fig.3. While in this section, the real practice of the nuclear plant operation will be introduced to apply for the NRA for the restart of the safety enhanced plant. In fact, it is a hectic work of analysis and documentation as will be described one by one in this section.

4.1.1 Selection of analysis codes for effective SA measures

The analytic evaluation process to reduce the effective SA measures will be conducted by the following steps:

(i)Describe major event progression from a lot of initiating events to SA by grouping accident sequences obtained by PRA practice,

(ii)Select major evaluation indexes in accordance with major event progression,

(iii)Choose major physical phenomena by considering operational control of transient/accident,

(iv)Utilize EMDAP for PWR plant to establish SA analysis process for effectiveness evaluation of SA countermeasures, and

(v) Select appropriate SA codes which can describe important physical phenomena in SA and which can be used for evaluating the effectiveness of SA measures.

EMDAP in (iv) is the abbreviation of Evaluation Model Development and Assessment Process, and this is a hierarchical and structural analysis method proposed by AESJ in 2008 as the Execution Guideline of Statistical Safety Analysis. [1]
The detail of EMDAP will be described in 4.2.

4.1.2 Documentation work to show the effectiveness of SA measures

Documentation work will be conducted by the following order to show the effectiveness of SA measures:
(i)Describe Analytical Models and Calculation methods employed in the selected SA codes for the important phenomena,
(2)List up the database information of experimental analysis, benchmark analysis, *etc.*, to be used for validating the SA codes, explain how to do for scaling from those database information to the actual plant condition, and give the uncertainty of the SA codes,
(3)Clarify the method of how to confirm the appropriateness of the selected SA codes,
(4)Discuss on the influence of the SA analysis to the effectiveness evaluation of the SA countermeasures for the purpose of demonstrating the appropriateness of the SA measures.

4.2 EMDAP mapping chart and its application

The essential character of EMDAP mapping chart is that it will first list up different kinds of terminologies used for the safety analysis of nuclear power plant (*i.e.*, plant structure, models and equations, variables and parameters of various equations, and the physical phenomena) and then gives the interrelationship among those terminologies. The basic structure of the EMDAP mapping chart can be described as shown in Fig. 4.

In Fig.4, the hierarchical structure of the target plant as the “matter” will be first represented by the four levels of System-Subsystem-Module-Component. Concerning various constitutive equations which describe the related phenomena for a specific “Component”, the conditions on “Phase” with its “Geometrical pattern” assumed in the constitutive equations will be separately described by “Field” and “Transport process”, and then correlate with the keywords which describe “physical phenomena”, where two ways correspondence can be possible of either (i)”Transport process” to ”physical phenomena keywords” or (ii) “Field” + ”Transport process” to ”physical phenomena keywords”.

The actual EMDAP mapping charts employed in the PWR applicant’s document are different charts for the cases of prior to the core damage and after that. The EMDAP mapping chart as shown in Fig. 5 is a part of it prior to the core damage, and only the part of reactor core in the primary loop. This part takes notice on the differences of the physical phenomena simultaneously proceed in the reactor core, *that is*, heat generation by nuclear fission, structural change of nuclear fuel by neutron irradiation, and thermal-hydraulics in fuel assemblies, and in accordance with the three kind of phenomena classification, they expand the correspondence of “Field” and “Process” for each phenomena. The correspondence with phenomena keywords are not shown in Fig. 5.

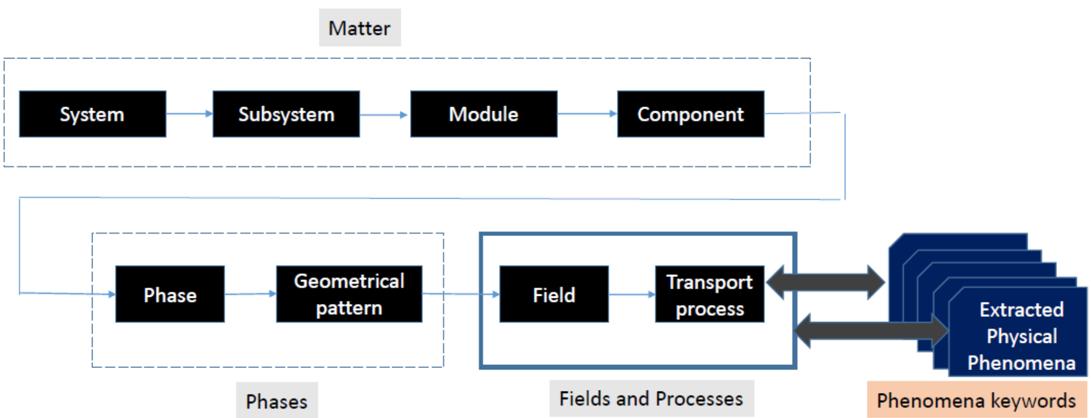

Fig.4 Basic structure of EMDAP chart.

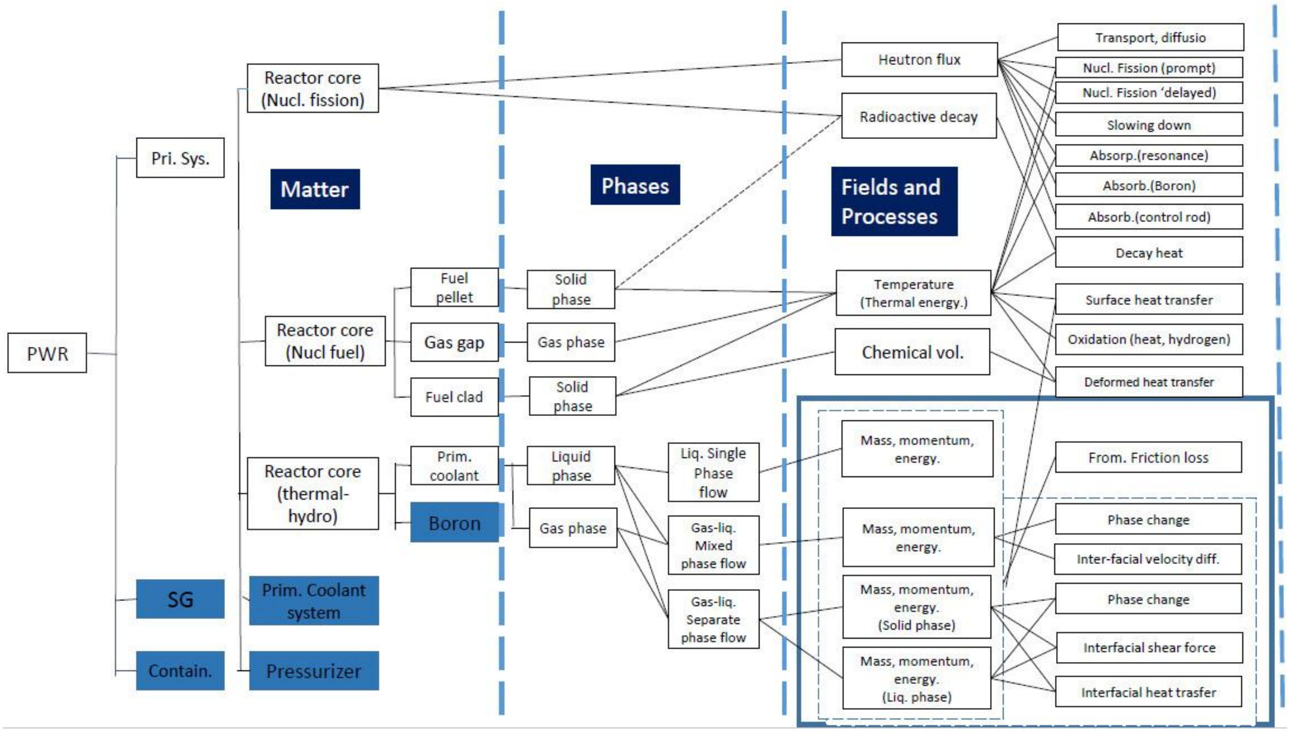

Fig.5 Example of EMDAP chart for PWR – a part of EMDAP chart prior to core damage.

At this point, the author of this paper tried to list up all the keywords of physical phenomena from the EMDAP mapping chart of the PWR applicant in order to know what kinds of different phenomena are involved in safety analysis which includes the SA.

The keywords associated with the relevant safety analysis which will span prior to core damage are shown in Table 9, where the keywords are classified into Reactor core、Primary loop、SG and secondary loop and Containment based on the configuration of the PWR plant system.

Those keywords in Table 9 can be also classified by the aspects of neutronics, thermal-hydraulics, and material property.

On the other hand, the keywords associated with the SA after core damage is listed in Table 10 by taking notice on where the phenomena happen: Reactor vessel or Containment vessel.

Table 9. List of phenomena keywords prior to reactor damage.

Parts	Phenomena keywords
Reactor core	Power distribution change; Fission power; Feedback effect; Control rod effect; Decay heat; Temp. distri. in fuel rod; Fuel surface heat transfer; CHF; Fuel cladding oxidation; Fuel cladding deformation; 3d thermal-hydraulics; Pressure loss; Boiling and void distri.; vapor-liquid separation; CCFL; Boron content change; Thermal in-equilibrium;
Primary loop	Flow rate change; Pressure loss; Boiling; Condensation; Void ratio change; vapor-liquid separation; CCFL; Thermal in-equilibrium; Coolant discharge rate; ECCS charging flow; Heat transfer to structure; Boron content change;
SG and secondary loop	Heat transfer between 1ry and 2ndry loops; Water level in 2ndry side; 2ndry feed water; Coolant discharge rate;
Containment	Flow rate within compartment and between compartments; Heat transfer at liquid-vopor interface; Spray cooling; Natural convection cooling in recirculation unit; hydrogen processing; Hydrogen content change; Coolant discharge rate; Heat transfer to structure;

Table 10. List of phenomena keywords after core damage.

Parts	Phenomena keywords
Reactor vessel	Fuel heat up leading to reactor core damage; Reactor vessel rupture; FP behavior within 1ry loop; Relocation; FCI within reactor vessel; Molten core fragmentation; Debris particle heat transfer; Reactor vessel rupture and melting; Core debris heat transfer in lower plenum; FP behavior in 1ry loop;
Containment vessel	Ex-reactor vessel FCI; Molten core fragmentation; High pressure core debris discharge after reactor vessel rupture; Expansion of molten core debris on cavity floor; Debris particle heat transfer; Direct heating of containment vessel atmosphere; Heat transfer between reactor core debris and cavity water; Flow within and inter-components; Heat transfer at liquid-vapor interface; Hydrogen generation by water splitting and radioactive ray; Hydrogen processing; Density change of hydrogen; Spray cooling; Natural convection cooling by recirculation unit; heat transfer between core debris and concrete; concrete disruption and non-condensable gas discharge;

It is also possible to classify them by noticing whether the phenomena is physical or chemical.

According to the PWR applicant which will be explained in 4.3, those EMDAP mapping chart is said to be useful for the selection of safety analysis code and setting of the evaluation index to confirm the effectiveness of the SA measures.

4.3 Example of safety enhanced PWR to meet with regulatory requirements

One example of a Japanese PWR which was recently permitted to restart is shown in Fig.6, where general features on what have been reinforced in the plant are mostly illustrated. This is Ikata Unit 3 of Shikoku Electric Power Co. ltd, which is located in Ehime Prefecture, north-west of Shikoku island of Japan.

This plant had been operated for a long year before Fukushima accident, but the operation was stopped for a while after Fukushima accident. Since then Shikoku Electric Power Co.ltd had continued to improve the safety measures by the request of Nuclear and Industrial Safety Authority (NISA, predecessor of NRA) and local government, applied for the restart permission to NRA in 2014 after the announcement of new regulation standard by NRA in July 2013, and finally got the permission from NRA with the subsequent approval of the local government in March 2016. The plant will soon restart in 2016.

In Fig.6, the word “add-on type” means that the plant safety has been enhanced by adding various SA measures to the existing plant already experienced operation. The systems and equipment indicated by thick solid lines and broken lines in Fig.6 are the added part to strengthen the SA measures. (Although not indicated, there are other countermeasures such as anti-seismic and tsunami measures, anti-fire measures, anti-tornado measures, *etc.*) In Fig. 6, MCR means main control room of the plant. The MCR is connected to on-site emergency center, off-site center in the local area and the emergency response center in Tokyo via respective networks. The safety-important I&C information assembled in MCR will be promptly distributed online to those centers in case of plant emergency situation.

It is considered that a lot of time, person and cost were needed for the Ikata Unit 3 to pay for the additional construction works and analysis and documentation works to meet with the NRA’s request to permit the restart of the plant. The essential works by the applicant to pass through the NRA’s requirements and the examination will be summarized in the subsequent sections.

4.3.1 Selection of accident sequence by full use of PRA

As the first step work to the effectiveness examination of severe accident prevention measures, the selection of significant accident sequence groups are selected by making full use of PRA. The plant condition of the target plat is not the real Ikata Unit 3 plant with adopting SA measures but the past plant condition before Fukushima accident. The accident progression assumed in severe accident analysis and the related containment failure modes are illustrated in Fig. 7.

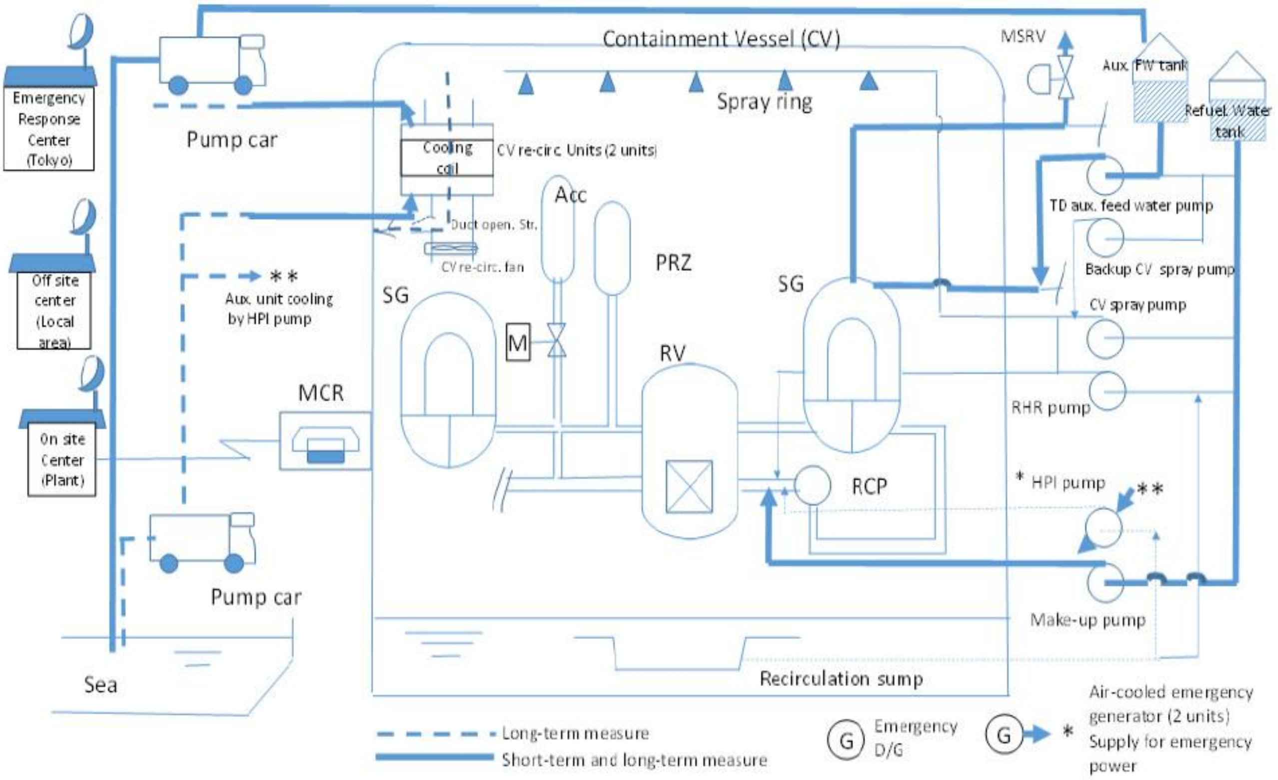

Fig.6 Add-on type safety enhanced PWR in Japan after Fukushima accident.

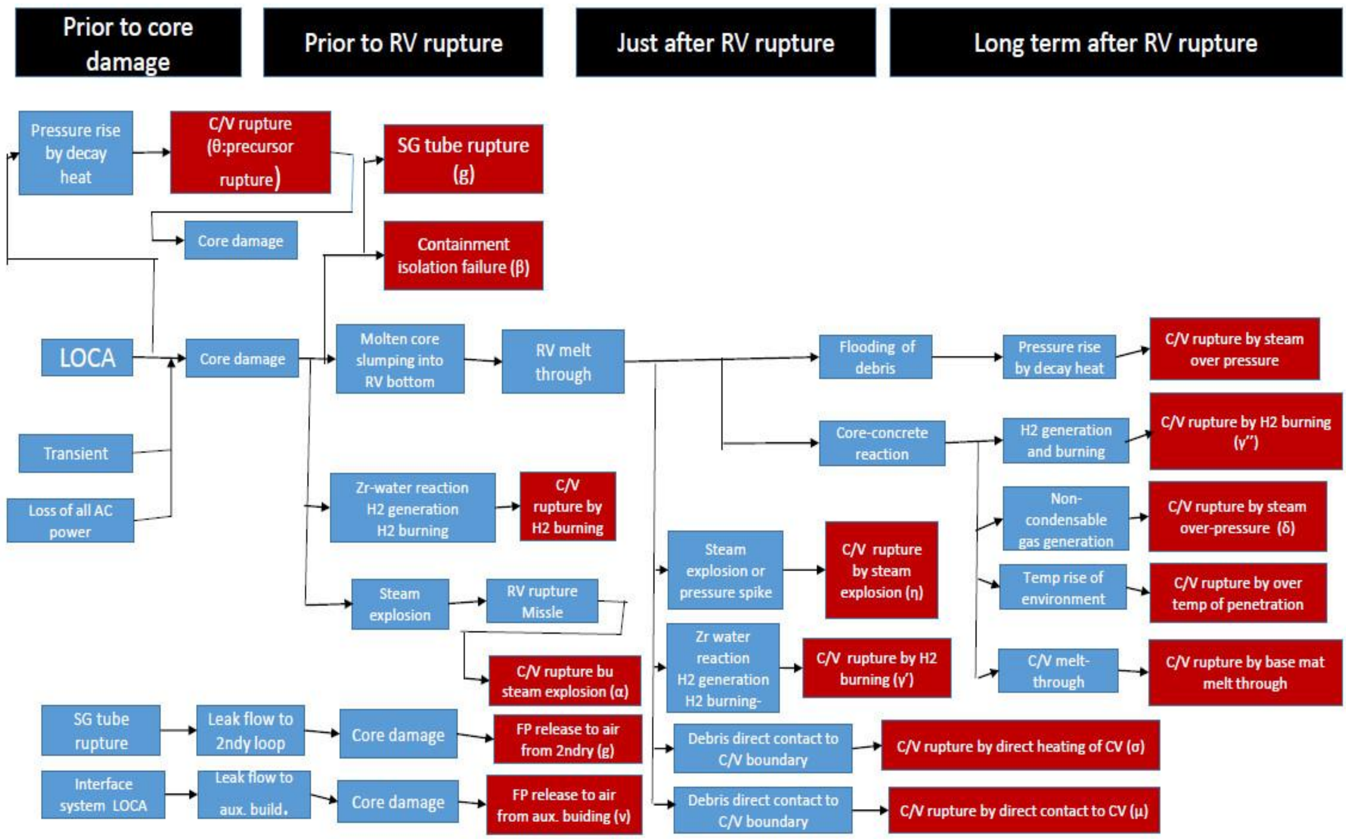

Fig.7 Accident progression assumed in severe accident and the related containment rupture mode.

The upper-most part of Fig.7 shows the four stages of progression of accident from left to right: prior to core damage, prior to reactor vessel (RV) rupture, just after RV rupture and long term after that. The lower part of Fig.7 shows the timings of various important accident phenomena as related with the four stages of accident progression and the sequential orders of those phenomena indicated by arrows. The phenomena drawn by red color indicate the important physical phenomena in SA analysis.

(1) Effectiveness evaluation of reactor core damage prevention measures

As for the PRA, the internal event PRA and the external event PRA considering both earthquake and tsunami are conducted. The other external events such as fire, flooding, *etc.* are treated as the effect to the initiating event in the normal internal event PRA. Many types of accident sequence groups are reduced by the above mentioned way of performing many kinds of PRA analysis.

As the result of both internal event PRA and Earthquake +tsunami external event PRA, the accident sequence of maximum core damage frequency (CDF) is the case of loss of cooling function of reactor auxiliary systems + RCP seal LOCA with total CDF of 2.4E-4(Event/Reactor Year) and contribution ratio to total CDF as 91.2% (CDF of internal event PRA 2.0E-4(Event/Reactor Year), CDF of earthquake PRA as 2.9E-5(Event/Reactor Year), and CDF of tsunami PRA as 1.3E-5(Event/Reactor Year)).

After the grouping of accident sequence, important accident sequences are selected by considering four factors of (a)common cause failure and inter-system dependency, (b)marginal time, (c)equipment capacity, and (d)representativeness.

The important accident sequence in the above-stated case of loss of cooling function of reactor auxiliary systems + RCP seal LOCA is selected as loss of external power + loss of emergency in-station AC power + loss of cooling function of reactor auxiliary systems + RCP seal LOCA, with all high influencing degrees in any of factors a, b, c, and d. The core damage prevention measures in this scenario is taken as forced cooling from secondary loop + air-cooled emergency electric power source + water injection to reactor core (by using charging pump of self-cooling type).

(2) Effectiveness evaluation of CV failure prevention measures

The next step is the selection of containment failure mode and the related accident sequence for the effectiveness evaluation of containment failure prevention measures. The containment failure modes are extracted by conducting level 1.5 internal events PRA and the quantitative evaluation to the external events for which PRA cannot be applied. Among the extracted containment failure modes, both the cases of containment vessel bypass and precursor containment vessel rupture are included in the part of effectiveness evaluation of reactor core prevention measures, because the both are already not anticipated to have the containment function at the onset of reactor core damage. The extracted containment failure modes are the following six modes among the modes shown in Fig.7.

(i) CV overpressure rupture （δ）
(ii) CV over temperature rupture （τ）
(iii) High pressure molten material ejection／CV atmosphere direct heating（μ、σ）
(iv) Ex-RV molten fuel and coolant mutual interaction（η）
(v) Hydrogen burning（γ、γ‘、γ’‘）
(vi) Molten core –concrete mutual interaction（ε）

The effectiveness evaluation of the CV failure prevention measures are made for each of those CV failure modes with assuming the severest condition of plant damage state (PDS), where PDS is the classification symbol for the core damage state given by level 1 PRA by noticing the kind of accident, speed of accident progression, state of water injection by CV spray.

The CV over-pressure failure by the accumulation of steam and non-condensable gas is evaluated to bring the largest CV failure frequency (CFF) with 2.0E-4 (Event/reactor year) and the contribution factor of 96.6% of total CFF. The corresponding measure of CV failure is water injection in CV by alternative CV spray pump and natural convection cooling in CV (by seawater).

4.3.2 Selection of safety analysis codes for effectiveness evaluation

All the electric power companies employing PWR for nuclear power use the PWR plants made by Mitsubishi in Japan. Before the Fukushima accident, those electric power companies also utilize the safety analysis code developed by Mitsubishi. However, for the application of the plant restart to NRA this time, all PWR companies in Japan became necessary to conduct severe accident analysis so extensively that they made the comparative evaluation of the available safety analysis codes including those developed in foreign countries. The discussion process on the selection of safety analysis code are so versatile that this subject will not be included in this paper, and only the result of the safety analysis codes selection is described in Table 11-1 to 11-3, in accordance with the four subjects of reactor core damage prevention, CV failure prevention, and fuel damage prevention in shutdown plant, respectively. As seen in Table 11, many US developed codes such as RELAP5 and MAAP are extensively used in severe accident analysis.

Table 11-1. Analysis codes used for the effectiveness evaluation of reactor core damage prevention.

Accident sequence group	Analysis codes
Loss of heat sink from 2ndry side	M-RELAP5
Loss of all AC power + Loss of cooling function of reactor auxiliary machines	M-RELAP5 COCO
Loss of cooling function of reactor containment	MAAP
Loss of reactor shutdown functions	SPARKLE-2
Loss of ECCS injection functions	M-RELAP5
Loss of ECCS recirculation functions	MAAP
Containment vessel bypass	M-RELAP5

Table 11-2. Analysis codes used for the effectiveness evaluation of containment vessel rupture prevention.

Containment rupture mode	Analysis code
Containment rupture by over-pressure and over-temperature	MAAP
High-pressure molten core emission/Direct heating of containment atmosphere	MAAP
Molten fuel and coolant interaction outside of reactor vessel in the paper	MAAP
Molten core and concrete interaction	MAAP
Hydrogen burning	MAAP, GOTHEIC

Table 11-3. Analysis codes used for the effectiveness evaluation of nuclear fuel damage prevention in shutdown plant.

Accident sequence group	Analysis codes
Loss of shutdown cooling function by RHR	M-RELAP5
Loss of all AC power	M-RELAP5
Leakage of reactor coolant	M-RELAP5

4.3.3 Uncertainty consideration in effectiveness evaluation

The uncertainty in the effectiveness evaluation of severe accident measures is considered to arise from three factors: (a)uncertainty of the occurrence of the related phenomena, (b)uncertainty in the modeling of the phenomena, and (c)uncertainty brought by analytic calculation of the related model. Concerning those factors (a) and (b), the applicants claim that the uncertainties of the important physical phenomena for the six CV failure modes are confirmed by conducting the sensitivity analysis. As for (c), they claim the degree of uncertainty can be obtained by setting the analytic conditions so that the effectiveness margins may decrease among the realistic design conditions.

4.3.4 Human factors issue on severe accident response

By taking into consideration of various environmental degradation of land-base, ground and road conditions caused by natural disasters such as earthquake and tsunami, the plant operators have to prevent reactor core damage and containment failure by making full use of SA countermeasures. For that purpose, the plant operator ordinarily set up the plant emergency response organization and allocate many staffs of emergency response teams in each of the reactors and on-site emergency response center. The human organization of the both teams and their allocated roles are illustrated in Fig. 8.

In Fig. 8 within the blue-colored block of right-hand side, six-group formation of emergency response team is indicated which are allocated both inside and outside of reactor building. Those six groups are (1)Information & communication group, (2)Broadcasting group, (3)Operation group, (4)investigation & recovery group, (5)technical support group, and (6)administration group.

Operators in the main control room are allocated to the operation group.

On the other hand, the gray-colored block in the left-hand side of Fig.8 shows the organizational allocation of staffs in the on-site emergency response center. The director and the vice director of the plant site and the senior reactor engineer, will take the commanding tower at the on-site center together with the head of six groups who are dispatched from each reactor. Staffs will be dispatched from Information & communication group to the off-site center in order to help smooth communication between the both centers.

Next, the whole configuration of the accident manuals are described in Fig.9, which are equipped in both main control rooms of each reactor and the on-site emergency response center.

The upper part of Fig. 9 shows the three kinds of accident manuals used in the main control room. On the other hand, the lower part shows the two kinds of manuals used in the on-site emergency response center. The manuals from left to right-hand side on either side of main control room and on-site emergency response center will correspond to design basis accident, core damage prevention and containment rupture prevention.

According to the application document of Ikata Unit 3, it illustrates the successful scenario of SA prevention for the case of loss of cooling function of reactor auxiliary equipment + RCP seal LOCA, as illustrated in Fig. 10.

The upper-most part of Fig. 10 indicates the elapsed time after the onset of accident. Until 100 minutes to 4 hours after the onset of accident, the staffs being on duties at the plant will conduct on respective actions as assigned, and after this initial stage, the off duties staffs will be called and be assembled to the plant and conduct their roles of the assigned groups.

The rest of all the Fig. 10 shows the time progression of major events and the corresponding actions in this accident scenario. At the time of accident, the reactor stops instantly. 10 minutes later, all AC powers are lost. 30 minutes later, initiate forced cooling by secondary loop. 52 minutes later, complete isolation of auxiliary feed water tank. 80 minutes later, resume forced cooling by secondary loop. 2.2 hours later, primary loop pressure reaches 0.72 MPa.

The below part indicates the action timing of the responding teams during the accident progression as described in the above. Those actions to be fulfilled by respective teams are assuring external power by generator-car, connection of feeding line to the pump-car to pump up the seawater, hose connection to the injection socket and cooling water injection to the reactor by pump-car.

According to the staffs of Ikata plant, Shikoku Electric Power Co. ltd., they exhibited right response in accordance with the prescribed scenario by the on-site training of the real operating staffs working in the plant. However, in an actual accident situation the real situation would not be the same as those assumed in the scenario, and when assumed condition would change, the actions to correspond to meet with the changing situation might change in many ways. Therefore, it would be necessary to enhance the ability of commanding staffs so that the emergency response team can exhibit their performance in any real situation successfully.

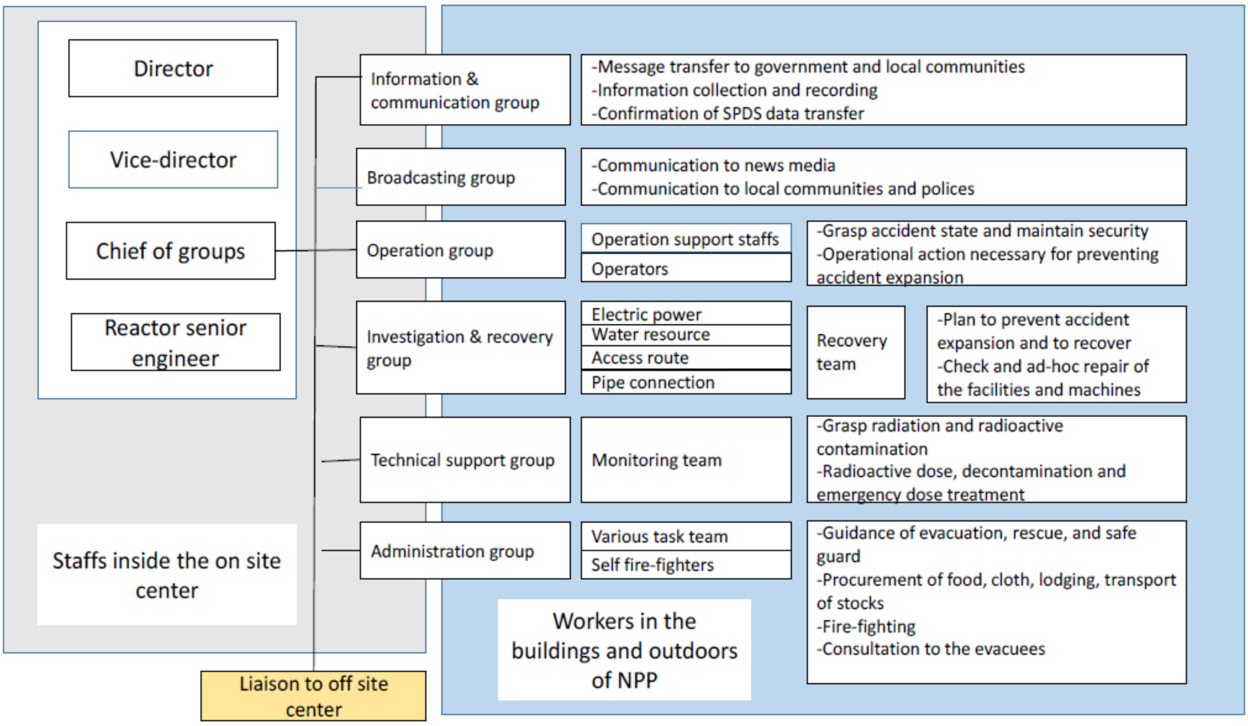

Fig. 8 Typical organizational structure of emergency response in a PWR plant in Japan after Fukushima accident.

Accident level and the goal of responsive action:			
	Desogm basos event	Prevent cpre damage	prevent containment vessel (CV) rupture
Main control room(MCR)	-Alarm handling procedure -Trouble and accident handling procedure (Part I): Event base procedure for design basis event	-Trouble and accident handling procedure (Part II): Maintain intactness of nuclear fuel in cases of design basis events(Safety function based procedure and event based procedure)	-Trouble and accident handling procedure (Part III): Prevent radioactive release to the environment by maintaining CV intactness in cases of leading to reator core damage
	Emergency response procedure		Accident management guideline
Onsite emergency response center	-Securing nuclear reactor facilities in cases of severe accident coocurrence and large scale destruction of the facility: Significant damages of reator core, nuclear fuels in the spent fuel pit, and large-scale damage of the reactor and the by the collision of large airplane and the terrorist attack.C14		-Integral management guideline to prevent accident progression and mitigating the consequence in case of core damage when the above Part III procedure will no more succeed. -Two types of guidelinesexist: monitoring function based guideline and that of whole evaluation of accident progression.

Fig. 9 A set of accident management procedures to be equipped in both main control room and on-site emergency response center.

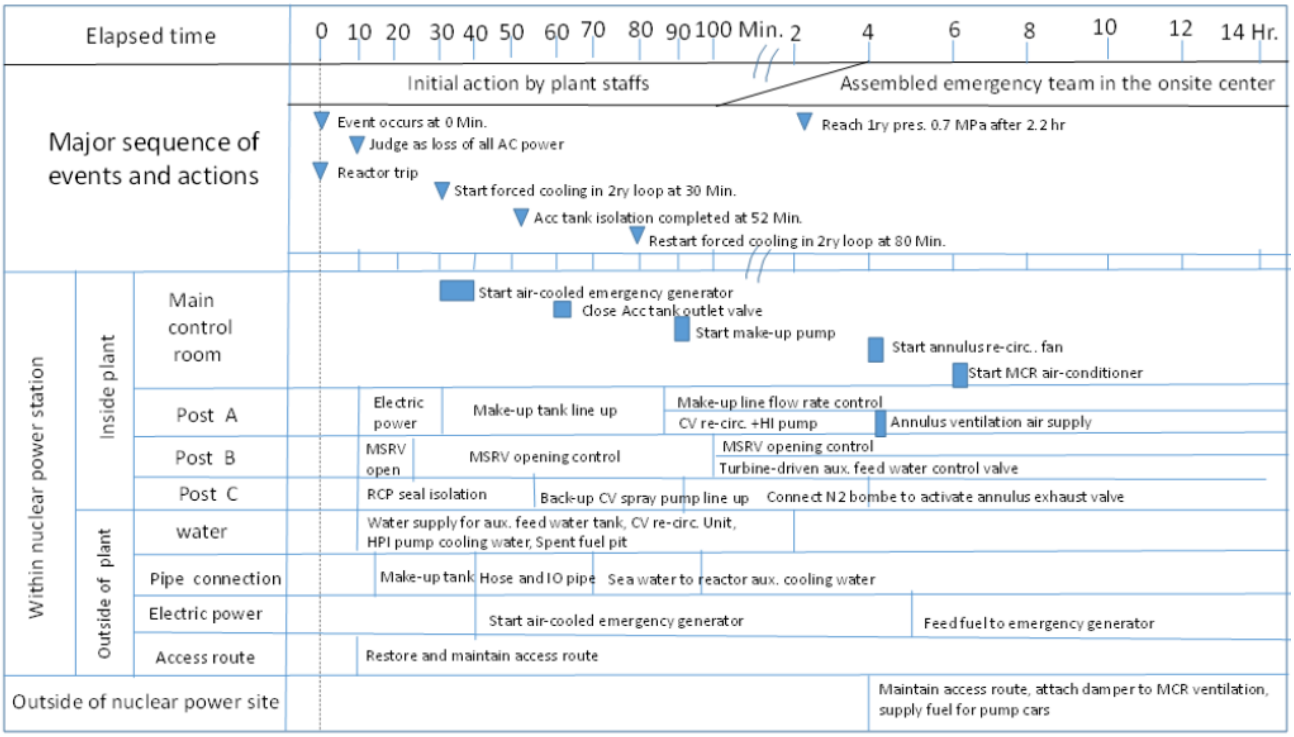

Fig.10 A whole scenario of emergency response team in case of the most probable severe accident in a PWR plant in Japan.

5 Discussion

Introduction of safety enhanced regulation standard by NRA to reflect the lessons learned from Fukushima accident has led the nuclear power operators to enhance the safety measures against severe accident. The in-depth application of PRA and SA analysis has been firstly conducted by Japanese nuclear industry to meet with the NRA's request to validate the effectiveness of the countermeasures of SA prevention (both core damage prevention and containment rupture prevention). Those are considered to be a great significance to the improvement of LWR safety in Japan.

However, in order to fulfill further restart of many PWRs and to sustain safe and stable operation further, it is thought necessary to make further effort in two aspects as mentioned below, in dealing with the analysis and application related with SA.

(1)Improve efficiency of whole process of safety analysis

There are many different subjects and phenomena dealing with the severe accident of nuclear power plant. And the phenomena regarding the severe accident have not been fully understood in many ways. So that the problems to improve the whole efficiency rest on two major issues: save computation time and decrease the uncertainty of the computed results.

(2)Improve the ability of emergency response team.

The essence of NRA's strengthened safety standard after Fukushima is thought to be the reinforcement of "hardware-oriented" SA countermeasures to be resilient in case of very severe conditions brought by almost all kinds of natural and human-caused disasters. The specific features of the emergency response in nuclear power plants and the similar nuclear facilities are to cope with "radioactive release" accident which may be superimposed by natural or human-caused disaster. The severe accident may not happen so frequently, and those hardware-oriented gadgets would be dormant in daily operation of nuclear power plant, and the drills of emergency response by plant staffs may become the ritual set by NRA in the meantime. However, the natural disasters and human-caused disasters might well happen by seeing the tendencies around the world these days. Therefore, human factors oriented SA prevention measures should be directed to the preparation for such "thinkable" severe situation to raise the ability of emergency response team by the concept of "learning organization".

The notion and ideas to respond with the two issues raised in this section, will be discussed in the subsequent sections.

5.2 Improving the efficiency of SA analysis

(1) Meaning of utilizing EMDAP

As for the application of Probabilistic Risk Assessment (PRA) for nuclear safety area, various statistical methods have been developed such as Monte Carlo method and models of common cause failure to evaluate specific effects of probabilistic nature of the target system. On the other hand, the original motivation of EMDAP has been developed to evaluate the uncertainty of the numerical calculation by two-phase thermal-hydraulic analysis codes which have been widely used in nuclear safety analysis.

However, the motivation of using the EMDAP which the operators who want to restart PWR use in their application document to NRA is difficult to recognize the reason to use it. It seems they use the EMDAP as the means to prove that the uncertainty of their severe accident analysis is not so influential, but it is hard to admit by the lack of modeling capability from the present level of understanding various severe accident phenomena in the nuclear reactor.

To sum up, it is difficult to find the merit of using EMDAP for the purpose of demonstrating that the uncertainty of severe accident analysis can be improved. However, since EMDAP mapping chart itself is represented by a well comprehensive structured knowledge on the target plant system with correlating with the physical analysis method and the phenomena in concern, there will be many possibilities of development more advanced and smart computational environment than what have been doing by the applicant. This may be realized by utilizing object-oriented knowledge processing and then to integrate it with more advanced simulation technologies. One example of such application would be the integration of SA analysis and PRA which will be discussed in the next section.

(2) Integrated frame of SA analysis and PRA

As the means for advanced risk evaluation for LWR, Risk-informed Safety Margin Characterization (RISMC) pathway has been developed by INL in US by the integration of SA analysis and PRA[2,3]. In this RISMC, the INL uses a unique computation frame called RAVEN (Reactor Analysis and Virtual control Environment) as is shown in Fig. 11.

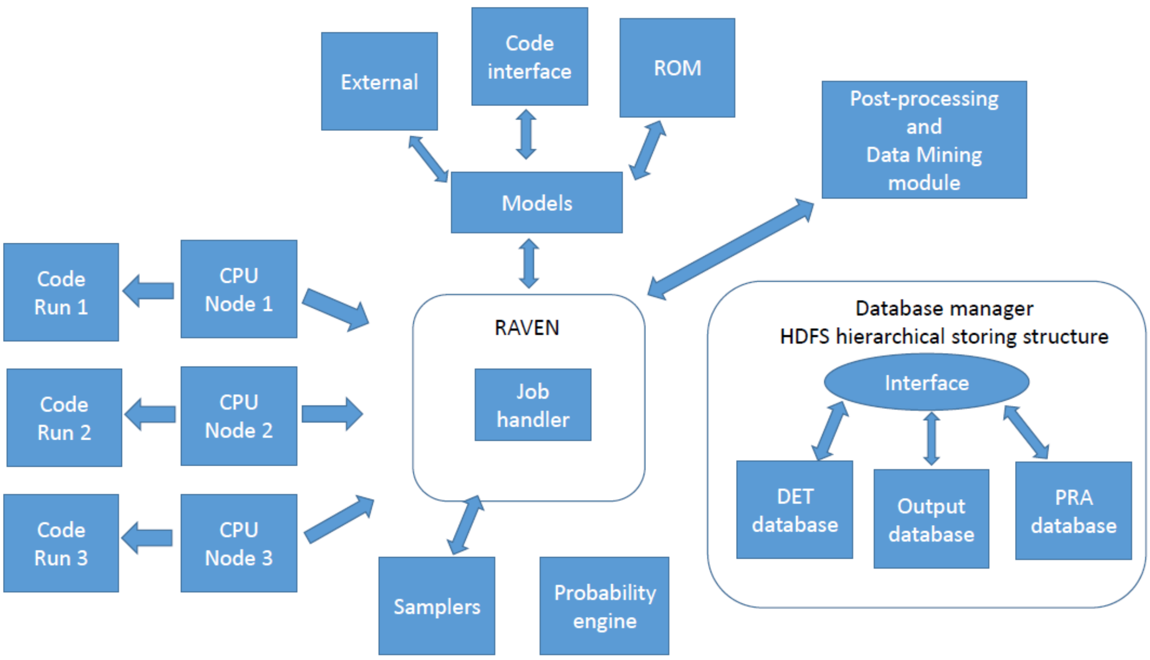

Fig.11 Overview of RAVEN statistical framework.

In Fig. 11, the RAVEN is composed by the following entities:

(i)Models: Interfaces to detailed computer codes, Reduced Order Model (ROM) and the external sources of computational tools,

(ii)Samplers: Control of sampling method for statistical analysis (Monte Carlo method, lattice grid sampling, Dynamic ET, *etc.*),

(iii)Database manager,

(iv)Post processing and Data mining,

(v)Parallel computing.

This framework is thought to be a general frame to integrate detailed computer simulation program into the statistical processing with the traditional ET/FTA procedure used in PSA.

By making full use of this framework, it will be possible to obtain the Reduced Order Model (ROM) as the substitute of original detailed safety analysis codes such as RELAP5. This is to improve the computation cost for the statistical analysis such as used in Monte Carlo Simulation. The typical example applications of RISMC would be probability estimation of rare events such as reactor core melting, future prediction of severe accident progression (time estimation of meltdown, effect of countermeasures to avoid SA.

5.3 Application of plant DiD risk monitor for designing SA procedures

The author of this paper has been developing a new risk monitor system, in order not only to prevent severe accident in daily operation but also even to serve as to mitigate the radiological hazard just after severe accident happens and long term management of post-severe accident consequences[4]. The conspicuous features of the proposed risk monitor basically lie on the two points, to be compared with the existing risk monitors: (i)The range of risk is not limited to core melt accidents but includes all kinds of negative outcome events, *i.e.*, not only precursor troubles and incident but also any types of hazard states resulting from a severe accident, and (ii)The whole system of the proposed risk monitor system is constituted by two layered systems as depicted in Fig.12.

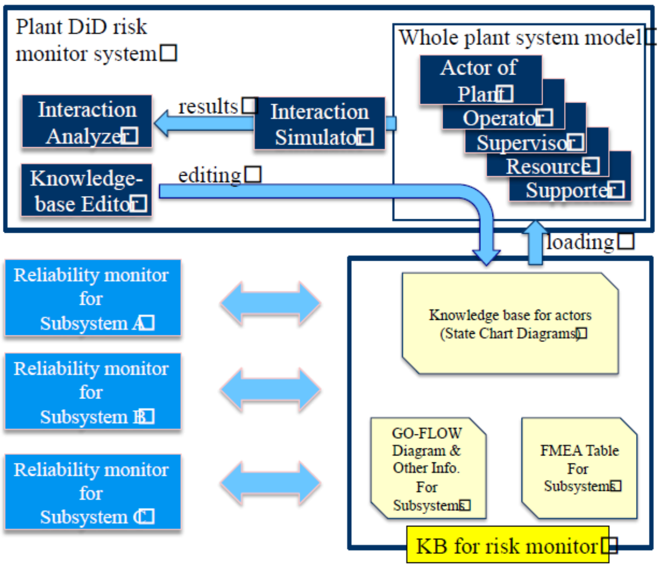

Fig.12 Whole framework of the author's proposed risk monitor system.

It is basically composed by a Plant Defense in Depth (DiD) Risk Monitor and several Reliability Monitors. The Plant DiD Risk monitor simulates dynamic plant situation, which is generated by interactions among all actors such as the plant and related people to cope with the situation, and evaluates plausible risk state from the situation. And the several Reliability Monitors evaluates the reliability of individual subsystems to fulfill their expected functions successfully under the prescribed situations, which are given by the Plant DiD Risk Monitor.

In Fig.12, various Knowledge Bases (KBs) which will be used for both Plant DiD Risk Monitor and Reliability Monitors are listed up in the block which is indicated as “KB for risk monitor”. The plant DiD risk monitor will identify every potential risk state caused by any conceivable event in the plant system as a whole where not only internal events but also external events arising from common cause factors and human factors should be taken into account.

The details of the software system of the developed Plant DiD Risk Monitor are described in the authors' recent publication [5]. In this paper only the result of a case study is introduced for the severest case of the safety enhanced PWR (as seen in Fig. 10 for this scenario). Although the plant actor is not connected to a plant simulator, it can offer different human-machine interaction by assuming different condition in the scenario, as can be seen from the result of this case study as shown in Table 11.

In Table 11, case 1 is the original assigned condition of emergency response team which consists of 2 supervisors, 8 operators and 17 supporters. The checkpoint for starting alternative water injection into core by charging pump is failed to do it within the time limit of 2 h 20 min: It is done at 2 h 44 min 24 sec in the case 1. The cause of the delay is considered as the lack of operator: although there are 8 operators in the main control room at the beginning of this scenario, 7 operators have moved to the field and only one operator remains in the control room. Since this one operator has to do all the tasks one by one, the operator must postpone the delayed task until completion of the previous tasks such as starting forced cooling of secondary system. To avoid this problem, one supervisor is shifted to help this operator in the case 2. The simulation for the case 2 is conducted, which is under the assignment of 1 supervisor and 9 operators and 17 supporters. In this case, all the checkpoints are done before their time limit, but checkpoint task to judge the accident should be delayed to be compared with case because there are many tasks immediately after the accident by one supervisor in this case. So the completion to judge the accident should be delayed. In the case 3, only the task of starting alternative water injection into core by charging pump is assigned to supervisor by modifying its procedure. The rest assignment conditions for the people are the same with case 1. The result of case 3 shows that all checkpoints are successfully done before the time limit and these are not delayed actions to be compared with the other cases.

These investigations are done easily and rapidly by the plant DiD risk monitor. The process for the investigation is very effective to understand the procedure, personal assignment, and potential problems among actors, so that the plant DiD risk monitor can be widely applied for the improvement of many human factors issues associated with the introduction of SA countermeasures in the plant management. It will range from the designing the SA procedures, to the education and training of the emergency response team.

Table 11. Result of case study by Plant DiD risk monitor

Checkpoint task	Time limit to finish the task (Hr.:Min.:Sec.)	Case 1	Case 2	Case 3
		Supervisors: 2	Supervisors: 1	Supervisors: 2
		Operators:8	Operators:9	Operators:8
		Standard procedure	One supervisor is moved to operator	Procedure is changed with the original staff organization
A. Judge accident	0:10:00	0:06:10	0:07:29	0:06:07
B. Start forced cooling of 2ry system	0:30:00	0:18:04	0:19:43	0:18:00
C. Supply electric power from alternative generator	1:00:00	0:37:12	0:38:51	0:37:09
D. Start alternative water injection into core	2:20:00	2:44:24	2:00:17	1:38:11
E. Reach hot shutdown state	4:00:00	2:44:24	2.33:34	2:33:41
F. Able to supply seawater to aux. feed water tank	11:00:00	5:00:18	5:01:58	5:00:17
G. Able to supply seawater to CV recirculation unit	51:00:00	6:28:39	6:30:19	6:28:38

6 Conclusion

The overview of the present state of nuclear power in Japan since Fukushima accident was first made in this paper. The main cause of Fukushima accident happened in March 11th, 2011 had been ascribed to the organizational defects in nuclear safety regulation by many investigation committees which were established by Japanese diet, Japanese government, *etc*. Then, the Japanese nuclear safety regulation authority in the past had been completely reorganized to form a new nuclear regulatory authority called NRA, and the NRA issued a new safety regulation standard for LWR nuclear power plants.

The new regulation standard established by NRA strengthened the severe accident measures in many ways from the past practice. Especially the NRA requested to utilize PRA not only for internal events but also for various external events which includes earthquake, tsunami, fire, *etc*. In this paper, the detailed procedure set by NRA for the nuclear power plant operators was explained to apply for the restart of their plants to pass the examination set by NRA. The major issue is how to assure the NRA on the effectiveness of the countermeasures against severe accident by the combinatory and extensive use of PRA and severe accident analysis. The procedures taken by the applicants were also introduced to get the permission by NRA as the real example to restart the safety enhanced PWR of Ikata Unit 3 of Shikoku Electric Power Co. ltd.

Lastly, the author stressed the problem of long time effort of the applicants to get through the licensing approval by NRA, and proposed to introduce a couple of ideas to improve the related safety analysis process by utilizing a new advanced IT methods. They are: (i) Improvement of PRA and SA analysis environment by adopting RAVEN frame being in development at INL, and (ii)Improvement of designing the on-site emergency response planning and the education and training of the emergency response team by the use of risk monitor system developed by the authors.

List of acronyms

AESJ	Atomic Energy Society of Japan
BWR	Boiling water reactor
CDF	Core damage frequency
CFF	Containment failure frequency
DBA	Design basis accident
DiD	Defense in depth
ECCS	Emergency core cooling system
EMDAP	Evaluation model development and assessment process
INL	Idaho National Laboratory
KB	Knowledge base
LOCA	Loss of coolant accident
LWR	Light water reactor
NRA	Nuclear regulatory authority
PDS	Plant damage state
PRA	Probabilistic risk assessment
PWR	Pressurized water reactor
RAVEN	Reactor Analysis and Virtual control Environment
RCP	Reactor coolant pump
RISMC	Risk-informed safety margin characterization
ROM	Reduced order model
RV	Reactor vessel
SA	Severe accident

Acknowledgment

This review paper was reproduced from the author's lecture at the 21st International Workshop on Nuclear Safety and Simulation Technology (IWNSSR2016), May 9th, 2016, in Harbin Engineering University. The authors of this paper largely borrowed the technical information from Shikoku Electric Power Company, for her applying the restart of their PWR plant, Ikata Unit 3, to the governmental organization of NRA. The author would like to express sincere thanks to the staffs of the Department of Nuclear Power, Shikoku Electric Power Co. ltd., for their offering of the related technical documents as well as their kind explanation on the contents. The related technical information has been open to the public and is basically available from the internet operated by NRA. Since those technical information including various safety analysis codes used by the PWR applicant are so many and versatile, therefore they are not listed in the references of this paper but only to indicate the URL of NRA. （http://www.nsr.go.jp/index.html）

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