Preventing Recurrence of Severe Accidents at Nuclear Power Plants

Report by the Committee on the Prevention of Severe Accidents at Nuclear Power Plants

April 22, 2013

The Committee on the Prevention of Severe Accidents at Nuclear Power Plants

Forward

The magnitude 9.0 Great East Japan Earthquake followed by a massive tsunami devastated Northeastern Japan and triggered an unprecedented severe reactor accident (hereinafter, severe accident) involving extensive release of radioactive material in the atmosphere.

With a strong sense of crisis on the urgency to clarify the causes of the accident from scientific and technical perspectives, Hiroyuki Abe proposed the establishment of this committee to offer solutions that would contribute to preventing recurrence of severe accidents.

The importance of investigating the severe accident from various perspectives by parties of different standings is without question. However, nuclear power generation is the fruits of technical R&D, and this inevitably calls for an investigation by scientists and technical experts. At the core of this investigation are six nuclear experts in support of Hiroyuki Abe's strong views that experts involved in promoting nuclear power generation and nuclear safety should be held responsible for investigating the causes of the accident.

The belief in "absolutely safe", or the "safety myth" of a nuclear power plant has unfortunately spread widely throughout Japan. However, there is nothing that is "absolutely safe"; all things in life, both natural and man-made carry a certain risk in opposition to their benefits. For nuclear power plants, not only the government and the operators, but also experts in the nuclear field are responsible for having neglected to appropriately communicate risks associated with nuclear power to the public.

The report provides basic approaches or concepts for prevention of the recurrence of severe accidents like the Fukushima accident, as well as the establishment of nationwide consensus on the benefit and the risk in utilization of nuclear power.

Committee on the Prevention of Severe Accidents at Nuclear Power Plants

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SUMMARY

The Background

The vast amount of energy released by controlled nuclear fission chain reactions for power generation has realized affluence and convenience for humankind. On the other hand, nuclear power generation involves radiation risk associated with fission products produced by nuclear fission, 90% of which are radioactive materials that continue to produce decay heat relative to their half-lives. Unless adequate amount of the decay heat is removed, fuel cladding (containing nuclear fuel and fission products) may fail, which may lead to the release of radioactive material into the atmosphere.

TEPCO's Fukushima Dai-ichi Nuclear Power Plant accident reaffirmed the dire consequences of e "radiation risk" . The prime factor contributing to the occurrence of the severe accident was the lack of fundamental attitude towards addressing "nuclear safety" and a lack of awareness on the roles and responsibilities to safety by all stakeholders including the operators (utilities), the government, manufacturers, the academia and the local government bodies, even though the significance in prioritizing safety was recognized.

On the basis of these reflections, the investigation, analyses and assessments on the TEPCO's Fukushima accident should be carried out and utilized for prevention and mitigation of future severe accidents. Consequently, an operational/management framework for implementing post-accident process of TEPCO's Fukushima Dai-ichi Nuclear Power Plants should be established as soon as possible and the re-establishment of the fundamental concept and objectives on nuclear safety should accompany this. Furthermore, establishing and maintaining a strong safety culture — a work environment where management and employees are dedicated to putting safety first—is an urgent issue that should be addressed through collaboration with the international community. Japan has a duty to fulfill these endeavors as a lesson learned from the disaster.

Key Issues of the Fukushima Dai-ichi Accident

The magnitude 9.0 Great East Japan Earthquake followed by "once-in-a-thousand-years" tsunami struck nuclear power plants located along the Pacific Coast in Northeastern Japan with impacts ranging from minor to catastrophic. The 15 meters above sea level tsunami resulting from the overlapping of multiple tsunami waves (constructive interference) induced by multiple earthquakes caused losses of all power sources and cooling functions at the Fukushima Dai-ichi Plant.

The size of the tsunami that hit TEPCO's Fukushima Dai-ichi Plant – tsunami height of 13.1 meters and inundation height of 15.5 meters against the site's ground level of 10 meters - triggered a series of events, initiated by damage and loss of functions of many of the key components due to inundation, causing SBO and functional loss of cooling components (e.g., water pumps), and

subsequent loss of DC power and final heat sink, which led successively to failure in decay heat removal, and to fuel damage and melting. The hydrogen explosion of Unit 1 delayed the accident management of other units, leading to loss of decay heat removal of Units 2 and 3, and consequently to extensive release of radioactive materials in the atmosphere and the sea.

Preventing the Recurrence of Severe Accidents

For preventing the recurrence of severe accidents, firstly, fundamental concepts and objectives on nuclear safety should be re-established (for example, by referring to "Fundamental Concept on Nuclear Safety – Part I: Nuclear Safety Objectives and Fundamental Safety Principles") and shared by all stakeholders, whereby, each party must fulfill their responsibility to ensure safety in accordance with the fundamental concepts and objectives.

Secondly, a new framework for ensuring the safety of plant systems should be established by developing new concepts on the prevention and the mitigation of beyond-design-basis events, together with relevant measures, and applying them to plant operation. In dealing with beyond-design-basis conditions (referred to as "severe accident conditions", and response measures against them, referred to as "accident management"), an accident management framework on provisions against various accident sequences should be created, reinforced by a consistent and continuous application of relevant new technologies.

Thirdly, an assessment on the significance of, and time allowance for the recovery of key safety functions required in preventing or mitigating severe accidents should be carried out, together with specific presentation of the recovery procedure. Preferably a comprehensive framework on the recovery procedure of key safety functions should be developed. It should be digitalized with the procedure manual to ensure that the complex procedures are carried out effectively.

Fourthly, high competency is required for the manager and the operating staff in the recovery operations beyond-design-basis accidents. For ensuring preparedness against such incidents, provisions are necessary for not only fundamental education and trainings, but also the establishment of a safety culture emphasizing of putting safety first, along with personnel exchange and enhancement of qualification programs. A system of assigning experts to each plant to deal specifically with the prevention and management of severe accidents should also be established. For securing high quality, competent experts and staff for plant operation which is complex and involve risk, and for a clear assignment of responsibilities, a qualification program with requirements corresponding to the job level in terms of skills and knowledge, responsibilities, and work conditions including compensation, should be established.

Fifthly, the regulatory body is responsible for the inspection and monitoring of the effectiveness of accident management measures (on both tangible and intangible aspects) carried out by the operators without omission. Both the operators and the regulatory body must maintain a process of enhancing

accident management measures independently and by liaising with each other, updating and making necessary revisions to the accident management measures.

Recommendations

With view to the severe consequences of TEPCO's Fukushima Dai-ichi NPP accident, a new framework for the continuous enhancement of severe accident management (for beyond-design-basis accidents involving significant fuel damage) should be established for the operation of existing plants in Japan. Preparedness and response measures in addressing extreme natural disasters as large-scale earthquakes and tsunamis, and other initiators of severe accidents should be formulated quickly, properly, and in good sequence, with account taken on design and siting conditions, etc., of each plant.

No measure in any industry will warrant 100% safety regardless of its completeness. There will always be some kind of risk or uncertainties. The measures are developed with a focus on minimizing the level of risk. This should be communicated to the public in gaining understanding and establishing consensus on the benefit of nuclear power generation.

Recommendation 1

Anticipating 'unforeseen' natural disasters or human events associated with nuclear incident is imperative. A fundamental approach in anticipating the 'unforeseen' (event) is essential for ensuring nuclear safety, and shall be developed.

Recommendation 2

A framework for ensuring nuclear safety should be re-established, whereby safety review guidelines and standards on safety should be re-assessed without being subject to preconceptions for developing a globally established framework of nuclear safety.

Recommendation 3

All related parties in the nuclear power community must recognize responsibilities commensurate with assigned roles, and establish the top priority in ensuring safety. The regulatory body, in particular, must determine fundamental principles on the prevention and mitigation of consequences of severe accidents by hearing the opinions of a broad spectrum of experts. The operators must determine severe accident measures and implement them effectively with a sense of vigilance.

Recommendation 4

The State and the operators should independently and/or jointly –along with scientists and experts in nuclear technology field through professional societies –establish risk communication on nuclear

power generation with the public as well as promote activities in establishing public consensus on the benefit and risk of nuclear power generation.

Followings are the recommended specific measures.

Recommendation 5

The regulatory body shall regulate plans and inspections on severe accident prevention and mitigation measures proposed and prepared by operators. In the examination of the measures, all internal events (including human error events, etc.), natural phenomena and human-induced events associated with severe accident should be included. By cooperating with experts and licensees, the regulatory body should develop effective accident management by combining measures, including the use of a variety of components and equipments for preventing and mitigating severe accidents.

Recommendation 6

Reliability of safety functions for the prevention and mitigation of severe accidents shall be ensured through elimination of common cause failures, by ensuring independent effectiveness through distributed arrangement and diversification of safety functions.

Recommendation 7

Specific measures for accident management should be flexible as to address unanticipated conditions which may not be dealt with by permanent facilities. Thus, transportable and mobile facilities (fixed on vehicles), and redundant connections should be provided for flexibly coping with all circumstances.

Recommendation 8

Operators must assign onsite accident management specialist(s) with a thorough understanding of the nuclear power generation system, having competence to understand or assume likely circumstances of the nuclear reactor under accident conditions, and the ability to make appropriate judgment in providing necessary directions to onsite staff.

Recommendation 9

Operators shall prepare an accident management procedure manual by confirming each item of the manual at the site, on the basis of which education, drills and exercises under all credible conditions shall be fully provided to the staff.

Recommendation 10

The regulatory body shall conduct inspection and surveillance of accident management without omission. Licensees and the regulatory body should independently, or in cooperation carry out reassessment for continued enhancement of accident management.

1. INTRODUCTION

The Tohoku Region Off The Pacific Coast Earthquake, one of the biggest earthquakes recorded in history with a Magnitude of 9.0 (M9) occurred on March 11, 2011. The earthquake induced crustal movement in the areas between Off-the-Coast Sanriku and Off-the-Coast Choshi, spanning over 450 kilometers in length and 200 kilometers in width, of 60 to 70 meters. Catastrophic tsunami that followed struck northeastern Japan and devastated power plants in the areas along the Pacific Coast. The tsunami with a height of 15 meters struck TEPCO's Fukushima Dai-ichi Nuclear Power Plant, triggering loss of reactor core cooling which subsequently led to the extensive release of radioactive material into the atmosphere.

With the occurrence of the earthquake, control rods were automatically inserted into all of the reactors in 12 nuclear power plants in operation located along the Pacific Coast of Eastern Japan, and cold shutdown was confirmed. Units 2 and 3 of TEPCO's Fukushima Dai-ichi Nuclear Power Plant and Units 1, 2 and 3 of Tohoku Electric Power Company's Onagawa Plant experienced beyond design basis seismic motion, however no abnormal conditions nor damages were reported. On the other hand, the tsunami height at all of the nuclear power plants of Onagawa, TEPCO's Fukushima Dai-ichi and Dai-ni, Japan Atomic Power Company's Tokai Dai-ni exceeded both the original design basis and the revised design values. In particular, although the tsunami height at Onagawa Plant was over 13 meters, the plant missed damage by a small margin because of the site's ground height of 14.8 meters (13.8 meters after ground subsidence). Tokai Dai-ni Nuclear Power Plant maintained integrity and reached a cold shutdown because a major part of the water-proofing bulkhead construction for cooling components and facilities was completed just before the earthquake event. Although many SSCs including residual heat removal systems suffered damage due to the tsunami inundation height of 14.5 meters at Fukushima Dai-ni Plant (tsunami height was 8 meters against the site's ground height of 12 meters), the plant was brought to a cold shutdown because accident management measures such as installation of temporary power supply equipments and seawater pumps had been arranged in advance.

The investigation on TEPCO's Fukushima Dai-ichi Nuclear Power Plant accident so far has been carried out by the Diet (**The National Diet of Japan Fukushima Nuclear Accident Independent Investigation Commission**), the government (**Investigation Committee on the Accident at the Fukushima Nuclear Power Stations of Tokyo Electric Power Company**), and the private sector (**The Independent Investigation Commission on the Fukushima Nuclear Accident**), with the results compiled into reports as of July 23, 2012, recommending continued investigation for clarifying many issues that remain unresolved [1), 2), 3)].

Although the reports were formulated on the basis of different approaches, all groups conducted

site investigation and made briefings on related parties, concluding as the cause for the accident was the lack of fundamental understanding on severe accident (synonymous with "*kakoku jiko*", the term was formerly "severe accident"; however renamed to "*ju-dai jiko*" (serious accident) under the Nuclear Regulation Authority Establishment Act). The accident measures focused on internal initiators to the exclusion of external events as natural disasters, and consequently, the shortfalls in preparedness measures for earthquake and tsunami events and in the understanding on SBO. In addition, the emergency response management onsite, crisis (emergency) management by the Cabinet Office, the regulatory body, offsite response center, head office of Tokyo Electric Power Company were in turmoil, losing all command and control. The prime cause that led to the core melt in Unit 1 was the lack of understanding and trainings in utilizing the IC. The delay in containment venting definitely added to worsen conditions, which led to the hydrogen explosion. Whereas, the Diet Investigation Commission claims that because of the ruptured primary piping system caused by the earthquake, not using the IC was a rational judgment, which other reports have not made a point of. A report on detailed analysis and assessment of the accident by the former Nuclear and Industrial Safety Agency (hereinafter, NISA) denied the rupture of the primary piping system. Means for alternative cooling was not provided for during the three days that the RCIC was in operation in Unit 2; and in Unit 3, alternative cooling method for the HPCI was not prepared for before the shutdown. Various reports have touched on the inadequacies in emergency response management of all related parties, the confusions caused, lack of effective management of resident evacuation, and the causes for these shortfalls, which this report will not extend to. The report will focus on the extensiveness of the recommendations on the prevention of severe accidents discussed in the investigation reports.

The **Diet Investigation Commission** emphasized the vulnerabilities in severe accident management that did not consider external events (e.g., earthquake and tsunami, etc.), human events (e.g., terrorist attack, etc.) and extended SBO, but was limited only to internal events (e.g., erroneous operation). Because severe accident management was not regulated and left to the voluntary discretion of the operators, the effectiveness of the measures diminished. The report also pointed out that the regulatory body did not reinforce measures for ensuring defense-in-depth although they were aware that the requirement in Japan was only up to defense-in-depth Level 3 against the international standards of Level 5. Additional flaws pointed out were neglecting to reflect in the Japanese regulatory framework, "Station Blackout and Advance Accident Mitigation (B.5.b)" requiring provision of safeguards and trainings for SBO, issued by the US NRC after 9.11 terrorist incident though this was well recognized by the Japanese authorities. However, the Diet Investigation Commission report did not extend to extensively discuss factors related to the severe accident to define clearly how future severe accident measures should be. Instead, it simply stated, "regular monitoring and updates on accident management must be implemented on the basis of the lessons learned on accidents, global trends on safety standards and the application of state-of-the-art

technologies, in order to maintain the highest standards and the highest technological levels globally" (Article 3 of Recommendation 6 "Reforming Laws Related to Nuclear Energy").

As with the Diet Investigation Commission, the **Government Investigation Committee** emphasized the significance of severe accident management that includes external events. In the recommendations, it points out the necessity of a comprehensive risk analysis and severe accident management in (4) "Analyses on Accident Prevention Measures and Disaster Preparedness" of 1. "Analysis of Key Issues" - "nuclear operators should conduct comprehensive risk analysis encompassing the characteristics of the natural environment including external events, of not only earthquakes and their accompanying events but also other events such as flooding, volcanic activities or fires, even if their probabilities of occurrence are not high, as well as internal events having been considered in the existing analysis. Nuclear regulators should check the operators' analysis." In the formulation of severe accident measures based on comprehensive risk analysis - "In order to ensure and maintain nuclear safety at nuclear power stations, vulnerabilities against a wide range of internal and external events should be identified for each facility through comprehensive safety assessment, and effective severe accident management measures that include assumption of core damage caused by events exceeding design basis should be developed. The effectiveness of such severe accident management should be evaluated through the PSA or other means." The issues pointed out are relevant, however, extensive examination on these issues have not been made in the report. In addition, the report is formulated on the premise that the operators must take initiatives in severe accident management with the regulatory authorities confirming the adequacies of the measures taken by the operators.

The Independent Investigation Commission of the private sector also pointed out the inadequacies in severe accident management, claiming that the reason for the shortfall in promoting severe accident management in Japan was because nuclear regulatory control placed emphasis on the hardware aspects as structural strength, which hindered the establishment of quantitative risk assessment. However, no specific recommendations have been made on the future enhancement of nuclear safety.

Reports on the investigation and analyses on Fukushima incident from different perspectives have been prepared by various organizations, including those by the Tokyo Electric Power Company. [4), 5)]

The Committee on the Prevention of Severe Accidents at Nuclear Power Plants made extensive analyses on the information collected and the activities carried out so far, and examined from the two perspectives of "what were the causes that led to the severe accident", and "what are the key issues that should be addressed in preventing the recurrence of such accident", to which recommendations for resolving these issues have been developed. With view to the geographical, natural and societal conditions – dense population in small geographical area, frequent earthquake and tsunami events -

of Japan, the need for enhancing safety of nuclear power facilities to prevent severe accidents and mitigate adverse consequences of radiation to the local public has been thoroughly discussed. For preventing recurrence of a severe accident, lessons should be learned from cases which have led to failure as well as to success.

In view of the New Safety Standards by the Nuclear Regulation Authority, the Committee has focused on the examination of, and the recommendations on measures for preventing severe accidents; disaster preparedness measures have not been examined due to time constraint.

On January 23, 2013, the Committee issued the "Interim Report on the Prevention of Severe Accidents at Nuclear Power Plants", with press releases and reported to the Nuclear Regulation Authority. The Committee also developed recommendations in response to public comments invitation on the draft "New Safety Standards" February 7th - February 28th, 2013 by the Nuclear Regulation Authority, and made a presentation to the Regulation Authority on February 26th, the details of which are outlined in the Annexes. The summary of the "Interim Report on the Prevention of Severe Accidents at Nuclear Power Plants" is in print under the title, "Commentary: Recommendations on Measures for the Prevention of Severe Accidents at Nuclear Power Plants" in the May issue of "Atmos", the Atomic Society of Japan journal.

For ease of understanding for those not familiar with nuclear technology, technical terms have been defined in **"Glossary"** at the end of the volume.

2. DEVELOPMENT OF SEVERE ACCIDENT MANAGEMENT IN JAPAN

2.1 Examination and Decisions by the Former Nuclear Safety Commission on the Basis of Three-Mile Island and Chernobyl Accidents

The term, “severe accident” first came into use in the wake of the Three-Mile Island incident. Internationally, several definitions exist -the OECD/NEA defines the term as “events leading to significant reactor core damage, initiated by an event exceeding design basis considerations (events within design basis consideration is “design basis event”), inducing failures in adequate core cooling or reactivity control by safety design measures. The severity of a severe accident will be measured by the degree of the core damage and the extent of the loss of containment facilities.” “Design basis event” refers to “events that could lead reactor facilities to an abnormal state and that should be considered in the safety design of reactor facilities and its evaluation.”

With the occurrence of the Three-Mile Island and Chernobyl accidents, the former Nuclear Safety Commission established Special Committee on Nuclear Accident Investigation, for examining causes of the accidents as well as to reflect the results of the analyses on the plants operating in Japan. At the same time, after the Three-Mile Island incident, Working Group on Containment Vessel and Working Group on Hydrogen Gas Measures were established under the Common Issues Committee of the Special Committee on the Safety Standards for Nuclear Reactors. The Working Group on Containment Vessel conducted the following: made reviews on the various approaches on light water type reactor facilities in the international community with particular consideration given to the trends in the US and EU; organization of knowledge and expertise on severe accident events accumulated through safety researches made so far; survey on the safety margins of the representative plants in Japan against severe accidents on the basis of PRA (Probabilistic Safety Assessment); and reported the results to the Common Issues Committee. (Refer Fig. 2-1)

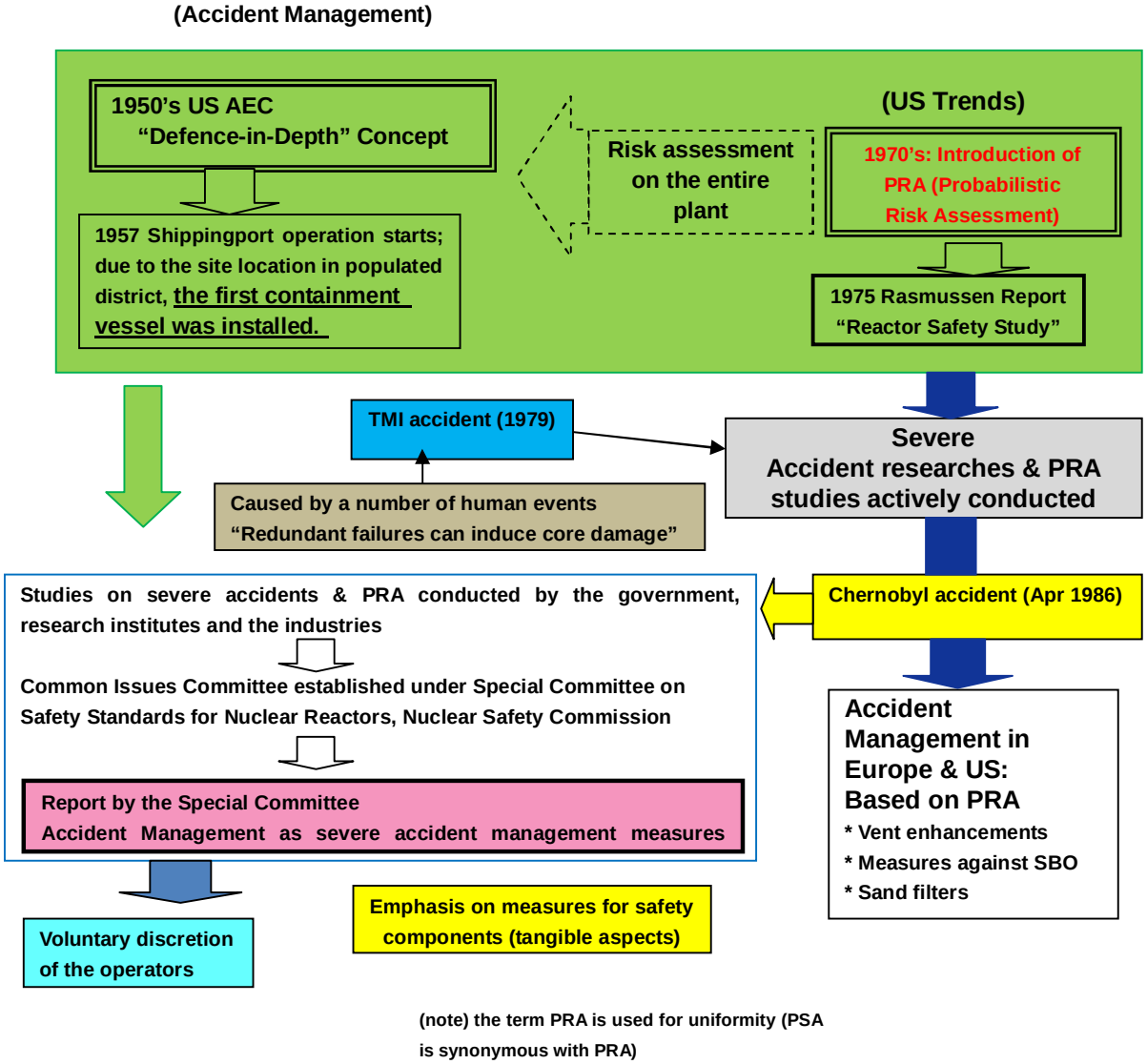


Fig. 2-1 Development of Severe Accident Management in Japan

The former Nuclear Safety Commission received "Examination Report on Accident Management for Severe Accidents at Light Water Power Reactor Installation – Measures for Containment Vessels" (hereinafter referred as the "Report") from the Common Issues Committee of the Special Committee on the Safety Standards for Nuclear Reactors (hereinafter referred as the "Common Issues Committee")[6] The report examined the fundamental approach Japan should take in dealing with severe accidents on the basis of: 1) the significance of measures for preventing escalation to severe accidents and for mitigating consequences of severe accidents (hereinafter referred as AM, or accident management) recognized for enhancing safety of light water reactor facilities; and 2) the trend in many countries of adopting measures for enhancement of reactor containment as part of

accident management.

The former Nuclear Safety Commission formulated the following policy after examining the report, encouraging further efforts by the licensees (nuclear reactor establisher) and the administrative body to enhance safety.

The safety of nuclear reactor facilities is sufficiently ensured through rigorous safety measures founded on the concept of graded protection (synonymous with defense-in-depth) set forth under 1) preventing abnormalities from occurring; 2) preventing abnormalities from escalating to accident conditions; and 3) preventing extensive release of radioactive material; throughout design, construction and operation phases, under the current safety regulations. These measures have significantly lowered the risk on nuclear reactor facilities, minimizing the likelihood of a severe accident to almost nil from an engineering perspective. The organization of accident management is expected to further reduce the risk. Accordingly, the Nuclear Safety Commission strongly recommended the licensees to voluntarily establish effective accident management and to ensure that the measures are adequately implemented in the event of emergencies. The framework on Japan's severe accident management is shown in Fig. 2-2.

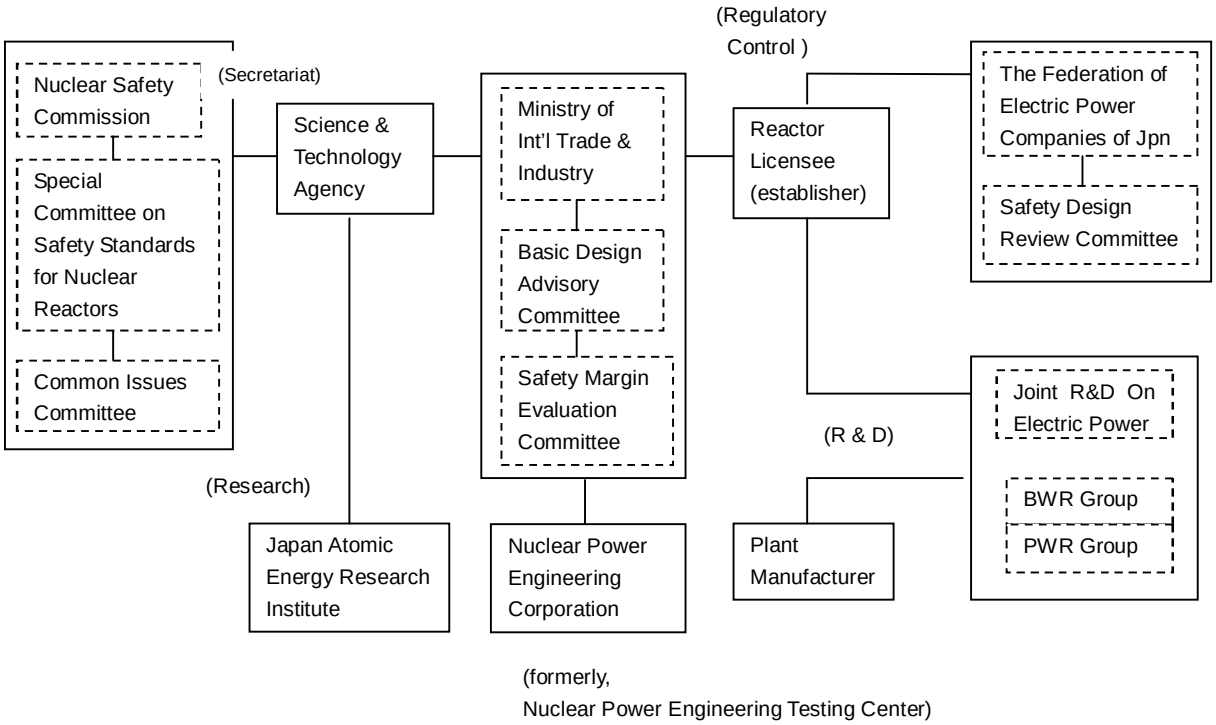


Fig. 2-2 Japan's Severe Accident Management (as of 1992) [7)]

2.2 Response of Ministry of International Trade and Industry (MITI) and Operators on Decisions by Nuclear Safety Commission

In response to decisions by the Nuclear Safety Commission, the Ministry of International Trade

and Industry (MITI) requested the utilities to newly develop accident management, not as regulatory requirement but as part of the ongoing voluntary safety measures in July 1992.

In response, the Tokyo Electric Power Company (TEPCO) conducted PSA (Probabilistic Safety Assessment) focusing on abnormal incidents (internal events) caused by component failures, etc., during power operation on all units of Fukushima Dai-ichi Plant. On the basis of results of the PSA analysis and expertise accumulated so far on severe accident events, a policy by TEPCO to further develop accident management for enhancing safety of nuclear power plants was compiled as "Report on the Deliberation of Accident Management Development" and submitted to MITI in March 1994.

Later, the establishment of key items extracted and presented in "Report on the Deliberation of Accident Management Development", such as the organization of accident management measures, framework for AM implementation, documented procedures, the operational aspects as trainings, etc., was completed; and the results were compiled as "Report on the Results of Accident Management Development" and submitted to the Ministry of Economy, Trade and Industry (METI, formerly, MITI) in May 2002[8)]. TEPCO reported that the effectiveness of accident management development in enhancing safety of nuclear plants was quantitatively shown by the adequate reduction of core damage frequency (CDF) and containment failure frequency (CFF). Report on the results of accident management development was submitted by each plant. The overview on accident management development carried out by Fukushima Dai-ichi Plant is as follows.

1) Development of Accident Management

Specifications of the reactor units in Fukushima Dai-ichi Plant:

Unit 1: power output: 460MW; type: BWR-3

Unit 2 – 5: power output 784MW; type: BWR-4

Unit 6: power output 1100MW; type: BWR-5

Accident management have been implemented under the following four functional categories: "reactor shutdown functions", "reactor and containment water injection functions", "containment heat removal functions", "support systems of safety functions". Implementation of new accident management, together with existing accident management are shown in Table 2-1 to Table 2-3[8)].

(1) Unit 1 (BWR-3)

1) Accident management related to reactor shutdown function

RPT (recirculation pump trip) and ARI (alternative rod insertion) was provided in addition to power output control through manual scramming, water level control, and boron injection.

2) Accident management associated with reactor and containment water injection

In the event of ECCS activation failure, the following containment water injection

enhancements were made in addition to existing methods of condensate feed-water systems, control rod drive system, manual activation of ECCS, manual operation of reactor depressurization and low pressure water injection: piping connections were modified to enable injection via core spray system from condensate feed-water systems and fire extinguisher systems; containment spray via containment cooling system; condensing steam by the spray system; and cooling of the accumulated debris in the pedestal (lower space of the pressure vessel), etc.

3) Accident management associated with containment heat removal

In the event of failure in activating containment cooling system and subsequent increase of containment pressure, arrangement for venting via inert gas and emergency gas processing systems, and the hardened vent was made.

4) Accident management related to support functions of key safety functions.

Sharing of power sources, organization of recovery procedures for emergency diesel generator, and the exclusive use of emergency diesel generator.

(2) Unit 2 to Unit 5 (BWR-4)

1) Accident management involving reactor shutdown function

Same as Unit 1.

2) Accident management related to reactor and containment water injection

Arrangement of automatic alternative water injection and reactor depressurization.

(1) Alternative water injection

Basically the same concept as with Unit 1.

(2) Automatic reactor depressurization

When the signal for low reactor water level is issued, safety relief valve will be activated for automatic depressurization of the reactor which will enable water injection via low-pressure ECCS, etc.

3) Accident management associated with containment heat removal.

Basically, the same as Unit 1.

4) Accident management related to support functions of key safety functions.

Basically the same as Unit 1.

(3) Unit 6 (BWR-5)

1) Accident management related to reactor shutdown functions

Basically the same as those of other units.

2) Accident Management Implementation Framework

For the development and implementation of accident management, various information related to plant parameters, etc., must be collected, analyzed and assessed to understand plant conditions, and

to comprehensively examine and select accident management measures. The organization that will implement accident management must be clarified, with the roles and responsibilities specified including that of the decision-maker, so that in the event of emergencies all resources of the plant may focus on accident management.

Under severe accident conditions, accident management will be carried out in close communications with the State and other external parties for sharing information and receiving advice and directions, etc. For this purpose, an organization for information and communication management needs to be established.

In addition, for the effectiveness of the accident management organization, a facility (or facilities) equipped with necessary resources such as documented procedures, communication equipments, plant parameter display devices for understanding plant conditions, etc., should be established.

On the basis of the above, TEPCO reports that it has examined and developed an effective accident management framework as follows.

(1) Development of accident management organization
    1) Determine the organization to implement accident management
    2) Assignment of the roles, including the decision-maker of the accident management organization
    3) Summoning of accident management personnel
(2) Development of accident management facilities and components
    1) Organization of facilities and resources to be utilized by the supporting organization(s)
    2) Availability of instrumentation components
    3) Reporting and communications systems, etc.
(3) Development of accident management procedure manuals, etc.
(4) Provision of education/trainings related to accident management.

2.3 Key Issues of Accident Management

As shown above, accident management developed by the operators so far focused solely on internal initiators with likelihood of leading to a severe accident. Measures for the loss of key components and functions due to common cause failures on the plant level induced by external events - in particular, offsite power failure due to an earthquake; inundation of the turbine building and the unavailability (failure) of emergency diesel generator and storage battery caused by tsunami – had not been given consideration. An item, “operation/manipulation in accordance with observed plant conditions regardless of whatever the initiator” barely gives reference to both internal and external events; however, simultaneous SBO had not been included in the assumptions.

There was a strongly held belief in the low power suspension frequency in Japan compared

globally (“only a short time is required for recovery event in the event of power suspension”) and the low failure probability of diesel generator activation, which were reflected in the Safety Design Guidelines; the adverse conditions of a long-term SBO was not considered in the regulatory requirements at the time.

The 1993 report by the Working Group on Total AC Power Loss Event under the Deliberation Committee on Analysis and Evaluation of Accidents and Failures in Nuclear Installations evaluated offsite power loss frequency in Japan as 0.01/year, tenfold lower than in US, with recovery time of less than 30 minutes (in US, the median value is 30 minutes); and EDG activation failure probability of 6×10^{-4}/requirement which is lower by 2 orders than regulated in the US - these are the founding basis to the belief in the high reliability of Japanese nuclear plants. However, if earthquakes, tsunami, and other adverse conditions are given consideration, these assumptions are unfounded. The operators and the regulatory body had not anticipated an extended power loss of the Fukushima accident, as shown by the commentary for Guideline 27 “Design considerations for power failure” - “the occurrence of a long-term SBO need not to be considered since recovery of power transmission line and EDG (emergency diesel generator) should be expected’.

1) Operator’s Response

(1) Reactor Core Cooling

Events in Unit 1 highlighted the fact that the use of IC for severe accident prevention was not at all acknowledged by the operating staff. This was the starting point. The high level radioactive debris scattered by the hydrogen explosion hampered actions for preventing core melt in Unit 2 and 3.

(2) SBO

1) Considerations given to the sharing of power sources, recovery of offsite power supply and emergency diesel generator was assumed to significantly reduce severe accident probability in PSA. However, the actual sequence of events turned against these assumptions. Another shortfall was consideration not given to common cause failures because of their supposedly low occurrence frequency.
2) Although power source cars were brought into the site under SBO conditions, it could not be smoothly implemented because of the time required.
3) Extensive time was required to ensure alternative DCs (including provisional storage battery).

(3) Containment Venting

Venting was delayed because installation of temporary storage battery and air pressurizer became necessary for operating the automatic valve.

(4) Alternative Water Injection (Depressurization of reactor pressure vessel, alternative water injection pipeline)

1) Components and equipments for reactor depressurization and subsequent water injection by means including the fire extinguisher, were organized as part of accident management. Extensive time was required for depressurization because of difficulties in manipulating the safety relief valves.
2) Attempts were made in injecting fresh water into the reactors via fire engines stationed at the site, however, as the internal pressure of some of the reactors exceeded the discharge pressure of the vehicle pump, not all were successful.

The sequence of events initiated by the hydrogen leakage from the containment vessel leading to explosion in the reactor building was what no one expected.

2) Response of the Regulatory Body

The former NISA (Ministry of Economy, Trade and Industry) was responsible for evaluating, auditing and directing accident management implemented by the operators, and to report the results to the Nuclear Safety Commission. Reporting to the Nuclear Safety Commission was carried out in accordance with formalities by NISA; however, how seriously and thoroughly the examination was carried out remains a question. In addition, there are no records on the continuous monitoring of education and trainings in the regular audit reports.

The Japanese regulatory body has not been receptive to efforts in the international community, such as the continuous enhancement of severe accident management in Europe and the US, and measures developed by the US to address rapid increase of internal pressure of BWR Mark I type containment due to its small capacity under severe accident conditions, by applying these measures to the Japanese regulatory framework.

The government (cabinet office)’s involvement in the accident response and recovery process of 3.11 Fukushima Dai-ichi NPP accident was so strong that the regulatory body could not enforce its response and support activities as intended.

Functions	Accident Management after March 1994	Accident Management from initial operations
Reactor Shutdown	● Alternative reactivity control (RPT & ARI)	● Manual scramming ● Manual operation of water level control & boron injection
Water injection into reactor & containment	● Alternative water injection (injection to reactor & containment via feed-water condensate systems and fire extinguisher pumps; injection to reactor from containment cooling system via shutdown cooling system)	● Manual activation of ECCS, etc. ● Manual operation of reactor de-pressurization & low-pressure water injection ● Alternative water injection (via feed-water condensate system and control rod drive mechanism)
Containment heat removal	● Means of containment heat removal » Alternative heat removal by means of the drywell cooler, reactor water clean-up system » Recovery of containment cooling component failure »Hardened reliable vents	● Containment heat removal » Manual activation of containment cooling system » Venting via inert gas system & emergency gas processing systems.
Supporting function of safety functions	● Means of power supply » Sharing of power sources (480V from adjacent plant) » Recovery of EDG component failure » Exclusive EDG	● Means of power supply » Recovery of offsite power source & manual activation of EDG » Sharing of power source (6.9V from adjacent plant)

Table 2-1 Application of Accident Management Measures

(Unit 1, Fukushima Dai-ichi NPP)

Functions	Accident Management after March 1994	Accident Management from initial operations
Reactor Shutdown	● Alternative reactivity control (RPT & ARI)	● Manual scramming ● Manual operation of water level control & boron injection
Water injection into reactor & containment	● Alternative water injection (injection to reactor & containment via feed-water condensate systems and fire extinguisher pumps. ● Automatic reactor de-pressurization	● Manual activation of ECCS, etc. ● Manual operation of reactor de-pressurization & low-pressure water injection ● Alternative water injection (via feed-water condensate systems and control rod drive mechanism, injection to reactor and containment via seawater pumps)
Containment heat removal	● Means of containment heat removal » Alternative heat removal by means of the drywell cooler, reactor water clean-up system » Recovery of residual heat removal component failure » Hardened reliable vents	● Containment heat removal » Manual activation of containment cooling systems » Venting via inert gas systems & emergency gas processing systems.
Supporting function of safety functions	● Means of power supply » Sharing of power source (480V from adjacent plant) » Recovery of EDG component failure » Exclusive EDG	● Means of power supply » Recovery of offsite power source & manual activation of EDG » Sharing of power source (6.9V from adjacent plant)

Table 2-2 Application of Accident Management Measures

(Unit 2 – Unit 5, Fukushima Dai-ichi NPP)

Functions	Accident Management after March 1994	Accident Management from initial operations
Reactor Shutdown	● Alternative reactivity control (RPT & ARI)	● Manual scramming ● Manual operation of water level control & boron injection
Water injection into reactor & containment	● Alternative water injection (injection to reactor & containment via feed-water condensate system and fire extinguisher pumps) ● Automatic reactor de-pressurization	● Manual activation of ECCS, etc. ● Manual operation of reactor de-pressurization & low-pressure water injection ● Alternative water injection (via feed-water system & control rod drive mechanism, injection to reactor and containment via seawater pumps)
Containment heat removal	● Means of containment heat removal » Alternative heat removal by means of the drywell cooler, reactor water clean-up system » Recovery of residual heat removal component failure » Hardened reliable vents	● Containment heat removal » Manual activation of containment spray cooling systems » Venting via inert gas system & emergency gas processing systems.
Supporting function of safety functions	● Means of power supply » Sharing of power source (480V from adjacent plant; 6.9kV from exclusive DGs for high-pressure core spray system) » Recovery of EDG component failure » Exclusive EDG	● Means of power supply » Recovery of offsite power source & manual activation of EDG » Sharing of power source (6.9V from adjacent plant)

Table 2-3 Application of Accident Management Measures

(Unit 6, Fukushima Dai-ichi NPP)

3. ACCIDENT DEVELOPMENT AND KEY ISSUES OF TEPCO'S FUKUSHIMA DAI-ICHI PLANT ACCIDENT

With the occurrence of the Tohoku Region Off The Pacific Coast Earthquake, followed by catastrophic tsunami on March 11, 2011, all nuclear power plants located along the Pacific Coast in the northeastern region reached a cold shutdown, with no serious damages or functional failures. However, the tsunami that followed greatly exceeded the design basis, to which no measures had been in place. Many of the systems experienced functional failures regardless of redundant or diversified arrangements because of the tsunami (common cause), which led to loss of all power sources, then to the loss of cooling functions and heat sink, and finally to fuel damage (core melt). The event caused hydrogen explosion, inducing extensive release of radioactive material release. This is the overview on the sequence of TEPCO's Fukushima Dai-ichi Nuclear Power Plant accident (refer Fig. 3-1)

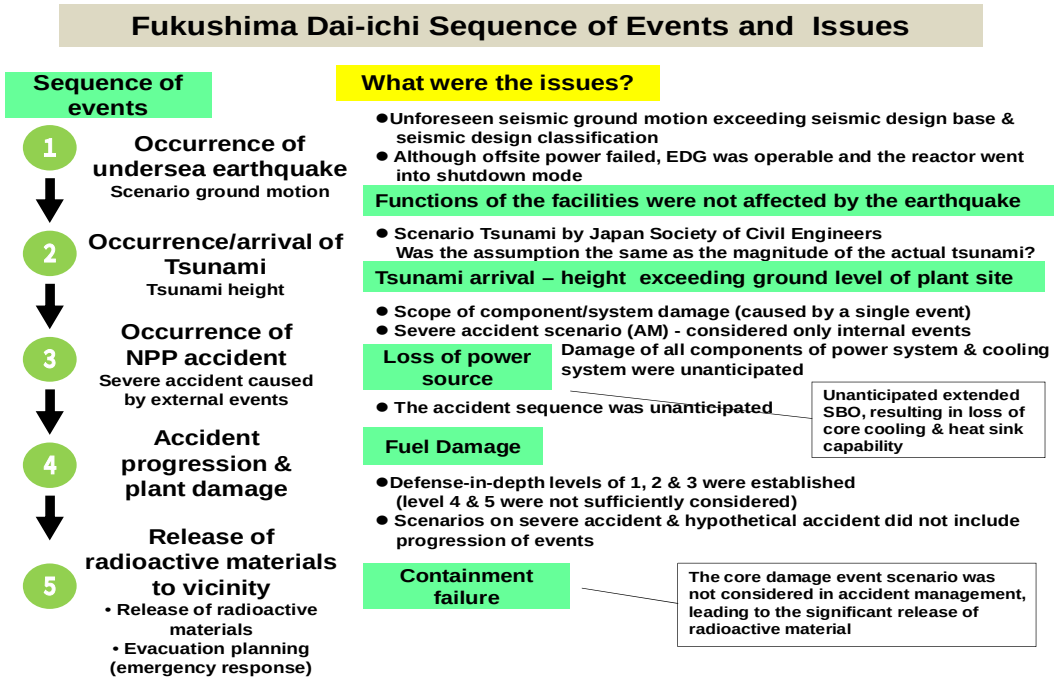


Fig. 3-1 Sequence of Events and the Issues of Fukushima Dai-ichi NPP Accident

3.1 Accident Management of Earthquakes

Design basis seismic motion and maximum tsunami scenarios have been developed through discussions by experts in the academia, and on the basis of agreement between all individuals involved in regulatory activities, experts and engineers, design standards was established and applied in safety assessments. However, the scale of the earthquake and the subsequent tsunami observed in

3.11 was far beyond the established scenarios.

At the on onset of the earthquake, control rods were inserted into the reactors of all 12 nuclear power plants in operation on the Pacific Coast and reached a cold shutdown. Although a portion of the seismic motion at Unit 2 and 3 of TEPCO's Fukushima Dai-ichi Nuclear Power Plant, and Unit 1, 2 and 3 of Onagawa Nuclear Power Plant exceeded design basis, none of the units experienced abnormalities or damages. The amplitude of the safety margin against design seismic motion of TEPCO's Kashiwazaki Kariya Plant had already been confirmed during the Chuetsu-oki Niigata Earthquake. In the 3.11 Earthquake, Onagawa Nuclear Plant situated closest to the epicenter was not affected. No safety issues were reported data-wise on the nuclear plants in Fukushima. Report by the National Diet of Japan Fukushima Nuclear Accident Independent Investigation Commission cites that although there are no records on events due to piping ruptures, the phenomena is not "unlikely". However, the Government report denied damages to the piping structure. From the results of analyses, NISA has concluded that no piping damages have occurred. **2), 9), 10)**

What highlights the series of events on 3.11 is that the design basis had been exceeded. Both Onagawa Plant and Kashiwazaki Kariya Plant have experienced beyond design basis incidents a number of times in the past. However, the SSCs had maintained integrity and no safety related events had occurred. The design seismic standards was tightened consecutively with the revision of the Seismic Design Guideline the preceding year of the Chuetsu-oki earthquake and after, due to damages to Kashiwazaki Kariya Plant by the earthquake. A new policy on back-checking was set forth for reflecting the revised safety standards on all nuclear plants in Japan. Around the same period, there was a debate on the adequacy of seismic assessment method which was based on response acceleration. Some argued that methods based on velocity, or energy rate were more appropriate for assessing "rupture". The debate had continued with no conclusion drawn up until the 3.11 incident. No one had seriously considered beyond design basis conditions and the necessary measures to this end. This section has given focus on the issue because seismic motion exceeded the design basis at both Onagawa Plant and Fukushima Dai-ichi Plant on 3.11, and beyond design basis conditions falls in the severe accident region. Design seismic basis should be reevaluated with due care and consideration given to the various aspects of seismic motion.

The consecutive crustal movement over the areas of 450 kilometers in length and 200 kilometers in width with a magnitude of 9.0 of the Tohoku Region Off The Pacific Coast Earthquake exceeded the assumptions of all seismic experts. At Okuma-cho and Futaba-machi of Fukushima Prefecture, where TEPCO's Fukushima Dai-ichi Nuclear Power Plant is located, the seismic intensity was 6.0+. Maximum seismic acceleration, parameter on earthquake intensity, has been recorded by the seismometer installed on the basement floor of the reactor building. In Units 2, 3 and 5, the maximum seismic acceleration was 550 G (cm/s^2), 507 G (cm/s^2), 548 G (cm/s^2) respectively, which

exceeded the scenario maximum response acceleration of 438G, 441G, 452G of design seismic motion Ss.

In the simulation analysis using data recorded on the seismometer, the seismic load impact on safety SSCs associated with reactor shutdown, core cooling, isolation of radioactive material, of Units 1 to 3 (in operation) and Units 4 to 6 (under shutdown mode), showed sufficient margin over seismic design basis (allowable stress, etc.). The operation records of each unit after the occurrence of the earthquake showed no abnormalities. Given the sufficient margin over capacity obtained through past assessments by comparing design values and actual plant values, it can be assumed that there was considerable margin over seismic motion at each units in TEPCO's Fukushima Dai-ichi Plant.

In particular, key safety issues to be addressed are the combined disaster caused by the earthquake, the tsunami and other associated events. At Onagawa Plant, a power panel caught fire which fortunately did not spread; however, the event implies the likelihood of combined disaster induced by earthquake and fire, which should also be given consideration.

Active Faults

One of the factors that contribute to the vagueness of seismic and tsunami assessment is the lack of effective liaison between different fields of science as physics and engineering science. This is why sharing information, opinions, and collaborating is necessary between experts in different fields and between professional societies.

Regarding the recently highlighted issue on "active faults", the seismic motion of 3.11 was induced by a distant earthquake and not associated with circumstances that directly relate to the plant. The new Regulatory Standards by the Nuclear Regulation Authority prohibits nuclear plant facilities to be installed directly above active fault lines. The following are the basic policies on seismic motion in the new regulatory standards.

«Basic Requirements»

1. For the purpose of ensuring a high safety level over the entire nuclear reactor facility (hereinafter, referred as "facility"), the following basic design policy must be satisfied.
    1) Facilities with key safety functions must be installed on grounds that have been confirmed with no outcrop of faults, etc., with likelihood of becoming active in the future.
    2) Facilities with key safety functions must be designed to maintain integrity against seismic force caused by earthquake ground motion (design basis seismic motion) with likelihood of significantly impacting facilities which is rare, but may occur during the service period of the facilities. In addition, the facilities must be designed to fully sustain adequate integrity

against seismic force on the basis of significance with regard to safety in the event of the loss of safety functions due to an earthquake and the subsequent of release of radioactivity to the environment.

3) Facilities must be installed on grounds with sufficient capacity against seismic force described in the preceding clause.

4) Facilities with key safety functions must be designed to maintain integrity against tsunami events (hereinafter, referred as “design basis tsunami”), with the likelihood of significantly impacting facilities which is rare, but may occur during the service period of the facilities.

2. The methods on the survey for developing design basis seismic motion and design basis tsunami should be selected with consideration given to the conditions of the application and its accuracy, etc., to ensure reliability and accuracy of the results.

Regarding 1), “Facilities with key safety functions must be installed on grounds that have been confirmed with no outcrop of faults, etc. with likelihood of becoming active in the future”, prohibits the construction or installation of nuclear facilities above active faults. However, since whether a fault is active, or not is difficult to determine and varies by the judgment of each expert, it may sometimes give rise to futile debates. With view to the accumulated studies on movement of the crust and movement of structures corresponding to fault movement, or vibration response analysis in recent years, “facilities with key safety functions must undergo seismic safety assessment with account taken on faults with the likelihood of becoming active in the future. If appropriate assessment may not be carried out, the facilities shall not be installed on grounds that have been confirmed with outcrop of faults, etc.” The practice in implementing adequate assessment will contribute to the future of technology development.

3.2 Accident Management of Tsunami and Its Influences

The maximum design tsunami values to be applied to nuclear power plants have been debated by the academia and professional societies. The tsunami assessment technique has been examined and formulated mainly by the Society of Civil Engineers and re-established through the application of state-of-the-art computing technologies. However, the difficulties in predicting natural hazards and preventing disasters is as evidenced by the tsunami that overran the tide embankment of Taro-cho, Iwate Prefecture and wiped off the small town. Most natural disasters that take place are unanticipated [11)].

The maximum tsunami that struck nuclear power plants on 3.11 was far beyond expectations. The unprecedented seismic ground motion magnified the scale of the tsunami, extremely complex and beyond the scope of tsunami assessment framework. This resulted in the damages to the plant, and the subsequent severe accident.

The size of the tsunami experienced by all of the nuclear power plants exceeded design assumptions. However, some of the sites had sufficient safety margin and was not affected. The size of the tsunami at plants including Onagawa, Fukushima Dai-ichi, Fukushima Dai-ni, and Tokai Dai-ni all exceeded not only the regulated standards, but also the most recently revised design values. The tsunami height at Onagawa Plant was 13 meters, and the site's ground height which subsided by 1 meter to 13.8 meters barely missed being damaged by the tsunami. At Tokai Dai-ni Plant, the cooling components and facilities in which tsunami water-proofing had just been completed maintained integrity and the reactors successfully reached a cold shutdown. At Fukushima Dai-ni Plant, although the tsunami height itself was 8 meters against the site's ground height of 12 meters, inundation height of 14.5 meters caused damages to many of the components. However, accident management arrangement made in advance was effective and the reactors successfully reached a cold shutdown. At Fukushima Dai-ichi Plant, the scale of the tsunami was totally unanticipated which exceeded even the recently revised values that was based on new technological expertise.

On January 11, 2011, Long-term Assessment Committee, Headquarters for Earthquake Research Promotion (Ministry of Education, Culture, Sports, Science and Technology), announced assumptions on 99% occurrence probability of M.7.5 (approx.) Off-the-Coast Miyagi Prefecture Earthquake (fault slip amplitude of 16 meters; M.8.0 if the earthquake conjuncts with the region off-the-coast southern Sanriku near the ocean trench) within the next 30 years. In the case of consecutive occurrence of Nankai and To-Nankai Earthquakes, the magnitude was assumed at approximately M. 8.5. The generally recognized tsunami assessment method in Japan before the 3.11 incident was "Tsunami Assessment Method for Nuclear Power Plants in Japan" by the Japan Society of Civil Engineers. At all nuclear power plants, re-assessment on tsunami height had been made on the basis of maximum earthquake evaluated by this method.

On March 11, 2011, the consecutive large-scale earthquake that occurred off-the-coast of Iwate, Miyagi, Fukushima, and Ibaragi Prefectures with a magnitude of M. 9.0 (the areas of 450 kilometers by 200 kilometers, with maximum slip amplitude of 60–70 meters) triggered an unanticipated massive tsunami. The size of the earthquake and tsunami was comparable in magnitude to the 869 Jogan Sanriku-oki Earthquake, and recognized as "once-in-a-thousand-years" event, which shows the limitations on predictive assessment on earthquake and tsunami events and their magnitude.

Due to the 15 meters beyond-design-basis tsunami height resulting from overlapping of multiple tsunami waves (constructive interference) at Fukushima Dai-ichi Plant, the emergency power supply components in Units 1 to 6 failed except for the following components - 125V DC power supply components installed on the middle basement floor of the turbine buildings of Units 3, 5, and 6, and air-cooled emergency diesel generator installed in the reactor building located at the highest ground level of 13 meters in Unit 6. The slight difference in height made a big difference on the functional integrity of the components, which is one of the important lessons learned from the incident.

The installation of emergency power supply components as emergency diesel generators, DC power sources, power boards in the basement floor of the turbine power generating room, which had no watertightness became the direct cause of the severe accident, inducing loss of functions of these equipments and consequently to SBO.

3.3 Accident Management of Beyond Design Basis Events

In Japan, five beyond design basis earthquakes have so far occurred, including Chuetsu-oki Earthquake. In response, the Seismic Design Guideline was revised in 2006, incorporating the likelihood of beyond design basis seismic conditions as residual risk, with requirements to reduce such risk through relevant measures. Each plant developed measures against beyond design basis events, and in some cases, as with Unit 1 and 2 of Hamaoka Nuclear Power Plant, decision was made on decommissioning on the basis of judgment including the benefit and cost.

Unfortunately, no specific design values had been established with regard to tsunami, and accordingly, no specific measures had been formulated against design basis, nor beyond design basis tsunami events. Not only the absence of regulatory requirements on tsunami, but the concept on design tsunami values itself had not been organized, and accordingly, there were no specific tsunami preparedness and response measures in place.

Performance objectives, such as CDF (core damage frequency) of 10^{-4}/reactor-year, and CFF (containment failure frequency) of 10^{-5}/reactor-year specified in the report by the former Nuclear Safety Commission are given as safety objectives for maintaining fatality rate at 10^{-6}/people-year of the site boundaries under accident conditions. The objectives are cited as follows: “the mean value of acute fatality risk by radiation exposure resulting from a nuclear facility accident of individuals of the public in the vicinity of the site boundary of the nuclear installation shall not exceed the probability of approximately $1x10^{-6}$ per year; and the mean value of fatality risk by cancer caused by radiation exposure resulting from a nuclear facility accident of individuals of the public residing in the area, but with some distance from the facility, should not exceed the probability of approximately $1x10^{-6}$ per year.” Risk assessments (PSA/PRA) conducted so far have not taken into account events with occurrence frequency below 10^{-7}/reactor-year.

Further, the fact that simultaneous functional loss of multiple units and common cause failure/accident have not been given consideration, nor measures developed to this end in accident management are issues that must be addressed in the future. With the arrival of the tsunami, many of the key safety components, including those with redundancy failed all at the same time in multiple units of Fukushima Dai-ichi Nuclear Plant. Factors that overlapped to promote sequence of events leading to the severe accident at TEPCO’s Fukushima Dai-ichi Plant were 1) SBO; 2) loss of cooling systems; and 3) loss of heat sink. Vulnerabilities in accident management were: 1) inadequate alternative power supply; 2) insufficient alternative pump capabilities; and 3) deficiencies in

preparedness against unanticipated events, including hydrogen explosion, containment rupture, SBO, etc.

The root cause for these events/factors may be attributed to the lack of a fundamental approach on accident management and management of severe accidents – in anticipating the likelihood of risks, against which scenarios should have been developed. Up to present, accident sequences have been developed on the basis of internal events initiated by a single failure of constituting components in severe accident management, for which measures would quantitatively ensure plant safety. Damage to multiple units causing simultaneous failures of components having the same functions, or common cause failures were considered as low probability events in previous assessments, which led to the poor accident management at Fukushima Dai-ichi Plant.

A thorough understanding and knowledge on accident sequences is essential in accident management. Vulnerabilities in the understanding on fuel damage and containment damage sequences, and the absence of appropriate preparedness and response measures against these events were shown in the Fukushima Accident. With Unit 1 for example, prioritizing the integrity of containment isolation function over mitigation and control measures such as core cooling via the IC and the manual operation of the valves (opening/closing) for venting led to the negative turn of events.[12)]

In the management of beyond design basis events (including those by terrorist attacks), measures should be developed for natural disasters of not only earthquakes and tsunamis but for other initiators, including how and when to take action.

3.4 Accident Management of SBO and Its Influences

TEPCO's Fukushima Dai-ichi Plant was equipped with 2 emergency diesel generators at each unit (Unit 6 had 3 components), in addition to 3 air-cooled emergency diesel generators. The emergency AC bus (480V) was connected to the DC battery charger (125V) between adjacent units (between Unit 1 and 2, Unit 3 and 4, and Unit 5 and 6) to maintain continuous use of the storage battery. The reinforcement of DC power supply enabled manual activation of emergency diesel generator after recovery from failure, as well as continuous operation of IC in Unit 1, RCIC in Units 2 to 6, and HPCI in Units 1 to 6.

Had the maximum tsunami height off the coast of Fukushima Dai-ichi Plant been below 10 meters, the emergency diesel generators installed in the turbine building, etc., at ground heights of 10 meters or 13 meters would have been available. However, the tsunami height exceeded 15 meters, and of the emergency power supply components installed before the accident, the only functional were 125V DC power supply equipments installed in the middle basement floor of the turbine buildings of Units 3, 5,and 6, and air-cooled emergency diesel generator installed in the reactor building located at ground height of 13 meters in Unit 6. The inundation of power supply components and power

systems was what no one expected. Had distributed arrangements and inundation measures such as pressure-resistant and water-resistant doors, etc., and installation of emergency power source equipments to higher grounds been implemented, the worst case scenario could have been avoided.

Immediately after TEPCO's Fukushima Dai-ichi Plant accident, NISA issued a directive on measures related to emergency onsite electric power components applicable to both BWR and PWR plants, in the event of not only earthquake and tsunami, but fire, explosion, typhoons, etc., as well. The installation of large size emergency generators and power supply cars on higher grounds in the directive have started. The results of technical assessment on TEPCO's Fukushima Dai-ichi accident calling for enhancing reliability of power supply systems, seismic integrity of substations and switching stations, and the quick recovery of related safety components, are being carried out by each nuclear power plants.

3.5 Hydrogen Explosion and Its Influences

1) Overview of the Explosion

The hydrogen explosion that occurred in TEPCO's Fukushima Dai-ichi Plant was caused by high temperature of the fuel cladding and zirconium fuel cladding-water reaction producing hydrogen, which leaked into the containment vessel and subsequently to the reactor building, and assumed to have exploded, reaching the explosive limit. The explosion in the reactor buildings of Unit 1 and Unit 3 are assumed to have been caused by the hydrogen gas in each unit. However, the explosion in Unit 4 is understood to be induced by the inflow of hydrogen generated in Unit 3 through emergency gas treatment pipe shared between Unit 3 and Unit 4. There was no hydrogen explosion in Unit 2, although core damage is assumed to have occurred as with other units. This was because the blowout panel of the reactor building was open, which released hydrogen to the exterior of the turbine building.

2) Causes of Hydrogen Leakage

The entire hydrogen outflow route from the reactor into the reactor building that led to the explosion has not been clarified. Hydrogen generated in the pressure vessel is assumed to have leaked into the containment through the melted joints of the control rods and incore monitor guide tube, damaged by high-temperature fuel debris falling to the bottom. The degradation of silicon rubber seals of the containment from high temperature and atmospheric pressure increase led to the leakage of the atmosphere containing hydrogen. The assumed outflow routes from the containment are:

- The connecting parts of the upper lid of the containment vessel
- The connecting part of the hatch for personnel and component entry/exit
- Electrical wiring penetration

3) Influences of the Explosion

Hydrogen explosion outside the containment was unanticipated, and the event in Unit 1 significantly affected the plant's recovery process. Debris scattered by the explosion damaged the power cables laid out for connection to the power board in Unit 2, and significantly delayed the recovery process. The power cables for seawater intake pipeline and boric acid injection system in Unit 1 was also damaged by the explosion. The incident was aired on TV, etc. in full and the psychological impact on the public was immense.

4) Lessons Learned

The hydrogen explosion at TEPCO's Fukushima Dai-ichi Plant not only tremendously affected the recovery process, but exerted immeasurable influence on the public. Although a rough sequence of the event is now becoming clear, a detailed mechanism of the explosion needs to be clarified. Measures should be taken to prevent retention of hydrogen inside plant facilities. Provisions should be made to enable operation of safety components even under SBO conditions, such as manipulating the opening of the top vents and blowout panels installed at the top of the reactor building, and flammable gas control equipments, etc. The mechanism on hydrogen retention that led to the explosion should be clarified so that appropriate measures and arrangement of relevant components in the reactor building and the containment vessel may be examined for each plant. In other words, accident management including selection and installation of appropriate components to control hydrogen generation and the organization of management procedures with consideration given to plant specific conditions, as well as development of personnel with competencies in managing hydrogen leakage from the containment should be established.

3.6 Why the Severe Accident was not Preventable

The issues on "what actions should have been taken in advance to prevent TEPCO's Fukushima Da-ichi accident" and "what measures should have been in place to prevent it?" are discussed in details in **Section 2.3** and **Section 3.3**. To summarize:

1) Preparedness for Severe Accidents

As in Europe and the US, examination of severe accident measures in Japan was initiated in the wake of the Three-Mile Island Accident by the former Nuclear Safety Commission, in liaison with the OECD/NEA (participated by advanced nuclear nations at the time).

The Nuclear Safety Commission made decisions after the examination, on encouraging voluntary initiatives of the operators to independently establish and to implement accident management (non-regulated). The decision at the time was in line with the global trends. However, many countries in the nuclear community later changed to take a regulated approach. Although the initiating event of TEPCO's Fukushima Dai-ichi Accident was the tsunami, actions that may have contributed to preventing the accident in advance are as follows:

(1) Severe accident management should have been regulated and not left to the voluntary discretion of the operators. The operators would need to allocate financial resources including the installation of safety components in keeping with the regulatory requirement. From the beginning, severe accident management should have included not only internal events (as single failure of components and errors in operator manipulation) but also external events (as earthquakes and tsunami), with measures developed for preventing or mitigating each event. There was also a lack of defense-in-depth approach in the development of measures against natural hazards involving residual risk or uncertainties.

(2) Given the flooding of the seawater pump at Madras Nuclear Plant, India caused by tsunami off the coast of Sumatra in 2004, the Nuclear Safety Commission made a trial calculation [13)] in 2008 on tsunami run-up height which was 15.7 meters by assuming wave source in the ocean trench off the coast of Fukushima Prefecture. Development of tsunami measures seems to have been considered to some extent, however, it was never formulated. Previously at TEPCO, two emergency diesel generators installed in the basement floor of the turbine building became inoperable due to inundation caused by seawater cooling pipe leakage. The lessons on the incident should have been utilized in severe accident management and tsunami measures much earlier, such as changes in the arrangement of emergency diesel generator and the installation of additional gas turbine generators to different locations.

It was unfortunate that regardless of the very high core damage frequency (CDF) obtained through core melt frequency risk assessment conducted by Japan Nuclear Energy Safety Organization in 2006 by applying the inundation accident case of Blayais Nuclear Plant (France) to Unit 1 of TEPCO's Fukushima Dai-ichi Plant, the regulatory body did not take necessary actions. This is an issue to be addressed in the future.

(3) Japan has made loose assumptions on SBO conditions – there was a strongly held belief that power suspension frequency is low; that only a short time is required for recovery in the event of suspension; and that the failure rate of activating the emergency diesel generator is extremely low in Japan. This is reflected in the Safety Design Guidelines as "long-term (30 min) SBO need not be considered". Even after Station Blackout and Advanced Accident Mitigation (B.5.b) was issued after 9.11 terrorist incident by the US NRC, requiring provision of safeguards and trainings for SBO, no actions were taken in response in Japan although this was conveyed to the Japanese regulatory body[2)]. An attitude of learning from international practices by introducing portable power supply components and trainings as the US had done should have been taken.

The regulatory body and the operators should have liaised in evaluating and verifying the applicability of the international practices to Japan circumstances. Had such actions been taken, the severe accident of 3.11 could have been prevented.

2) Preparedness and Response Measures

With view to response actions against the Fukushima accident:

(1) Provision of thorough education, trainings, drills on emergency preparedness and response measures for the supervisor, operating staff, etc., is a prerequisite. The major cause that led to the core melt in Unit 1 was the inadequacies in the understanding and trainings related to the IC. Whether sufficient communication had been established between the manufacturers and the operator on the changes in design specifications and safety standards from those of the original standards is doubtful. The delayed containment gas venting was a major factor that promoted adverse circumstances, which led to the hydrogen explosion and to extensive radioactive material release.

As described in Section 2, accident management mandates provision of education and trainings on a regular basis without omission. However, as shown by TEPCO's statement[13)], the circumstance was "inadequate training programs and drills with no substance based on the loose assumption that severe accidents are not likely to occur. Similarly, there was a shortage on the stockpile of necessary resources". This typically reflects Japan's nuclear establishment steeped in "safety myth", which should be eliminated. An earnest approach in preventing severe accidents must be taken by all parties involved in nuclear power.

(2) Appropriate measures to control progression of events were not taken due to the lack of understanding on accident sequences including time span from fuel damage to core melt, hydrogen generation, melt through of the reactor pressure vessel caused by fuel debris, temperature and pressure increase of the containment, and subsequent release of hydrogen and radioactive material to the reactor building. This may be attributed to the fact that in recent years researches on severe accident sequences have not been active because the area of light-water reactor technology was assumed as completed. However, thorough and intensive education and training on severe accident sequences should be provided for those involved in the operation and supervision of nuclear reactors.

(3) The regulatory body fell short of directing accident management processes because they did not commit themselves in accumulating expertise and understanding on severe accident sequences and in establishing adequate response measures, which is a virtual neglect of duties.

As such, if necessary resources and materials had been arranged and trainings on emergency preparedness and response provided on the basis of expectations that severe accidents do occur, the accident would have been adequately controlled.

4. FUNDAMENTAL CONCEPT ON NUCLEAR SAFETY

4.1 Nuclear Safety Objectives and Fundamental Safety Principles

Nuclear safety is defined under the societal context because there is no end point in ensuring safety. The level of safety will depend on the consensus established in society on what level of safety is acceptable. The meaning and significance of nuclear safety objectives and fundamental safety principles is closely tied to the values and consensus on safety established in society. Severe accident management had been left to the voluntary discretion of the operators, and defense-in-depth-based approach in severe accident management and emergency preparedness and response measures had not been established in nuclear plants in Japan. The Fukushima Dai-ichi Plant accident highlighted the various issues on emergency preparedness that should have been addressed, which has been pointed out strongly by various organizations [14)].

What were the factors that induced the severe consequences?

It was the insufficient understanding on the system, organization, framework, and the interactions between these elements necessary for fulfilling responsibility for safety; secondly, lack of understanding and resolve in utilizing nuclear power and the risks involved; and thirdly, lack of focus in the application of the defense-in-depth concept, of all individuals and organizations involved in nuclear power generation.

On the basis of the lessons learned, the organizations in public and private sectors in Japan must jointly strive to achieve the highest standards of safety in nuclear power generation. To this end, nuclear safety codes and standards for the enhancement of nuclear power generating facilities should be established.

Was there a shared perception of “responsibility”? “Sekimu” or responsibility, including obligations associated with assigned roles should be given consideration. Nuclear power generation has been promoted as part of the state policy in securing energy sources - ensuring nuclear safety is not only the responsibility of the operators. The local site is at the forefront of ensuring safety, and has the prime responsibility for ensuring safety. The operators must not only adhere to the rules, but are expected to make the best achievable efforts in ensuring safety. Whereby, the government has established various regulatory rules on the design and operation of nuclear facilities as the bases for submitting applications, conducting safety assessments and granting operating licenses. A framework and rules for ensuring safety of nuclear power plants has been established for issuing construction and operation permits. The government is responsible in this respect, which should also be taken into account.

Although manufacturers are in a position to receive orders from the operators on design, manufacturing and construction processes, they are held liable for safety design, quality assurance of products, and are responsible for ensuring safety as part of “product liability”.

Responsibility for the 3.11 accident does not rest solely on the operators. It is essential for all stakeholders in the nuclear power generation community, including the government (the regulatory body), the utilities, manufacturers, the academia, the local governments, etc., to recognize responsibilities for ensuring safety in the event of emergencies commensurate with the assigned roles. In addition, the involvement of the mass media and the public in the process perhaps should be given some thought.

On the basis of the lessons learned, investigation, analysis and assessment on the accident should be conducted, the results of which should be utilized for future measures. Subsequently, regulatory standards and other related rules that define the requirements and the approach to nuclear safety should be reevaluated for establishing appropriate operational and management framework.

"Nuclear Safety Objectives and Fundamental Safety Principles" has been established by the Atomic Energy Society of Japan, recommending a shared understanding of "fundamental concept on nuclear safety" for promoting peaceful uses of nuclear energy and nuclear power generation on the basis of lessons learned from TEPCO's Fukushima Dai-ichi Plant accident. The IAEA (International Atomic Energy Agency)'s "Fundamental Safety Objectives and Safety Principles" was referred to in the formulation and combined with the circumstances in Japan and the lessons learned from TEPCO's Fukushima Dai-ichi Plant accident. Nuclear power generation is not simply an alternative to thermal power generation. Radiation risks that transcend national borders are greater than the benefits gained through nuclear power generation, and thus, a common basis for sharing understanding on the concept in ensuring safety is very important.

4.2 Defense-in-Depth Concept

All activities involve risk. The humankind has taken the pathway in utilizing enormous amount of energy generated through nuclear chain reaction for peaceful purposes by means of nuclear power generation. Nuclear chain reaction and nuclear power generation involve various risks. Significant risk associated with nuclear power generation is decay heat generated by nuclear fission products and radiation. The understanding on "radiation risk" and the necessity for the safety management of radiation has been shared globally, and regulated by international organizations such as the IAEA and ICRP (International Commission on Radiological Protection) and controlled by each State. "Radiation risk" associated with nuclear power generation has been thoroughly examined on the basis of extensive scenarios. "Nuclear safety", the fundamental concept of which is defense-in-depth[5), 15)], a protective strategy based on multiple and redundant layers of physical barriers against risk (refer Fig. 4-1), has been established for the elimination of radiation risk.

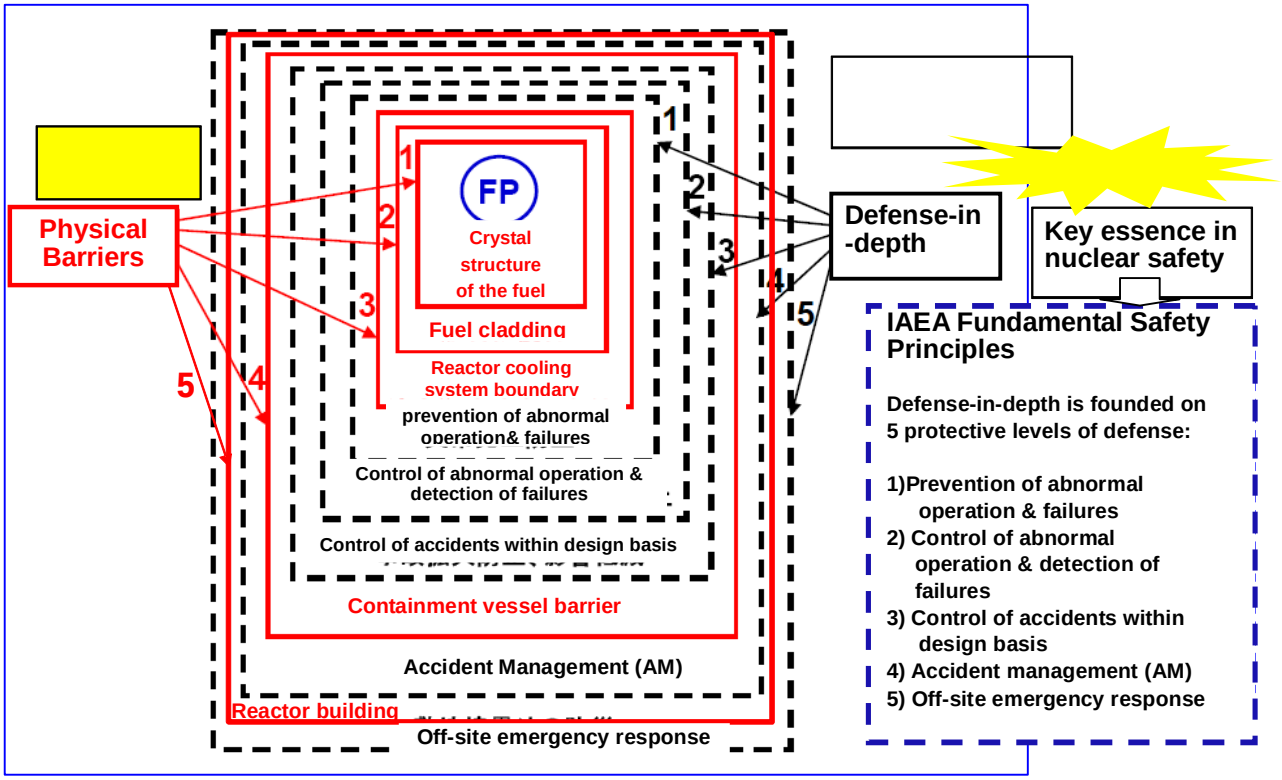


Fig. 4-1 Physical Barriers and Defense-in-Depth[12] Against Radioactive Material Release (Partial revision of Reference 12)

The "Nuclear Safety Objectives and Fundamental Safety Principles" determines that "all practically possible efforts must be made to prevent nuclear and radiation incidents and mitigate their impacts" ("Principle 8: Prevention and Mitigation of Accidents"). In the Principles, defense-in-depth is defined as the key means for preventing and mitigating accidents. Defense-in-depth is basically a concept on "preventing accidents", "controlling escalation to serious consequences", and "preventing harmful consequences of accidents to the public". In Japan, the term was formerly referred to as "redundant defense"[1)] by the former Nuclear Safety Commission, and evolve on three levels of defense for nuclear reactor facilities – 1) prevention of abnormal operation and failures; 2) control of abnormal operation and prevention of escalation to accidents; and 3) prevention of abnormal release of radioactive material.

The IAEA standards defines defense-in-depth on the basis of 5 levels of safety barriers including control of severe plant conditions, and mitigation of radiological consequences arising from extensive release of radioactive material, etc.

In Japan, the former Nuclear Safety Commission conducted a review to organize the concept of defense-in-depth, and the former Nuclear and Industrial Safety Agency initiated the organization of the concept as part of severe accident management review, which is currently being undertaken by

the Nuclear Regulation Authority.

A few documents exist that define or describe concept on “defense-in-depth”, which include the recent publications by the IAEA, “INSAG-10: Defence in Depth in Nuclear Safety” and “IAEA SAFETY STANDARDS SSR-2/1 Safety of Nuclear Power Plants: Design Safety Requirements”.

The IAEA defines defense-in-depth as “a hierarchical deployment of different levels of equipment and procedures in order to maintain the effectiveness of physical barriers placed between radioactive materials and workers, the public or the environment, in normal operation, anticipated operational occurrences and in accidents at the plant.” The objectives of defense-in-depth are: (a) to compensate for potential human and component failures; (b) to maintain effectiveness of the barriers by averting damage to the plant and to the barriers themselves; (c) to protect workers, the public and the environment from harm in the event that these barriers are not fully effective.

Similarly, US NRC defines defense-in-depth as “the approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures,” which is basically the same as defined by the IAEA. In the NRC News No. S-04-009 (June 3, 2004), “The Best-Laid Plans”, shows the understanding of defense-in-depth as the integration of safety, security and emergency response.

1) Necessity of Defense-in-Depth

The concept of ‘defense-in-depth’ was formulated initially as “elastic defense”, a military strategy, and seeks delay rather than prevent the advance of an attacker to buy time and cause additional casualties in exchange for yielding of larger territory by the attacker.

Currently, it is widely used for non-military situations. In nuclear plants, large amounts of radioactive material residing in the reactor involve potential radiation risk consequences to people and the environment which must be controlled at all costs. For this purpose, the concept is used as an active arrangement (strategy) in protecting people and the environment.

If a single safety measure ensures the full protection of people and the environment, then no additional measures are required. However, initiating events and events that arise in the sequences leading to the release of radioactive material in the atmosphere with harmful consequences on people and the environment contain uncertainties and unanticipated conditions. Since safety measures are generally laid out on the basis of specific assumptions, to the exclusion of other possible scenarios or unexpected events, the effectiveness of the measures contain uncertainties and are not 100% fail-safe.

Thus, graded protection against uncertainties contained in a single safety measure needs to be

provided by other measures to enhance reliability in preventing consequences of risk to people and the environment, which is called the defense-in-depth concept.

US NRC determines the goal of defense-in-depth as arrangement against uncertainties in NUREG-1860. Defense-in-depth has been determined as an element in NRC's safety philosophy that is used to address uncertainties by means of successive measures including safety margins to prevent and mitigate damage in the event of abnormalities or accidents, etc., at nuclear facilities.

2) Fundamental Concept of Defense-in-Depth

The fundamental concept on defense-in-depth, or 'redundant defense' (term that was generally used in Japan), is founded on independent, multiple layers of defense for all activities related to safety, which will detect, compensate, or correct with appropriate measures in the event of failures or malfunctions. Graded levels of protection are provided so that in the event of a failure of one level, other levels are available to ensure safety[2].

The fundamental element of defense-in-depth is the independent effectiveness of different levels of defense, so that if one level fails, other, or the subsequent level of safety are not affected and will be available.

(1) By applying defense-in-depth concept to the entire realm of safety activities, including organization, behavior, design and operation, protection will be provided against anticipated events and accidents during operation that include external and internal events, such as SSC failures and human events.

(2) System design based on defense-in-depth includes process management that limits the acceptable level of failure through feedbacks. Physical barriers are protected by maintaining plant operation parameters within a clearly defined range. Discreetly designed system will prevent 'cliff edge effect', where an extremely abnormal plant behavior is triggered by a small deviation leading to damage.

3) Defense-in-Depth Levels of IAEA [3]

(1) The IAEA has applied the concept in the design of nuclear power plants with a goal to prevent harmful consequences of radiation to people and the environment, to provide protection against and mitigate harmful consequences, and determined the following five levels of defense (defense-in-depth-specific functions, SSCs and procedures).

Table 4-1 shows the goals of each defense-in-depth level and essential means for achieving the goals.

(2) Plant conditions given consideration in the design are roughly classified into "operating conditions" and "accident conditions", with the former sub-classified as "normal operation" and "anticipated transients", and the latter as "design basis accident" and "design extension conditions (DEC)"[4]. Defense measures for the four operating conditions correspond to

defense-in-depth levels 1 to 4.

	Defense-in-Depth Level	Goal	Essential Means	Related Plant Conditions
Design Basis	Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction & operation	Normal operation
	Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems, and other surveillance features	Transient condition to abnormal state (Anticipated Operational Occurrences, AOO)
	Level 3	Control of accidents within design basis	Engineered safety features and accident management procedures	Design basis event (A single, anticipated initiating event)
Beyond Design Basis	Level 4	Control of severe conditions including prevention of accident progression & mitigation of severe accident consequences	Complementary measures & accident management including defense of containment vessel	Redundancy failures Severe accident Design extension conditions
Emergency Response	Level 5	Mitigation of radiological consequences of significant release of radioactive materials	Off-site emergency Response	Disaster prevention

Table 4-1 Defense-in-Depth Levels of IAEA

1) Level 1 is oriented towards the prevention of abnormal operation and failures. Appropriate quality level and engineered safety features (e.g., application of redundancy, independence and diversity) are incorporated for a sound and conservative design, construction, maintenance and operation of nuclear plants.
2) Level 2 is aimed at the control of abnormal operation and detection of failures. Deviations are detected and prevented to inhibit any abnormal development from anticipated events during operation.
3) Level 3 provides control over design basis accidents. In the event of failure of Level 2 in preventing development of AOO (anticipated operational occurrences) and anticipated initiating events, Level 3 provides control over sequence to severe consequences (accidents) and ensures safety shutdown.
4) Level 4 ensures control of severe plant conditions, including accident development and mitigation of severe accidents, protection of confinement, as well as ensures that radioactive release is kept as low as achievable.

5) Level 5 covers functions in mitigation of radiological consequences of significant release of radioactive material, which requires emergency centers with appropriate equipments and on-site and off-site emergency response plans.

4) Defense-in-Depth Measures and Safety Assessment

(1) Defense-in-Depth Measures

1) Safety Classification and Defense-in-Depth measure

In order for the SSCs related to defense measures to fulfill requirements on safety, they should be classified on the basis of their function and significance with regard to safety. Up to present, the SSCs (safety functions) have been classified in the order of safety significance for each levels of defense (separate classification). However, the classification should correspond to the entirety of the defense-in-depth framework. The following factors should be given consideration in the classification: safety function to be performed by the component; consequences of failure to perform a function; frequency that the component will be required to perform a safety function; period throughout which the component will be required to operate, etc.

For beyond design basis conditions, SSCs corresponding to measures for level 4 defense-in-depth should fully ensure protection.

Measures for the achievement of defense levels may be carried out by either permanent facilities or transportable equipments, or by combining both, so far as they are reliable and fulfill safety functions. In general, defense measures for design basis events are implemented by appropriately operated (the availability of appropriate operating procedures and adherence to such procedures) permanent facilities that guarantees reliability and performance. For beyond design basis conditions, permanent facilities (mainly with newly constructed plants) or transportable equipments will be flexibly arranged to ensure reliability and performance of the measures depending on the viability of the arrangement of permanent facilities and other circumstances. If transportable equipments (ensures performance but not reliability) need to be set up, additional measures to ensure reliability should be arranged. The arrangement and process in warranting reliability is called accident management procedure. In case human manipulation is required, account must be taken on the uncertainties associated with human factors (HF).

2) Ensuring reliability of measures against beyond design basis conditions

Although single-failure criterion need not be applied in determining reliability or quality required of the SSCs with safety functions aimed at controlling beyond design basis conditions, design solutions as redundancy or diversity may be applied to ensure reliability commensurate with the safety significance of the SSCs. Environment conditions under

severe accidents should also be taken into account in determining reliability or quality (performance) required of the safety functions.

3) Ensuring reliability of human operation

Practical and effective defense measures against beyond design basis conditions in the spectrum of level 4 defense-in-depth is ensured by not only quality (performance) and reliability of the safety facilities, but appropriate operator manipulation based on operating procedures, as well as human competence in flexibly dealing with unexpected occurrences. Accordingly, a manual in dealing with various circumstances under severe accident conditions should be developed, as well as education and trainings provided regularly for developing competencies to flexibly deal with each circumstance. The effectiveness and reliability of protective measures should be evaluated on the combined performances of human operation and SSCs through drills and simulation trainings.

(2) Safety Assessments

1) In the safety assessment, a series of concurrent failures of redundant elements will be given for assessing the effectiveness in preventing beyond design basis conditions, or severe accidents (refers to "prevention of severe core damage" of defense-in-depth level 4 in Table 4-1), and a series of accident conditions (severe accident conditions) will be given to appropriately assess containment vessel load. Measures in the spectrum of defense-in-depth level 4 will be assessed on whether they fulfill the criteria in their effectiveness for beyond design basis conditions through deterministic, probabilistic methods and engineering judgment.

2) Optimal forecasting technique may also be used in the assessment. Conservative assessment will be applied if the first event, event progression, assessment model and input data contains significant uncertainties against the judgment criteria. When operator manipulation is considered in the assessment using a deterministic approach, it should be highly reliable and practical.

4.3 Fundamental Concept on Design Basis and Measures For Design Basis Events and Beyond Design Basis Events [16)]

Plant design has so far been founded on design basis events in the scope of defense-in-depth levels 1 to 3. Key safety components have been established with redundancy and diversity to maintain integrity against various design basis events. Design considerations against external events have so far focused on earthquake ground motion, in which beyond design basis conditions have been dealt with conservatively on the basis of past records and scientific simulations, allowing sufficient safety margin to retain structural integrity under minor beyond design basis conditions.

The need to establish preparedness and response measures against beyond design basis conditions and to re-examine design basis standards has been recognized through a number of beyond design basis earthquakes in the past. However, because these earthquakes events had not led to jeopardizing

the integrity of nuclear plant operation and its facilities, verifying sufficient margin of SSCs, it may have generated the delusion that nuclear facilities have sufficient margin over all natural disasters. There seems to have been a strongly held belief that the design basis of nuclear facilities was effective against all events, including beyond design basis events, discounting the likelihood of circumstances leading to severe accidents.

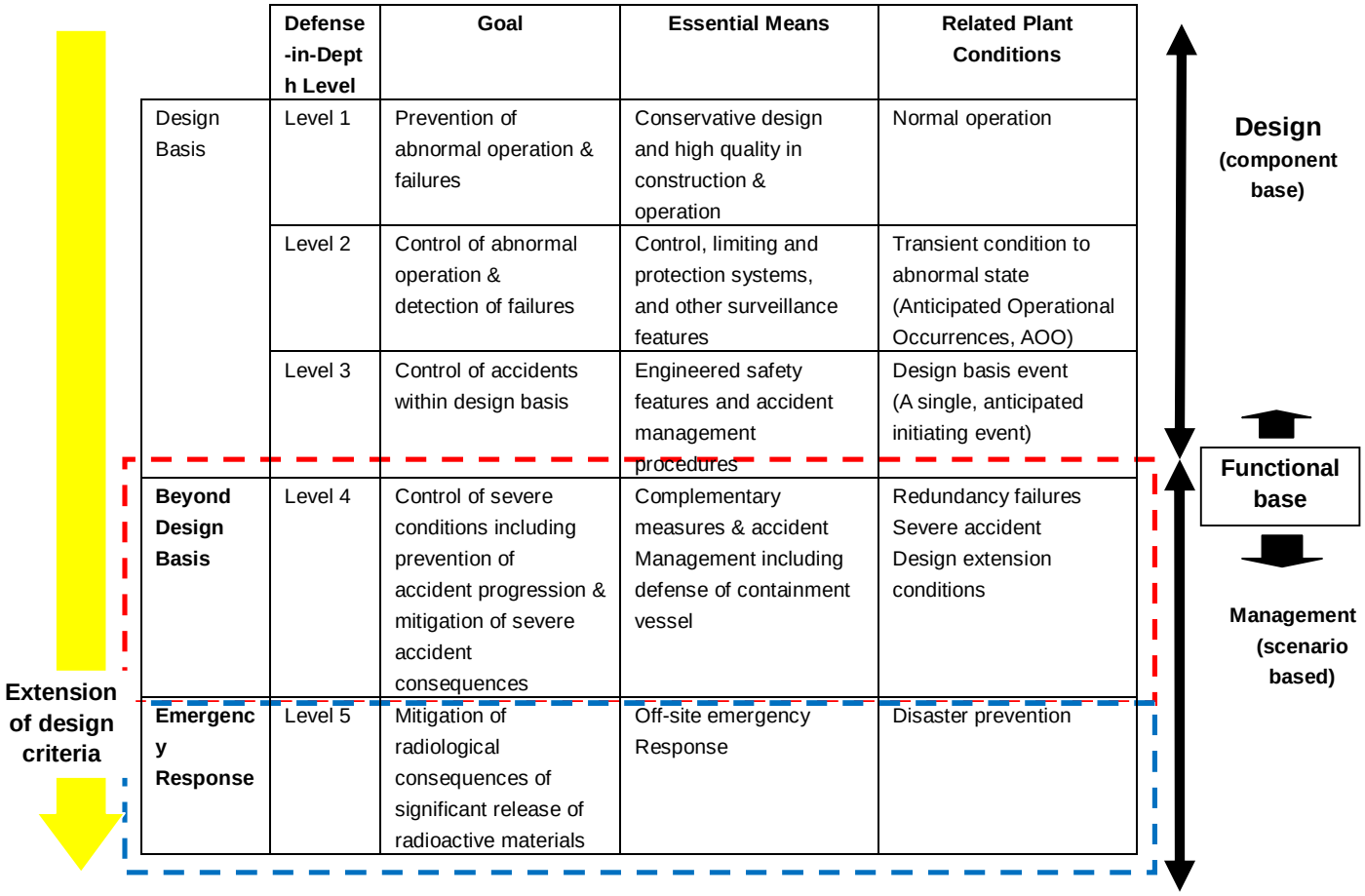
For beyond design basis conditions, the concept of "preclusion of preceding defense levels" of defense-in-depth was applied; for example, a scenario on significant radioactivity release to the containment caused by an accident was created for assessing the adequacy of the siting conditions. However, the scenario was founded on the premise of containment integrity with no mechanism-based assessment on the sequence of events.

Safety assessment must be made on the entirety of the nuclear plant system in the design process. With view to the 3.11 incident, extensive consideration should be given not only to failure and functional loss of a single component but simultaneous failures and functional losses of multiple units and common cause failures, and the interactions between the systems including those that propagate the losses and failures. Since management of beyond design basis events, or events in the realm of severe accidents (defense-in-depth level 4) vary depending on the type of events and circumstances, as many scenarios on events likely to occur should be developed, and subsequently, preparedness and response measures should be formulated for each scenario. A continuous process of developing scenarios and relevant preparedness and response measures is essential for eliminating unforeseen events. Given the scenarios that may not be simulated even by state-of-the-art expertise and technologies, SSCs and procedures for accident management should be standardized and regularly reviewed, so that a more effective and systematic accident management may be established.

Plant design is formulated on the basis of design basis conditions (accidents), to which design rules as redundancy and diversity is applied to SSCs for the effective management of various events. Design basis concept so far has included a broad spectrum of events and challenges against design basis, on the premise that beyond design basis events do not occur. In view of the Three-Mile Island and Chernobyl accidents, there had been a strong focus on internal events underlined by a strong belief in the design integrity of SSCs and the unlikelihood of the occurrence of beyond design basis events.

Whereas, in dealing with external events (natural hazards), under the geophysical conditions of frequent earthquakes, Japan has conducted numerous studies and developed measures against earthquakes from the very early stages of introduction of nuclear power generation. Lessons learned from the 1995 Great Hanshin Awaji Earthquake and state-of-the-art expertise were incorporated into the revised Seismic design guidelines in 2006. Back-checks were conducted at each plant with

necessary reinforcements made. In the revised guidelines, seismic design basis was reevaluated and measures against beyond design basis events developed; risk assessment as a means for assessing plant safety was highlighted, which resulted in encouraging the voluntary initiatives of the operators in “conducting “residual risk” assessment”, and in the establishment of PRA assessment technique.



	Defense-in-Depth Level	Goal	Essential Means	Related Plant Conditions
Design Basis	Level 1	Prevention of abnormal operation & failures	Conservative design and high quality in construction & operation	Normal operation
	Level 2	Control of abnormal operation & detection of failures	Control, limiting and protection systems, and other surveillance features	Transient condition to abnormal state (Anticipated Operational Occurrences, AOO)
	Level 3	Control of accidents within design basis	Engineered safety features and accident management procedures	Design basis event (A single, anticipated initiating event)
Beyond Design Basis	Level 4	Control of severe conditions including prevention of accident progression & mitigation of severe accident consequences	Complementary measures & accident Management including defense of containment vessel	Redundancy failures Severe accident Design extension conditions
Emergency Response	Level 5	Mitigation of radiological consequences of significant release of radioactive materials	Off-site emergency Response	Disaster prevention

Table 4-2 IAEA’S Defense-in-Depth Concept and Corresponding Design Bases

Kashiwazaki Kariya Nuclear Power Plant experienced beyond design basis seismic motion in the Chuetsu Offshore Earthquake in 2007. However, because the back-check process was already in place, sufficient margin was ensured and key SSCs maintained structural integrity. Consequently, the case led to prompting seismic back-checks in all nuclear power plants in Japan.

Developing and applying design rules as seismic design basis to SSCs is important. However, as shown by the 3.11 earthquake and tsunami, rare events with very small occurrence probability do occur and will lead to circumstances exceeding design basis. The arrangement of preparedness and response measures against beyond design basis events, or accident management must be established. Extensive scenarios on failures and malfunctioning of plant facilities under beyond design basis

circumstances should be developed, whereby, the conditions of each plant should be given consideration in flexibly determining functions necessary for maintaining “nuclear safety”, or safety of the plant system as a whole. Natural disasters are difficult to predict and control, where whatever measures taken may sometimes bring the same results as taking no actions at all. As the second best solution, preparedness measures should include drills and trainings on various accident scenarios to better flexibly deal with unanticipated events.

In the management of beyond design basis events leading to severe accidents or serious accidents (Nuclear Regulation Authority is preparing draft new standards including management of “*ju-dai jiko*” (serious accidents), renamed from the previous term, “severe accidents”), the approach on the “preclusion of preceding defense levels” of the defense-in-depth concept has so far been applied. However, because the preceding levels of protection was eliminated, measures taken has not been specific. Beyond design basis conditions should be understood as conditions under which preceding defense levels is non-existent; and thus, a systematic framework and measures to deal with each accident scenario should be established.

The design requirements of nuclear power generating system is set forth in warranting safety. Fig. 4-3 shows key safety functions of nuclear power facilities, such as control functions for “shutdown”, cooling functions for “cooling”, boundary functions for “containment”, as well as the commonly shared key functions for power supply, shielding, air-conditioning, etc. These functions are further divided into “sub-functions” or sub-systems in accordance with the governing roles, or elements. Functions consist of key elements of not only components, but procedures and management. Representative SSCs, or structures, components, systems comprising the functions of PWR and BWR facilities are shown - by extracting the configuration of SSCs with key safety functions associated with defense-in-depth shown in the diagram, functionality or the integrity of the entire plant system can be understood through the status of the SSCs (integrity or failure). In particular, key focus should be placed in establishing a framework for ensuring functionality of the minimum required functions during accident progression by taking into account not only single component failures but simultaneous failures in multiple units, interactions between functions and the likelihood of propagation of damages and failures, as well as common cause and duplicated failures.

The key fundamental functions of nuclear facilities such as the boundary function, cooling function, control function, and other functions as power supply capabilities are shown in Fig. 4-4. The design of nuclear facilities is governed by the adequate integration and maintenance of these functions, which warrants the integrity and safety of SSCs against any design basis events.

Whereas, measures for beyond design basis events vary depending on the type and circumstances of each event, and thus, extensive scenarios must be developed with adequate measures created for each scenario. By establishing codes and standards on SSCs and procedures, a more systematic accident management framework may be established.

Functions	Functions defined under regulatory policy	Structures, systems & components	
		PWR	BWR
Example of Boundary Function	1)Reactor coolant pressure boundary	Component & piping systems composing the boundary (excluding instrumentations as small bore-piping & related components)	Component & piping systems composing the boundary (excluding instrumentations as small bore-piping & related components)
	3) Over pressure protection of coolant pressure boundary	Pressurizer safety valve (w/ disclosure function)	Safety relief valve functions of SR valve
	6) Radioactive release containment, radiation shield & mitigation of radioactive release (1)	Reactor containment vessel, annulus, containment vessel isolation valve, containment spray system (CSS), annulus air clean-up system	PCV, PCV isolation valve, PCV spray cooling system, flammability control system (FCS)
	6) Radioactive release containment, radiation shield & mitigation of radioactive release (2)	Atmosphere clean-up system, flammability control system (FCS)	Reactor building; SGTS; filtration, recirculation & ventilation system (related systems); exhaust tube (SGTS exhaust tube support functions)
Example of Cooling Function	3) Maintenance of core Geometry	Reactor core support structure, fuel assembly (excluding fuel)	Reactor core support structure, fuel assembly (excluding fuel)
	4) Heat removal after reactor shutdown	Residual heat removal systems: residual heat removal system, auxiliary feed water system and the following systems related to SG secondary isolation valve: main steam system, main steam safety valve, main steam relief valve (MSIV; manually controlled)	Residual heat removal systems: RHR, RCIC, HPCS, SR valve (safety relief valve functions), automatic depressurization system (ADS; manually controlled)
	5) Core cooling	Emergency Core Cooling System (ECCS): low pressure coolant injection system (LPCI), high pressure coolant injection system (HPCI), accumulator system	ECCS: RHR, HPCS, LPCS, ADS (Automatic Depressurization System)
Example of Control Function	2) Prevention against excess reactivity	Control rod drive system Housing	Control rod (CR) coupling
	1)Reactor emergency shutdown	Control rod system in reactor shutdown system	Scram function
	2) Sub-criticality maintenance (1)	Reactor shutdown system	Control rod & control rod drive system
	2) Sub-criticality maintenance (2)	Reactor shutdown system	Standby Liquid Control system (SLC)
Others	7) Activation signals of engineered safety facilities & reactor shutdown system	Safety protective system	Safety protective system

	8) Related critical safety Functions	* Standby onsite power supply system * Reactor control room & radiation shielding of control room * Air-conditioning & ventilation system * Component cooling water system * DC power supply system * Control air system (CAS) (all of the above are MS-1)	* Standby onsite power supply system (related systems), DG fuel transport system, DG cooling system * Reactor control room, radiation shielding of control room, standby air conditioning & ventilation system * Emergency cooling water system * DC power supply (all of the above are MS-1)

Table 4-3 Key Fundamental Function & Constituting Systems and Components

Figure 4-4 presents the relationship between defense-in-depth and key safety functions. Each function comprises of sub-functions classified by the governing roles, or elements, and incorporated into each level of defense-in-depth. Examples of functions that work as back-ups in the event of failure of another function are shown in the diagram. The preconditions, or the fundamental driving source of all functions and of most SSCs is power supply security. Obviously, power supply must be backed up by alternative sources for which interactions and the correlation between associated functions should be clarified.

Events exceeding the scope of level 3 defense-in-depth fall in the region of severe accidents, or defense-in-depth level 4, to which provisions against various unanticipated circumstances should be arranged. Not only consideration of measures based on extensive scenarios focusing on the hardware aspects of key design basis SSCs, but measures emphasizing human factors and intangible aspects, including the utilization of all available SSCs including standard components is the key to accident management in this region. For example, it is essential that the valves may be opened and closed by manual operation in the event of an SBO.

Defense-in-depth level	Boundary	Cooling	Control	Common
Level 1 Normal conditions	Preventing proliferation of fission products in the coolant (PS-3) Radioactive materials storage (PS-3) Coolant pressure boundary (PS-1) Containment of reactor coolant (PS-2) Maintenance of Reactor coolant (PS-3) Radioactive materials storage (PS-2) Closure of safety Relief valve (PS-2)	Maintain core geometry (PS-1) Spent fuel pool water injection	Prevention of excess reactivity (PS-1) Circulation of reactor coolant (PS-3)	
Level 2 Prevention	Over pressure protection of coolant pressure boundary (MS-2)	Heat removal after reactor shutdown (MS-1) Safety shutdown functions outside control room (MS-2)	Emergency reactor shutdown (MS-1) Maintain sub-criticality (MS-1)	
Level 3 Mitigation	Containment of radioactive release [PCV] (MS-1)	Core cooling (MS-1) Mitigation of reactor pressure increase (MS-3)	Maintain sub-criticality (MS-1) Controlling power output increase (MS-3)	
Level 4 Accident Management	Radioactive release containment [reactor building, gas treatment] (MS-1) Severe accident management [PCV event] (MS-3)	Severe accident management [Make Up Water System, FP systems] (PS-3)	Severe accident Management (MS-3)	

Table 4-4 Relationship Between Defense-in-Depth and Key Safety Functions

4.4 Ensuring Safety of Operating Plants - Back-Fitting

In ensuring nuclear safety of operating plants, a continuous process of reevaluating methods, framework and processes for ensuring safety by incorporating state-of-the-art technologies, or implementing quality assurance PDCA, in other words, the back-fitting process is important. The following are the basis to the back-fitting procedures which should be organized.

1) Introduction of System Safety

Fundamental concept of system safety should be based not on the integrity of a single component or a single function, but should encompass the entire system that include equipments, pipes, other structures, and passive and active systems, the assessment on which will ensure comprehensive safety of the plant as a single system. Through this method, not only the integrity of single power generating unit but safety of the entire plant system, including those with multiple units may be ensured.

For ensuring safety of nuclear power plants, the concept of "system safety" should be introduced, an integrated framework founded on the correlation between all constituting elements of systems within a plant. Changes in the functional status of each element and each system are translated into the context of the framework, which enables to assess and understand the safety status of the entire plant.

Points of consideration in assessing "system safety" of operating plants are as follows: first, the plant must be configured of tangible materials and components, where performances, such as strength, etc., of the materials and components are visible; secondly, criteria on structural integrity and the concept of ensuring safety is time dependent, or depends on the time point and circumstances that the assessment is carried out.

For enhancing technical assessment[16] on ageing plants, a method should be developed for predicting reduction of safety margins with ageing degradation, along with efforts in improving prediction accuracy by applying state-of-the-art technologies. A comprehensive safety assessment system that verifies the integrity of ageing plants under normal operation, transient and accident conditions should be established, with consideration given to safety margin for each plant type and system design; and plant specific maintenance records, etc. related to inspection, repair and replacement of components, structures, instrumentations comprising the plant system.

2) Fundamental Concept

In assessing the correlation between operating period and plant system reliability (functional integrity vs functional failure risk), the assessment should focus not only on physical aspects of degradation (e.g., degradation of structural materials), but on "functionality" – where the roles of SSCs are represented by relevant functions. The changes in functional performance with time (or degradation) are evaluated through functional failure probability and functional failure risk.

Design safety standards change with the accumulation of new expertise, changes in safety concept, as well as through the clarification of phenomena mechanism. Design modifications on the basis of changes in assessment standards and safety concept should be adequately applied to operating plants. The purpose of system safety assessment is to appropriately and accurately assess the latest plant

status.

For long-term operating plants, assessment on functional failure caused by degradation should be carried out over the entire plant system. Criteria for the assessment as safety limit and safety framework should be based on current standards, not design stage criteria which should clarify issues to be addressed and necessary measures that should be implemented. Modifications based on the changes in safety concept and application of new expertise should be made throughout the 40-years design lifetime. For plants in extended operation, assessment based on current standards should also be made to ensure safety [17].

Fig. 4-5 shows the life stage of nuclear power plants from design, construction to operation and time dependent changes such as changes in safety standards and degradation.

"System safety" assessment enables quantitative assessment on the safety level at any point in the lifetime of nuclear power plants, of newly constructed and ageing plants alike, the concept of which is shown as follows.

Degradation assessment combines evaluation of both physical and functional aspects of degradation. The assessment evaluates functional degradation of the systems and not individual degradation factors. By evaluating functionality, functionality and changes in the conditions of functionality (degradation) may be quantified. In degradation management, parts replacement, modification of piping materials, revisions in welding method have been dealt with separately. Although some of the modifications required specification changes, they did not involve changes in design standards, and thus no back-fitting was required. Most SSCs have been replaced by state-of-the-art materials and current design standard SSCs with time, and very far from degraded conditions.

One of the key points in system safety assessment are quantitative assessment on the margin of "nuclear safety" by evaluating functional integrity of the systems at any time in point of operation. Secondly, in the assessment, safety standards applied in the design, manufacturing and construction phases are replaced by standards at the time point of assessment, which is called the back-fit rule. Thirdly, with view to TEPCO's Fukushima Dai-ichi Plant accident, consideration is given to load impact caused by external events and natural disasters to assess plant integrity. In reference to Fig. 4-5, safety limit is generally not shown quantitatively in safety standards and design standards. However, safety limit should be quantitatively presented in defining safety goals of "nuclear safety", just as with the values of risk and accident occurrence probability. For example, acute fatality risk resulting from radiation exposure in the vicinity of the site boundary under accident conditions is determined as should not exceed probability of 10^{-6}/people-year. The acceptable level of risk involving radiation exposure is also given as safety goal. In addition, performance goals are presented as indices in judging conformity to the safety goal, such as CDF (core damage frequency) of not exceeding 10^{-5}/reactor-year. Although defining the maximum safety limit is difficult, public

consensus on the acceptable level of risk should be established. (The issue will be discussed in a separate section of the report)

In determining design requirements for SSCs that meet indices of safety goal and performance goal, formulating standards with individual requirements for each SSC item is difficult. Thus, safety assessment on SSCs designed on the basis of relevant performance standards and design safety standards developed for each function, should be carried out to confirm whether safety performance goal is fulfilled, or not. However in reality, it is difficult to specify the correlation between safety standards encompassing the entire framework of nuclear facilities and design requirements on individual SSCs. As such, safety requirements defined under a consistent set of rules and standards, and applied to individual item was generally considered to ensure coherency in its entirety. However, ageing plant assessment carried out so far has not emphasized coherency, nor has been quantitatively assessed, and standards for each items has not been re-examined for modification even though safety standards have changed over time. Over 30- to 40-years time period, various changes on the approach to safety standards have come forth, shifting between tighter and loser criteria, which should be dealt with flexibly according to circumstances. In particular, various interactions, correlation between components, configuration of the systems, propagation of the influences of functions, etc., should be clarified for an integrated safety assessment, which is the primary perspective of system safety assessment.

Footnotes

1. Few documents exist that define or describe concept on ‘multiple defense’ or ‘defense-in-depth’ in Japan. The Nuclear Safety Commission issued “Accident Management for Severe Accidents at Light Water Power Reactor Installations” on May 28, 1992 (partially revised October 20, 1997) stating, “ safety of the reactor facilities in Japan is sufficiently ensured by the current safety regulations by implementing strict safety measures in the design, construction and operation based on multiple defense concept to: 1) prevent the occurrence of abnormal events; 2) prevent abnormal events from spreading and developing into accidents; 3) prevent extensive release of radioactive material.” The term is also defined in the Glossary of “Common Important Items in Safety Regulations of Radioactive Waste Disposal” authorized by the Nuclear Safety Commission June 10, 2004.

In the “Exchange of Opinions with External Experts on the Promotion of Fundamental Policy of Current Measures - Fundamental Policy on Ensuring Safety” conducted by the Nuclear Safety Commission between February 2011 and March 2012, the Commission examined and initiated organization of the defense-in-depth concept by referring to the IAEA Safety Standards, etc. (Ikokigen No.8-2, March 7, 2012).

2. The objectives in establishing graded levels of protection is enhancing reliability

(effectiveness) of the defense framework by repeating layers of preventive and mitigation measures against escalation of abnormalities. By applying design rules such as redundancy and diversity to measures specific to one level of defense will enhance reliability of these measures, but will not strengthen measures associated with other levels of defense. The application of redundancy and diversity and the application of graded protection are separate features - redundancy and diversity may not replace graded defense, and vice versa.

3. Measures corresponding to all levels of defense are available under normal operating conditions. Under transient and accident conditions, measures associated with the level of defense relevant to specific plant conditions will be available. However, this does not mean that as long as a number of defense-in-depth levels are available, deficiency in any of the defense levels is acceptable.

4. Design Extension Conditions (DEC) refers to accident conditions exceeding design basis and conditions involving radioactive material release but controlled within the acceptable limit. The best estimation technique is used in the design process in consideration of beyond design basis accidents. DEC includes severe accident conditions.

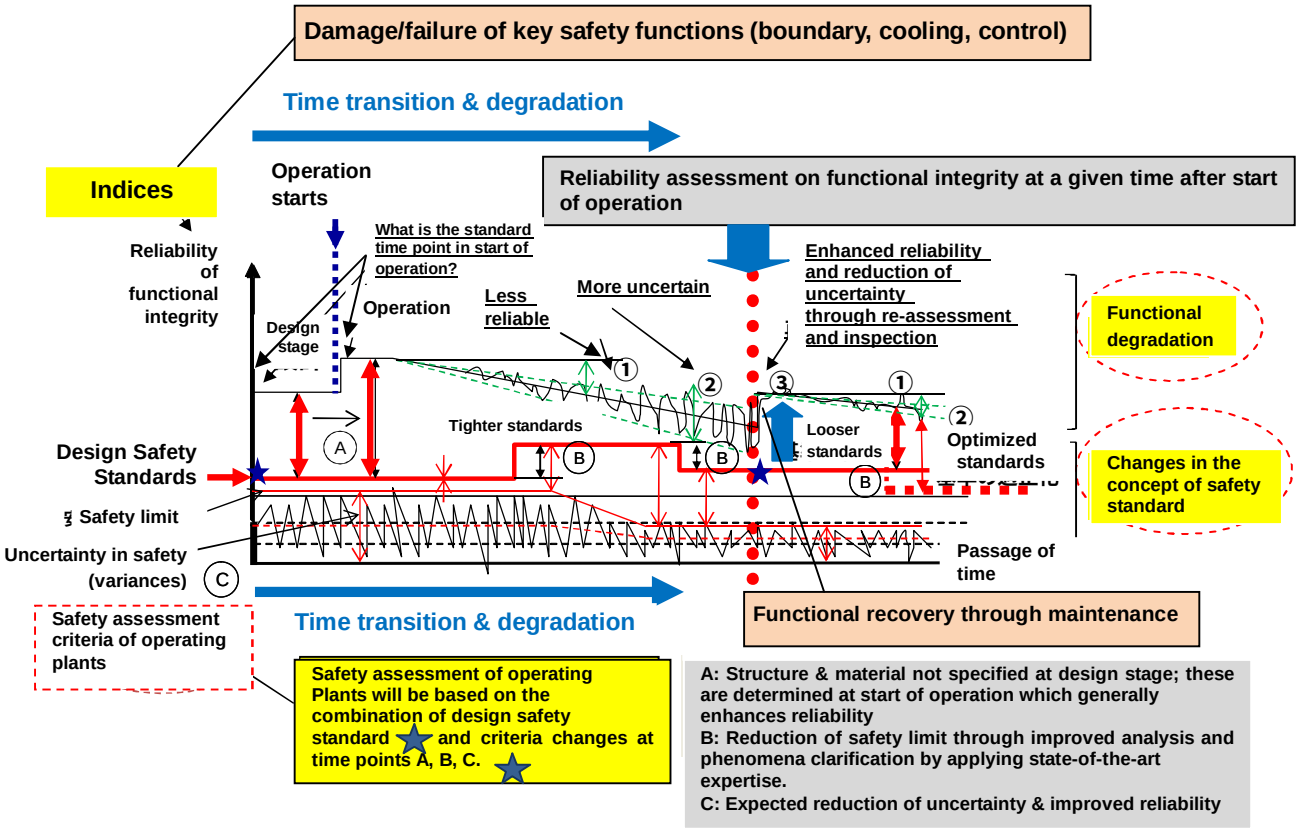


Fig. 4-5 System Safety Assessment Procedure

5. ENSURING NUCLEAR SAFETY

A number of direct causes underlying TEPCO's Fukushima Dai-ichi Plant accident have been identified, and the measures for these causes are being examined. The common fundamental essence underlying these direct causes, or factors that "induced development to a severe accident" should be extracted and analyzed for developing and implementing effective severe accident management.

On the basis of understanding on the 3.11 Fukushima incident and the framework and history in ensuring "nuclear safety" in Japan, analysis on the fundamental causes of the accident have been presented, together with recommendations on severe accident management in this section.

5.1 Transition from New Technology Introduction to Fundamental Safety

The first light-water nuclear reactor technology was introduced into Japan from the US in 1960's. Together, codes and standards on structural integrity, pressure vessels and piping systems established by the ASME (American Society of Mechanical Engineers) was also introduced and utilized. Emphasis was placed on the accurate development of theory in the areas as physics and engineering, rather than the establishment of philosophy and approach on nuclear power generation and nuclear safety in Japan. There was a strong focus on structural strength for a long period of time in the Japanese regulatory framework because of the circumstances that nuclear power generation was introduced, and the understandability of nuclear physics and engineering theories. Japan has experienced numerous failures and accidents such as steam generator condenser tube rupture, stress corrosion cracking, fuel rod damage, etc., at nuclear power plants since the introduction. Numerous assessments, researches etc., have been conducted in resolving these issues, which greatly enhanced safety of nuclear facilities and were reflected in the safety codes and standards. The endeavors in mitigating radiation exposure of the workers and in achieving the highest standards of safety have led to the development of next-generation reactors and advanced standard reactors and enhanced researches on component systems. However, in recent years, efforts in addressing nuclear safety has diminished with the maturitization of nuclear power generation and the reduction on the allocation of personnel and financial resources by national and private institutions.

Japan has made tremendous investments in the acquisition of, and contributed to the enhancement of enormous volumes of codes and standards. However, this consequently led to reduced concerns over the safety of actual plant facilities and systems, which was further promoted by regulatory control over "nuclear quality assurance". Although the introduction of quality management was instrumental in ensuring product quality and the integrity of each component, it extended over the areas of work procedures which resulted in devoting time and labor on creating vast amount of documents.

On the other hand, the experiences of the Three-Mile Island and Chernobyl incidents have led to

the global trend in the development of severe accident management measures. As shown in **Fig.2-1**, US has conducted studies on PSA (probabilistic safety assessment) from the early stage, applying the results in developing various safety measures in the wake of Three-Mile Island and Chernobyl incidents. In Japan, although many research studies including experiments and analyses related to severe accident management were made, regulated severe accident management and PSA implementation, together with the development and implementation of safety measures was delayed, lagging behind in the safety approaches taken by the US and European countries. Japan's approach on severe accident management was in line with the evolving global standards around 1990; however, after decision was made in encouraging voluntary initiatives of the licensees on accident management, there was no continuous follow-up on accident management of each plant by the regulatory body.

The primary focus should be in identifying key functions required for ensuring "nuclear safety" on the basis of all possible conditions that can be assumed for nuclear facilities. Nuclear safety not only involves nuclear facilities but extends to the Japanese society as a whole in the absence of a safety culture.

The tendency of the Japanese regulatory body, local governments and the mass media was making large social issues of minor issues (for example, "problem reporting was delayed by an hour"), which normally concluded in political settlements rather than in identifying and clarifying relevant technical issues. The element underlying these circumstances was the close relationship, or the "safety agreement" between the local governments and the operators. The local government re-evaluated safety by organizing an independent group of experts after safety assessment was carried out by the central government (a triple regulatory structure) to ensure safety, which contributed to the loss of focus in ensuring the safety of nuclear facilities.

Generally speaking, "unplanned shutdown" is a key safety criteria emphasized by the nuclear industry. With the very few unplanned shutdowns in Japan as compared with other countries, this has become the basis to fostering safety myth, the "absolute safety" of nuclear facilities in Japan. Accordingly, a general notion was formulated in the Japanese nuclear community that "making risk assessment on what is safe is unnecessary", which hindered development of risk assessment that included external events, and risk assessment for evaluating safety of nuclear power reactors did not become regulated. Formulation of the safety myth highlighted minor issues involving structural integrity, which led to deviating away from the fundamentals of ensuring "nuclear safety".

Because the boundary between voluntary initiatives of the operators and regulatory requirements had not been defined clearly throughout the historical process of ensuring nuclear safety in Japan, and because prudent opinion leaders, demanding mass media, and law suits had to be dealt with, both the regulatory body and the operators became seeped in "safety myth", which hampered efforts in ensuring "nuclear safety" in Japan. Ensuring "nuclear safety" should be a common goal

shared by all parties including the regulatory body, operators, local governments, the residents and prudent opinion leaders. Critical but worthy opinions and recommendations should be accepted with an open mind, and referred to for the achievement of "nuclear safety" through collaboration by all parties involved. The regulatory body should liaise with the operators in promoting and ensuring safety of nuclear power facilities. At the same time, the regulatory body should independently ensure full surveillance of nuclear facilities, separately from the pursuit of the commonly shared goal of ensuring nuclear safety.

5.2 Breaking Away From "Safety Myth" and the Establishment of Risk Communication

The operators have emphasized "safety myth of nuclear power plants" such as "nuclear power plants are absolutely safe" and "severe accidents do not occur" in communicating to the public. Many have pointed out that the belief in the absolute safety of nuclear facilities hampered the development of severe accident management.

For example, the investigation report on the JCO accident prepared by the former Nuclear Safety Commission cites that the root cause for the accident was the overconfidence in safety, the belief in the absolute safety of nuclear power held by the Japanese nuclear community, pointing out the need for efforts in ensuring safety with due consideration given to risk. In view of the recommendations made in the investigation report, "2000 White Paper on Nuclear Safety", issued a decade after the JCO accident made the following points.

"Although "absolute safety of nuclear power" is not a belief necessarily held by many in the nuclear community, what led to the creation of the "safety myth"? The following factors may have given rise to the development.

- Over-confidence on the reliability of the design of key facilities such as the pressure vessel and the containment vessel, with tighter safety requirements as compared to other industries.
- Over-confidence on the long-term track record of maintaining safety, not having experienced events with adverse consequences to human life.
- Weathering of past accident experiences.
- Public acceptance (PA) activities in promoting construction of nuclear facilities.
- Aspiration (Strong desire) for absolute safety.

Under these circumstances, the significance of the simple yet fundamental fact that nuclear safety is ensured through day-to-day activities became obscure and replaced by clichés as "nuclear power is safe", which proliferated through PR activities on public acceptance of nuclear plants.

However, these circumstances underestimate "safety culture" in maintaining and enhancing nuclear safety by all related parties. Many of the accidents and failures in the past have been caused

by human factors. All parties in the nuclear community should come face to face with risk involving nuclear power and maintain efforts in clarifying and reducing risk to as low as reasonably achievable."

As pointed out in the White Paper, the JCO accident has revealed vulnerabilities in considering human factors (organizational factors). Similarly, the 3.11 incident revealed deficiencies in the consideration of natural disasters.

Various reports on TEPCO's Fukushima Dai-ichi Plant accident have pointed out more clearly the adverse effects of the safety myth. The report by the private sector[3)] describes the formulation of safety myth as based on social circumstances - the relationship between the nuclear industries, government organizations, local governments, and politicians, and the circumstances where no one doubted safety myth which hampered the development of severe accident management. The report by the government gives no reference to the term "safety myth", however, quotes a comment by the director of Nuclear and Industrial Safety Agency affirming "it was extremely difficult to contradict past circumstances that had denied the occurrence of a serious accident in explaining to the local residents". This explicitly shows that safety myth had hindered the development of severe accident management.

Further, the report by TEPCO[18)] ("Overview of Fukushima Dai-ichi Nuclear Power Plant Accident and Nuclear Safety Reform Plan") points out as the root causes for inadequacies in the preparedness for the accident were the lack of "safety consciousness", "technical competencies", and "communication capabilities". It makes further analyses on the underlying elements as a "negative spiral" created by conjugation of factors such as management stance on "focusing on plant factor as the key management issue" and "the assumption that safety has been solidly established", which had deeply took root in the organization. Particularly, the conviction (OR) apprehension that admitting risk would require implementation of additional measures without which plant operation would be suspended, led to the unwillingness in introducing risk communications, and fostered wishful thinking that safety had been solidly established. The term "safety myth" again is not used in the report, however, this clearly indicates that the establishment of safety measures was hampered because the operating organizations could not contradict what before they had explained was safe.

The fundamental principles in ensuring nuclear safety is the continuous process of verifying the approach to safety management on the basis of new findings and state-of-the-art expertise associated with events including natural disasters and human events, operating experiences of the facilities, results of safety researches, which came to be established as a system called periodic safety reviews. However, severe accident management was not included in the implementation of periodic safety reviews, a significant defect that was grounded on the existence of "safety myth".

Once “safety myth” takes root, when new findings that contradict safety myth or cases that should be reflected in safety assurance activities come forth, actions are taken to circumvent the issue, or in some cases, no actions are taken at all. The greater the significance of the finding or the case, the more in evading the issue in fear of impairing social acceptance of nuclear power – the adverse effects of safety myth that has hampered the establishment of nuclear safety.

The break away from safety myth can be realized through the admission of risk and demonstrating efforts in managing risk - a sincere approach in risk communications to the public to gain understanding on nuclear safety.

It is essential to communicate to the public not only what the existing risks are, but that the level of existing risk is acceptable on the basis of scientific grounds. The process should include verification of the current safety status, which is linked to continuous safety enhancement efforts in recognizing and improving deficiencies. Points of consideration are shown below.

1) Benefit and Risk

There is no absolute safety associated with any kind of a system (railways, aircrafts, cars, etc.). Benefits generated through the use of a system will inevitably involve physical, mental, or financial risks.

TEPCO’s Fukushima Dai-ichi Plant accident reaffirmed consequences of risk associated with nuclear power plant operation. However, other means of power generation also involve various risks. For example, the use of petroleum with uneven distribution and limited reserve for thermal power generation has given rise to various border disputes and involves significant energy security risk. Coal fire power involves emission of large volume of greenhouse gases and release of particle elements that may cause health hazards which must be controlled. As such, the best appropriate energy sources should be selected extensively from a broad perspective. To this end, assessment on risk associated with nuclear power generation against the cost in maintaining safety should be carried out and compared with those of other energy sources.

Further, safety objectives specifying target safety level of the ongoing safety enforcement measures should be established and shown to the public.

2) Conditions of Acceptable Risk (Safety Objectives)

Although there are many advantages to nuclear power generation as compared to other energy sources, it also involves generation of radionuclides, or fission products, released by reaction within the atomic nuclei, as uranium or plutonium. Fission products that continue to decay and produce heat need to be cooled down even after reactor shutdown, along with retainment of radioactive material in the containment. TEPCO’s Fukushima Dai-ichi accident afflicting devastating damage to the

residents in the vicinity of the plant and to the people in Japan was caused by the failures in cooling and containment of radioactive material. In minimizing such risk, what level would be considered as acceptable in terms of safety?

"How safe is safe enough" has been debated globally among experts, and to this end, many countries have determined safety objective presented as probabilistic figures, supplementing deterministic rules.

In Japan, the former Nuclear Safety Commission proposed "Safety Goal (draft)"[19] after examining the issue. The goal presented acceptable level of risk arising from activities in the use of nuclear power quantitatively, as regulatory requirements on the extent that the licensees must control low probability risks This has enhanced transparency, predictability, rationality, coherency of the regulatory activities. Further, the establishment of "Safety Goal" presenting public risk has provided the basis for efficient and effective dialogue between the government and the public on nuclear regulatory control, for example in the development of guidelines and standards.

Three levels of goals -qualitative, quantitative and performance goals have been given under the safety goal. Qualitative goal is shown as the ultimate goal, affirming that "the likelihood of adverse health consequences to the public induced by releases of radiation or radioactive materials arising from activities in the use of nuclear energy should not outweigh public health hazards incurred through everyday life."

Quantitative goal presents specific numerical values embodied in the qualitative goal – "the mean value of acute fatality risk by radiation exposure resulting from a nuclear facility accident of individuals of the public in the vicinity of the site boundary of the nuclear installation shall not exceed the probability of approximately $1x10^{-6}$ per year; and the mean value of fatality risk by cancer caused by radiation exposure resulting from a nuclear facility accident of individuals of the public residing in the area, but with some distance from the facility, should not exceed the probability of approximately $1x10^{-6}$ per year." Individuals subject to these goals are limited to residents in the vicinity of the site boundary of nuclear installation, and risk of radiation exposure is 1/100 of annual fatality rate by car accidents as shown in **Fig. 5-1**, targeting significantly lower level as compared with risk incurred through everyday life.

As shown by TEPCO's Fukushima Dai-ichi accident, safety objectives should give consideration not only to individual fatality risk but environmental risk with equal significance which will be described later. Sources of everyday risk shown in the diagram are actual statistics, whereas, values for comparison against the safety objectives are assumptive calculations containing uncertainties, and thus, care should be taken in comparing the two values.

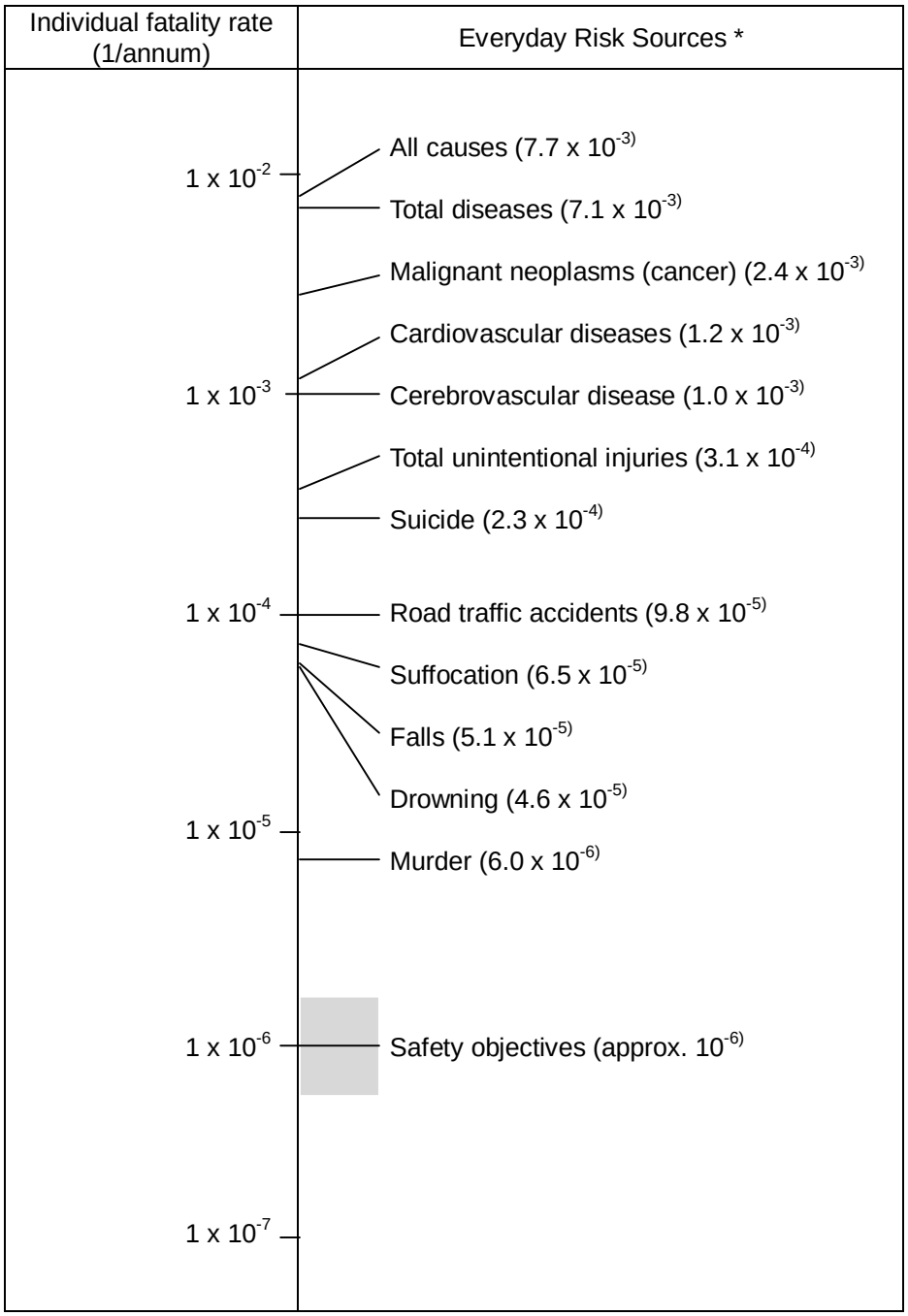


Fig. 5-1 Conceptual Framework on Safety Objectives for Nuclear Incidents
(* reference: "2001 Demographic Statistics", Ministry of Health, Labor and Welfare)

As performance goal, parameters describing characteristics of the facilities are presented as indices in judging conformity to the safety goal. The performance indicators of which are CDF (core damage frequency) of 10^{-4}/reactor-year and CFF (containment failure frequency) of 10^{-5}/reactor-year are determined for accident scenarios on all internal and external initiating events (except for malicious, or deliberate human events). However, the given figures are not fixed values. The

requirement in the application is “in all activities involving nuclear installation, including design, construction and operation, risk reduction measures must be planned and implemented so that radiation risk to workers and the public does not exceed one 1 millionth annually, or kept as low as reasonably achievable.” If necessary measures are planned and implemented on the basis of the requirements described above, then it would not mean that the safety goal is not fulfilled even if the result of risk assessment exceeds the value of one 1 millionth. (Refer “Interim Report on the Investigation and Review on Safety Goals", Special Committee on Safety Goals, Nuclear Safety Commission, December, 2003)

3) Consideration of Environment Contamination

Human fatality risk is used as indices for the above safety goal. TEPCO’s Fukushima Dai-ichi accident has shown that adverse consequences on human life and health caused by radiation exposure can be significantly mitigated through protective measures such as evacuation, etc. Radiation effects or injuries to the local public has not been observed in Fukushima. On the other hand, extensive environment pollution has devastated the living bases and the infrastructure of the local public with a long-term negative impact. The cost for the clean-up will be a significant burden on the people of Japan. Therefore, indices on large-scale environment pollution and its acceptable frequency level should be included in the safety objectives. Some countries have already regulated these items as safety objectives, which Japan had given consideration in the past as follows.

A committee comprised of experts in the nuclear industry and research institutes was established (as part of voluntary research activities) by the Nuclear Safety Research Association with the objective of clarifying items that should be considered in severe accident management, proposed a performance goal on the containment vessel as the basis for containment vessel design of next generation light-water reactors. (refer “Guideline on Severe Accident Considerations in the Design of Next-Generation Light Water Reactor Containments, July 1999 http://www.nsra.or.jp/safe/cv/index.html)[20)]. Because the goal defines the conditional probability on the premise that an accident will occur as well as the occurrence frequency of significant radioactive material release, it may be utilized as a safety goal for the entire plant system, although determined as performance goal for containment vessels.

The goal comprises of three levels of qualitative, quantitative and supplementary goals.

(1) Qualitative goal
   1) Maintain a sufficiently small occurrence probability of circumstances requiring short-term countermeasures such as evacuation, etc.
   2) Maintain to an insignificant level the occurrence probability of deterministic effects of radiation exposure and long-term evacuation.

(2) Quantitative goal (numerical values specifying qualitative goal)

1) FP (fission product) Containment Retention Factor (CRF-1) $<10^{-6}$/reactor-year
2) FP (fission product) Containment Retention Factor (CRF-2) $<10^{-7}$/reactor-year

(3) Supplementary goal

1) CCFP (conditional containment failure probability) of not exceeding 0.1 (strict adherence to the goal is not necessary if CDF $<<10^{-6}$/reactor-year); and CDF of not exceeding 10^{-5}/reactor-year.
2) Occurrence frequency of early containment failure not exceeding 10^{-7}/reactor-year.
3) Occurrence frequency of core damage due to containment bypass not exceeding 10^{-7}/reactor-year.

FP containment capacity is called CRF or Containment Retention Factor, obtained by containment release/environment release, and defines the reducibility of fission products released into the environment presented as coefficients. In order to determine CRF, benchmark dose rate for each environment influencing factors (items) in the qualitative goal need to be set forth. By referring to "Guidelines on Nuclear Emergency Preparedness in the Adjacent Areas of Nuclear Power Stations" (Nuclear Safety Commission June 30, 1980), ICRP Publication 41, ICRP Publication 63, IAEA Safety Series No.115-1, the effective dose rate and childhood thyroid disease dose rate that require short-term countermeasures are determined respectively at 50mSv and 500mSv; for averting deterministic effects incurred by total body radiation exposure at 0.25Sv; and the effective dose rate for considering long-term evacuation at 1Sv, by assuming external exposure to radiation reflected by the deposit surface and internal exposure through inhalation of radiation scattered and emitted by particles, under the assessment period of lifetime (70 years). CRF is the coefficient determined for each plant on the basis of the preceding benchmarks and the amount of fission products in the reactor, classified in accordance to the fission product type as noble gases, gaseous iodine, particle materials.

For ease of understanding of (2) Quantitative goal - for example with Cesium (Cs^{137}) (half-life of 30 years), a typical chemical element that contribute to land pollution, the ratio of Cs^{137} released should be maintained at a level between 1/800 (release from exhaust vent) and 1/ 4500 (atmospheric release) of the total Cs^{137} volume contained in 110Mgw light-water reactors common in Japan under normal operating conditions (approximately, 2×10^{17}Bq). The amount of Cs^{137} released in TEPCO's Fukushima Dai-ichi Plant accident is currently assumed at between 6×10^{15}~15×10^{15}Bq, and the target goal is 1/100 on the order of TEPCO's incident. To meet the release requirements, containment integrity should be maintained, as well as the arrangement of high-performance filtered vents to enable venting in the event of a failure.

One characteristics of the guideline by the Nuclear Safety Research Association is that not only the safety goals are defined, but events under severe accident conditions that should be considered

in the safety assessment and method for assessing these events are given. The guideline can be used as reference for the preparation of codes and standards by the Nuclear Regulation Authority. However, attention should be paid to the fact that only internal events are dealt with in the guideline; external events are left for future development.

Safety goals that take into account environment pollution established by countries as UK and Finland defines the source terms on extensive radioactive material release (radioactivity release from the containment vessel) and sets release frequency goals as a benchmark for controlling risk on extensive environment pollution. Finland defines extensive radioactive material release as 10^{14} Bq for Cs^{137}, and sets release frequency goal of not exceeding 5×10^{-7}/reactor-year. The release frequency goal is on the same order as that presented by the Nuclear Safety Research Association.

4) Issues to be Addressed on the Safety Goals

Examples of safety goal prepared by the Nuclear Safety Commission and that developed by Nuclear Safety Research Association has been described so far. The following are issues that should be examined further.

The guidelines by Nuclear Safety Research Association deals with risk associated with internal events but not external events. In addition, the environment pollution frequency goal of not exceeding 10^{-7}/reactor-year is a value too small to verify the achievement of safety. Accordingly, appropriate values with consideration given to external events should be determined for the frequency (safety) goal. Further, items as method for assessing the extent of the achievement of the goal; how uncertainties should be dealt with; and how events not included in the PRA should be handled, etc., should be examined.

The greatest advantage in presenting quantitative goals on risk (or safety goal) is that it enables a reasonable and practicable way of ensuring safety, and that the efforts in ensuring safety is shown with scientific ground to the public. Presenting extremely small values that cannot be verified through scientific assessment is questionable. A reasonable and applicable safety goal in view of the current and future standards of science and technology should be proposed for building consensus with the public. Although it is very difficult to verify extremely small values, the accuracy on the extent of the achievement of the goal can be enhanced by combining the values with the supplementary goals corresponding to each phase of PRA assessment. Specifically in PRA:

Frequency of extensive release of radioactive material

= occurrence frequency of initiating events

x failure probability of mitigating functions against core damage

x conditional probability of extensive containment damage under core damage conditions.

Accordingly, focus should be given not only to the values of extensive release of radioactive material but also to the conditional probability of extensive containment rupture under core damage conditions in verifying the results of PRA, to confirm whether the influences of various phenomena likely to occur after core damage and those of severe external events are given consideration. The "Goal on CCFP (conditional containment failure probability) not exceeding 0.1" of the preceding (3) "Supplementary Goal" is one such example. The US NRC uses the same criteria on the conditional probability of large early release of radioactive material during a core damage accident for licensing review of new nuclear power plants.

Further, it would also be of significance in setting release frequency goal corresponding to the quantity of radioactive material released that do not involve large scale environment pollution. Release frequency goal for controlling core damage frequency may be used alternatively for light-water reactors. However, regarding recycling facilities such as fuel fabrication and re-processing facilities, release frequency goal corresponding to the expected release volume is appropriate because of the limited number of scenarios, if any, on large-scale damages similar to those of light-water reactors in judging necessity of severe accident management and adequacy of measures.

Although a number of issues need to be addressed in defining the safety goals, results of examination outlined in the draft Safety Goal established by the former Nuclear Safety Commission and "Performance Goal for Containment Vessel" by the Nuclear Safety Research Association should be utilized to make improvements, to establish dialogue with the public on the acceptable level of risk, and to subsequently develop and utilize safety goal agreed on by the public.

5) Provision of Sufficient Information

In order for the safety goal and the performance goal to be effective, significance of the results of PRA, including the limitations and uncertainties contained in assessing risk should be adequately defined. For example, the preceding draft Safety Goal by the Nuclear Safety Commission requires consideration of both internal and external events as initiating events. Unfortunately, PRA with consideration given to external events such as earthquakes and tsunami had not been conducted on TEPCO's Fukushima Dai-ichi Plant. The accident brought to light the meaninglessness of comparing PRA results that had not taken consideration of risk associated with external events against the safety goal. In explaining risk to the public, due care should be taken to clarify the scope of risk, demonstrate policy in assessing risks that were not included in the scope, and how residual risks had been dealt with. As necessary, information on reasonable and practicable efforts carried out should be provided if it is difficult to judge whether the safety goal is satisfied, or not, because of large uncertainties contained in the assessment method. It is essential that these information are provided to the public, without which discussions on the acceptance of risk will not be justified.

The efforts in communicating acceptance of risk go hand in hand with gaining public understanding on the meaning and significance of specific safety measures taken by the regulatory body and the operators.

5.3 Roles and Responsibilities of Scientists and Engineers

All those involved in nuclear power, including the operators, the regulatory body, scientists and technical experts should strive continuously to ensure safety by recognizing risk involving nuclear power and reducing such risk to as low as reasonably achievable. Scientists and technical experts in the nuclear field have engaged themselves in exchanging academic information and ideas, however have not been proactive in providing and communicating risk information involving the use of nuclear power to the public. This was what led to the development of the safety myth and the continued presence of adverse influences, and is a lesson that should be engraved in the minds of scientists and engineers. All scientists and technical experts should not only fulfill responsibility commensurate with the assigned roles, but are accountable for, and should proactively communicate information on risk involving nuclear power to the operators, the regulatory body and the public. To the operators and the regulatory body, scientists and technical experts should indicate specific influences on risk by the application of new expertise and technologies and encourage their examination; verify the logical bases of the safety aspects and prompt re-examination if they contain uncertainties or are questionable, to prevent the development of another safety myth. To the public, information on the status of risk assessment and risk management, significance of new expertise, status of the application of new expertise, significance of the safety goal, etc., should be communicated. These efforts are critical fundamentals in formulating a shared understanding of nuclear power in Japan.

Because there are limits to individual efforts and capabilities, professional societies may play a significant role in integrating and enhancing the efforts of the scientists and engineers. Professional organizations as societies and academic councils should establish dialogue with the national public for formulating common understanding on nuclear risk.

6. SEVERE ACCIDENT MANAGEMENT

6.1 Utilization of Risk Information

Foreseeing the “unforeseen” is imperative in ensuring safety of nuclear facilities. A framework for extracting and examining all likely scenarios of incidents and their measures caused by natural hazards, human events and internal events that can be assumed should be established. It should include a thorough re-examination of TEPCO’s Fukushima Dai-ichi NPP accident, incorporating the lessons learned for enhancing measures for defense-in-depth level 4 (accident management) and defense-in-depth level 5 (emergency preparedness and response). The framework should also include a continuous process of identifying and applying important new findings and results of researches to nuclear facilities even after the enhancements have been made. For this purpose, a comprehensive assessment method for evaluating a broad spectrum of accident initiators, accident management and emergency preparedness and response measures should be developed. Probabilistic Risk Assessment (PRA) and risk information obtained through PRA are extremely effective means for this assessment.

There are some negative arguments that “the PRA method is not practical because of the uncertainties it contains”, “probabilistic theory is difficult to explain to the general public”, and “no matter how small the probability, accidents do occur (OR) likelihood of an accident may not be eliminated”. However, these criticisms may be founded on the lack of common understanding on the significance and utilization of PRA. The basis to the significance and utilization of PRA are described as follows.

PRA is a safety assessment technique where scenarios on postulated accidents at nuclear facilities are systematically extracted and subsequently, the occurrence frequency and consequences of accidents are evaluated and the safety level of nuclear facilities are presented as public risk. PRA on nuclear power plants are carried out in accordance with the scope of assessment; Level 1 PRA in assessing accident scenario leading to core damage and its occurrence frequency; Level 2 PRA for assessing accident scenario leading to containment damage and its occurrence frequency, as well as assessment on the amount of radioactive material release to the environment (source term); and Level 3 PRA for assessing environment impact caused by radioactivity release. Procedures and applicable areas of Level 1 to Level 3 PRA are shown in Fig. 6-1 (note: Level 1 to Level 3 PRA do not correspond to defense-in-depth level 1 to 3).

Because PRA assessment takes into account unlikely events and enables identification of accident scenarios important from risk perspective, relative weaknesses of the safety measures may be clarified for enhancing the measures. This is the general idea in utilizing PRA, and are the grounds to the significance of PRA in severe accident management. Reliability of the safety SSCs as well as

maintaining and enhancing accident management capabilities of the operating staff are essential under normal operating conditions. Through PRA, SSCs that are important for ensuring safety amongst the system configuration that is complex, as well as operating manipulations that are critical in preventing accidents and mitigating effects of accidents can be identified and quantitatively assessed. The results of quantitative assessment may be used to reinforce weaknesses in the design, reduce unnecessary safety SSCs and allocate maintenance management resources to SSCs with higher significance. Safety may be ensured and enhanced in a cost effective manner. As well, by reflecting records on SSC failures and malfunctions on component failure rates which is utilized in PRA, component reliability trends can be monitored. Highly scientific and rational maintenance planning can be achieved through the establishment of probabilistic safety goal and relevant SSC management goal (to maintain reliability of facilities in meeting the probabilistic safety goal) for nuclear facilities by the regulatory body or the operators. Progression of events and failures leading to a severe accident may induce beyond design basis phenomena, overheat and overpressure of SSCs. It is important that the limit capacity under these conditions are assessed from defense-in-depth perspectives for developing reasonable practicable measures. For the assessment, Level 2 PRA technique and its results may be useful. Development of performance goals on the capacity of SSCs under severe accident conditions on the basis of probabilistic safety goal should be examined.

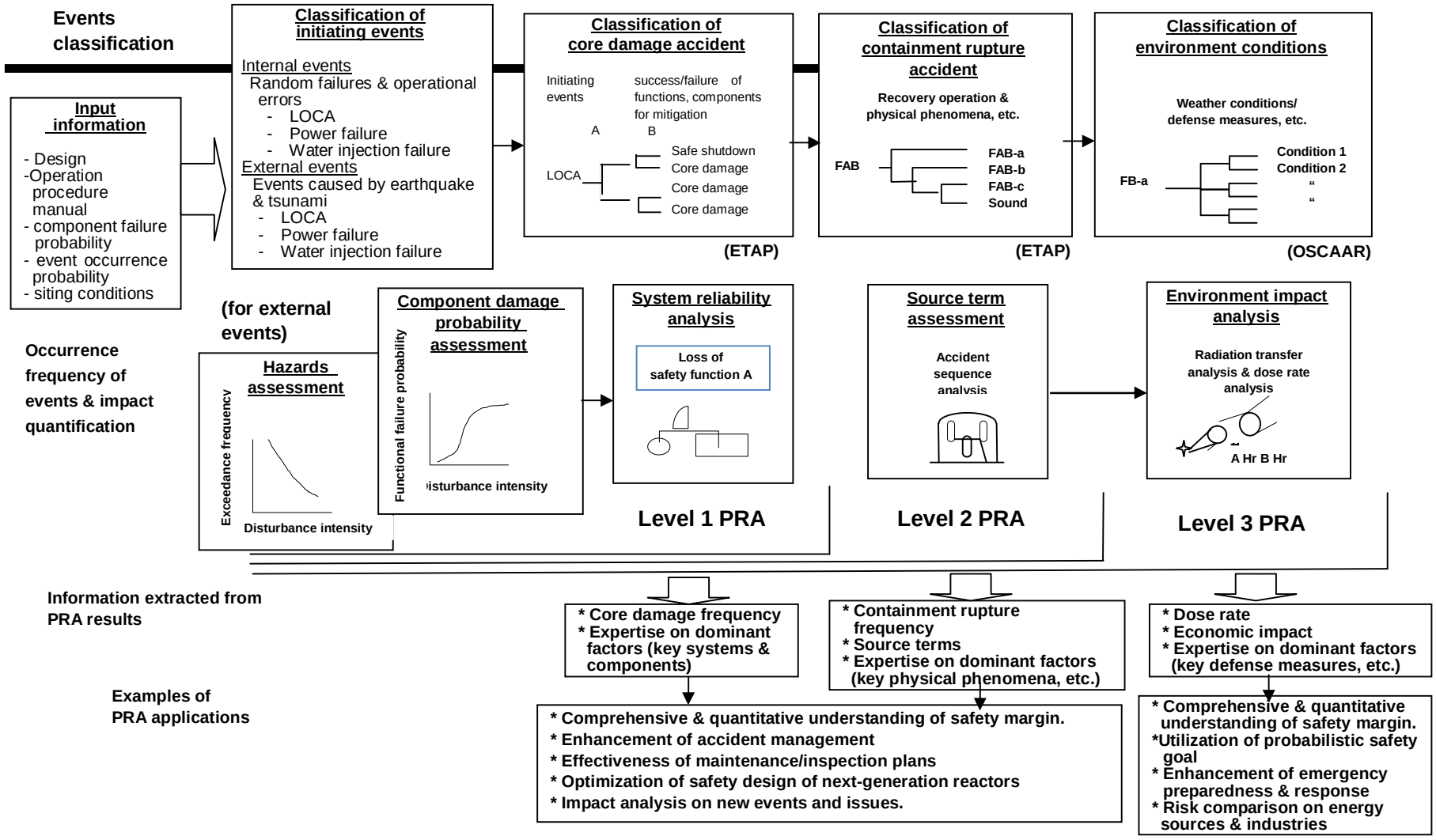
TEPCO's Fukushima accident has brought to light the significance of developing and reinforcing preparedness and response planning on the basis of various severe accident scenarios. For this purpose, effective measures against a broad spectrum of scenarios should be examined and developed, for which Level 3 PRA can be an effective tool.

Criticism against PRA as "being not practical because it contains uncertainties" arise from cases where the PRA is used for making judgment on whether a certain facility is safe, or not based on the values on core damage frequency (CDF) or containment vessel failure frequency (CFF) obtained through PRA. The scope of PRA in Japan extends only to internal events and earthquakes. PRA for events such as tsunami, fire, flooding, etc., are still in the development process, and even when they have been established, uncertainties on the assessment results still remain. There are also other events not given consideration in PRA, and thus, it would be difficult to determine safety based purely on the numerical results of PRA. However, PRA can be a useful information source in regulatory decision-making, for example, semi-quantitative information such as the occurrence frequency of beyond design basis earthquakes; uncertainties contained in beyond design basis earthquake occurrence frequency; risk associated with such earthquake; accident scenarios initiated by such earthquake, etc. Results of such assessment can be used as objective indices in judging whether the occurrence probability of unanticipated external initiators are extremely low, or not, which can be utilized by the operators and the regulatory body for confirming validity of their decisions.

There are still many issues to be resolved in PRA - not only the improvement of assessment technique for external events, but for internal events such as reliability assessment on human, mechanical and digital systems, quantitative assessment on common cause failures, etc., as well as organization of the database. The application of PRA should be promoted regardless of these issues with due consideration given to the limitations on the scope of assessment. In addition, the occurrence probability of various severe accident scenarios and the magnitude of disaster caused by these scenarios that can be assumed under the scope of Level 2 and Level 3 PRA should be disclosed to the public.

In “The Report of the Japanese Government to the IAEA Ministerial Conference on Nuclear Safety” issued after TEPCO’s Fukushima Dai-ichi accident, Lesson 27 “Effective use of probabilistic safety assessment in risks management” cites that “PSA has not always been effectively utilized in the overall review processes on risk reduction efforts at nuclear power plants. While a quantitative evaluation of risks associated with rare events such as a large-scale tsunami is difficult and may contain uncertainty even in PSA, Japan has not made sufficient efforts to improve reliability of the assessments by explicitly identifying the uncertainty of these risks. On the basis of knowledge and experiences regarding uncertainties, the Japanese Government plans to further actively and swiftly utilize PSA while enhancing safety measures including effective accident management measures based on PSA.” (note: PSA is synonymous with PRA; the original term is PRA used). As lessons learned from the 3.11 accident, the above should be implemented.

Fig. 6-1 Probabilistic Risk Assessment Procedure and Areas of Application (Example)



6.2 Severe Accident Management

In view of the IAEA's defense-in-depth concept described in Section 4, ensuring safety comprises of managing both design basis and beyond design basis events.

The key point in preventing severe accident through design is in clarifying the correlation between each system. The significance of the clarification is proportional to the complexity of the systems. Sequence of events in accident progression and relevant measures for each event can also be complex. Accordingly, an integrated safety assessment framework that takes into consideration the correlation and interactions between each system needs to be established (refer Section 4). The integrated framework will greatly enhance safety from the design stage to actual operation. As well, tightening design standards simply for the sake of enforcing safety may not benefit safety in the long run. The intent of design standards is warranting safety against all anticipated circumstances within design basis. This may create heavily equipped safety components not in proportion to the level of safety ensured considering the resources allocated. In particular, the type of external events and their magnitude to be considered in the design standards is something that should be determined on the basis of public consensus. For example, whether past earthquake records should be referred to, or extended in determining the scope of design seismic motion; as well, the scope of consideration in fault assessment. Obviously, measures for controlling risk on rare events with small occurrence probability are necessary. However, an appropriate set of design basis standards should be established, and the uncertainties, or residual risk contained should be evaluated, so that measures may be developed for reducing such risk on the basis of public consensus.

Thus, management of beyond design basis events is the key issue that needs to be addressed. The intent of accident management is controlling extension to severe accidents. Once-in-a-thousand-years events do occur as did TEPCO's Fukushima Dai-ichi accident. In dealing with beyond design basis events in equivalence to the magnitude of the Fukushima event, accident management framework for tsunami, earthquake and simultaneous occurrence of earthquake and tsunami should be established on the basis of extensive scenarios, with relevant safety SSCs and procedures organized and placed, and measures to manage each scenario.

Further, the organization of these procedures is expected to be diverse and complex. Given the issues on the education and training of operating staff, automated systems to compliment and reinforce human competence and judgment capability, and electronic calculation systems that provide directions on measures and procedures should be developed, to which state-of-the-art information technology (IT) should be applied.

Development of personnel to manage and lead severe accident management process is essential. The assignment of high quality personnel and dedicated professional staff (Chief Engineer of Reactors) with not only good understanding of reactor conditions, but have adequate judgment capability and logical reasoning should be established as a regulatory requirement. In addition to the

appointment of Chief Engineer of Reactors, enhancement of general operating staff should be carried out. The introduction of automated system described above should synchronize sharing of information and actions, and make possible an integrated accident management system.

Resilience, for controlling development of accident sequence to severe accident, managing the recovery and ensuring of safety is a key factor in accident management. A systematic framework where processes, procedures, and safety SSCs are organized in advance to enable prompt actions in the event of an accident is called resilience engineering.

Accident management extends to the recovery of functional failures caused by an accident. There are specific time limits (coping time) for the recovery of minimum safety functions to control extension to extensive core damage and melting. Accident management governs the requirements on which functions need to revive to which level within limited time frame to ensure safety. Accordingly, the two key factors of accident management (by resilience engineering) are the level of functionality required of the minimum safety functions to maintain safety, and the time limit for the recovery to the required level.

In dealing with malfunctioning and functional failures caused by hazards, resilience engineering focuses not on managing the magnitude of a hazard but on evaluating resilience of the systems and proposing means for the recovery of minimum safety functions to functional level and its assessment method. Resilience engineering bears on the formulation of a systematic framework for accident management and enables quantitative assessment on its contribution to nuclear safety.

Specific examples of accident management are:

1) Provision of diverse and redundant means for safety significant SSCs
2) Substituting required functions with the functions of other systems
3) Ensuring safety through provision of adequate training and procedure manuals, and the alternate use of human operation and mechanical operation, including automated systems.

On the basis of the above, the requirements on the recovery of specific functions to certain levels within certain time scope for ensuring integrity of the entire plant system should be determined with care and consideration.

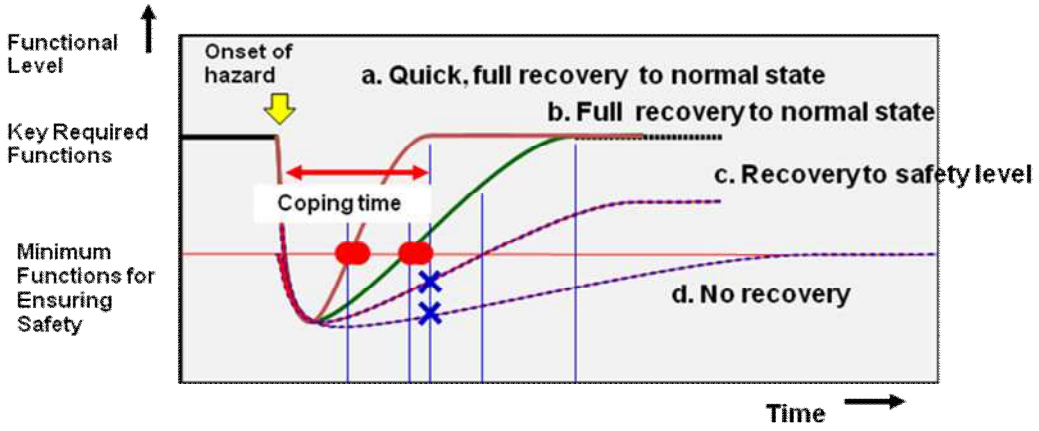


Fig. 6-2 Accident Management by Resilience Engineering

6.3 Leadership, Assignment of Roles, Clarification of Responsibilities and Collaboration

Regulatory standards and policies are stipulated to be revised promptly with the accumulation of new expertise in Japan. However, the revision process has been generally slow as shown by the 10-year review process on "Regulatory Guide on Seismic Design Safety Review" illustrated in Fig. 6-3 because of the specific nature of the field as well as diversity in the opinions of the committee members. Data accumulated on large-scale seismic motion of the 1995 Hyogo Prefecture Nambu Earthquake that struck the densely populated regions led to the revision of the seismic design safety guide stipulated in 1981. Finally, in 2006, after a period of about 10 years, a number of revisions including the extension of design seismic motion, modification of calculation method, introduction of "residual risk", etc., were made in the guideline. For applying the changes in the regulatory guidelines to operating plants, back-checks currently left to the voluntary discretion of the operators is under review for inclusion in the regulatory requirement. Although revisions on the guidelines have been made out of necessity, it is generally difficult, and takes a long process to modify and improve something already in place under the Japanese cultural climate. Fixed ideas and belief in the flawlessness of what had been determined dwell in the minds of many which is what must have hampered actions in making revisions or doing reforms. This is why strong leadership in fulfilling responsibility for safety is required. The current back-fitting issue exhibits the deep-seated structure of circumventing responsibility. If flexibly dealt with, the application of back-fits will certainly contribute to enhancing safety. [21)]

The key issue is that the responsibility for ensuring safety (commensurate with the assigned roles) of those involved including the regulatory body is ambiguous. The government took an approach in formulating numerous review committees for evaluating regulatory codes and standards, which created circumstances of blurred responsibility of individuals involved in the decision making. The awareness on the responsibility of the individual for the decisions made as a group by members of

the committee is not apparent. The roles and responsibilities of those involved including not only the regulatory body, but the operators, manufacturers, supporting organizations, professional societies, the academia in the nuclear community have not been defined with clarity, which accounts for the lack of leadership in the decision-making process so far, and the underlying cause that led to the severe accident.

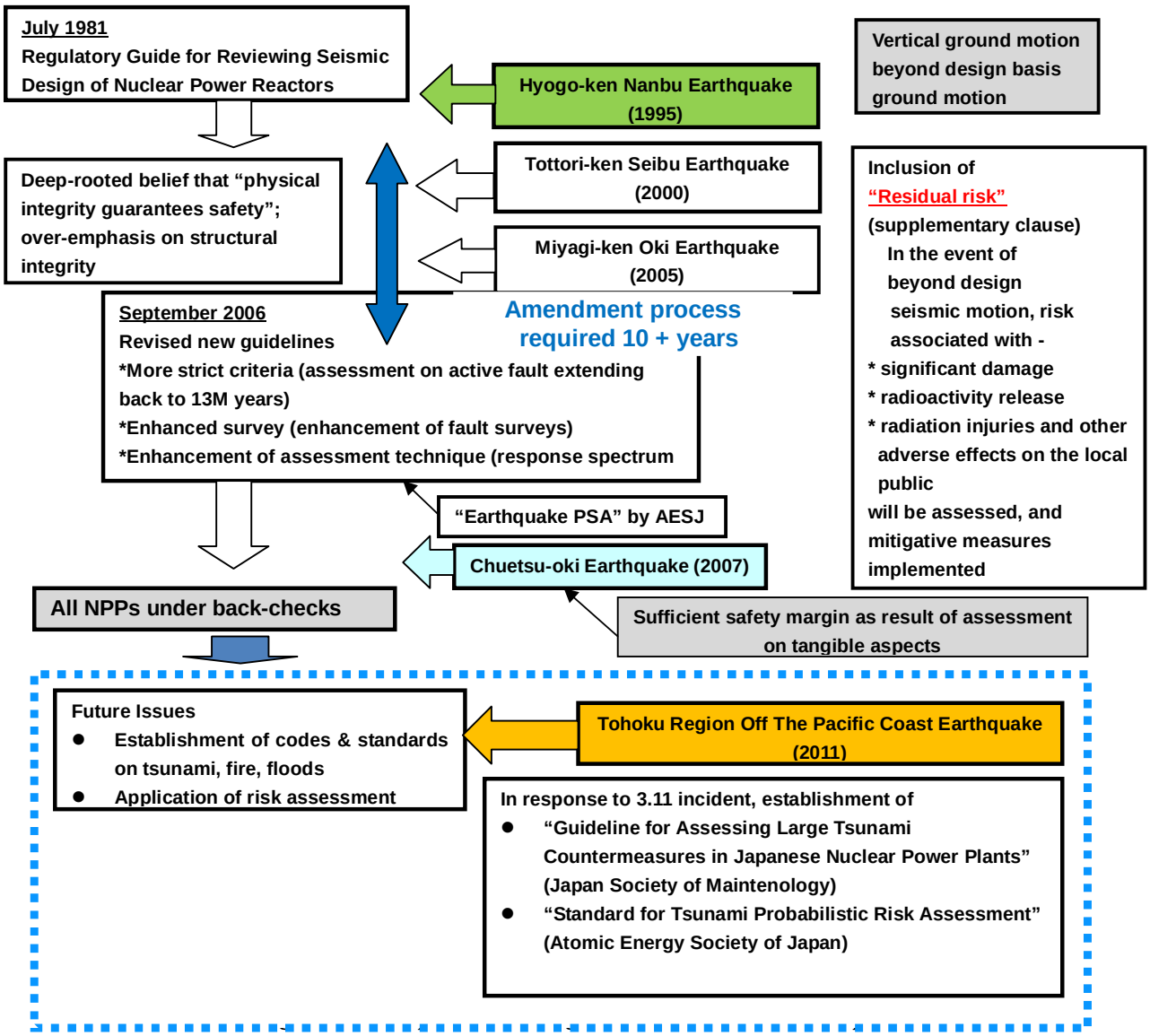
The newly established organization, Nuclear Regulation Authority and the Chairperson is expected to demonstrate strong leadership. In addition (Furthermore), the roles and responsibilities of individuals, of the Nuclear Regulation Authority, the nuclear industries, government organizations, the academia, professional societies, and all other related entities should be defined with clarity, the discussions on which should take place right away. Currently, the effective independence of the regulatory body is being emphasized. However, it is not only the regulatory body with a commitment for ensuring safety. Collaborative relationship, not back-scratching alliance nor isolation is important for promoting “nuclear safety”, by sharing information and new expertise and exchanging opinions.

Many are the issues awaiting to be resolved - such as fostering of safety culture, development of personnel exchange, enforcement of qualification systems, etc.

The development of safety culture involves the entire society. The approach to nuclear safety was described in Section 4. Public consensus has not yet been established on the treatment of risk which requires considerably more efforts, without which safety culture will not be nurtured.

Regarding personnel exchange, there are basically no interactions, exchange of communications between the regulator, the regulated and the third part who make fair and impartial judgment or assessments, for example in personnel development and training. This is a general tendency in Japan. From a fair perspective, all parties should take a sincere approach in ensuring “nuclear safety” regardless of the position or the policy of the party. However, customarily in Japan, people are judged by the organization they belong to and are expected to think in accordance with the stance of the organization. This is due to the Japan-specific social structure and characteristics of the Japanese which has hampered human resources development. Work force mobility, personnel development and exchange should be promoted to grow individuals with competence to manage various conditions and circumstances. This must certainly be one of the important lessons learned from the 3.11 incident.

With view to the vulnerabilities shown in the management of the Fukushima Dai-ichi accident, severe accident management specialist system should be established. At the same time, sufficient education and training should be provided for enhancing quality and competence of the operating staff. Those selected as severe accident management specialists will have prime responsibility for safety, and will clarify the responsibilities commensurate with assigned roles with due consideration given to the complexity and risk involving nuclear power plant.


July 1981
Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactors
Hyogo-ken Nanbu Earthquake (1995)
Vertical ground motion beyond design basis ground motion
Deep-rooted belief that "physical integrity guarantees safety"; over-emphasis on structural integrity
Tottori-ken Seibu Earthquake (2000)
Miyagi-ken Oki Earthquake (2005)
Inclusion of "Residual risk" (supplementary clause)
In the event of beyond design seismic motion, risk associated with -
* significant damage
* radioactivity release
* radiation injuries and other adverse effects on the local public
will be assessed, and mitigative measures implemented
Amendment process required 10 + years
September 2006
Revised new guidelines
*More strict criteria (assessment on active fault extending back to 13M years)
*Enhanced survey (enhancement of fault surveys)
*Enhancement of assessment technique (response spectrum
"Earthquake PSA" by AESJ
Chuetsu-oki Earthquake (2007)
All NPPs under back-checks
Sufficient safety margin as result of assessment on tangible aspects
Future Issues
Establishment of codes & standards on tsunami, fire, floods
Application of risk assessment
Tohoku Region Off The Pacific Coast Earthquake (2011)
In response to 3.11 incident, establishment of
"Guideline for Assessing Large Tsunami Countermeasures in Japanese Nuclear Power Plants" (Japan Society of Maintenology)
"Standard for Tsunami Probabilistic Risk Assessment" (Atomic Energy Society of Japan)

7. SPECIFIC EVENTS THAT SHOULD BE GIVEN CONSIDERATION IN THE PREVENTION OF SEVERE ACCIDENTS

7.1 Risk Assessment on Accident Scenarios

The examination of accident sequence leading to severe accident and associated risks are extremely important in preventing and mitigating severe accidents.

Accident sequences initiated by internal events leading to a severe accident has been examined from the early stages of discussions on accident management. However, on the basis of the advancement of probabilistic risk assessment (PRA) technique, the operators should continue to extract and categorize important accident sequences and implement PRA with consideration given to specific characteristics of each plant, and report the results of assessment and measures applied to the regulatory body. The PRA should include accident sequences caused by internal fire and cyber terrorism (undisclosed due to confidentiality of terrorist measures).

Although accident sequences on external events (natural phenomena and human events) leading to a severe accident have been evaluated by earthquake PRA, those for tsunami including internal flooding has only started after TEPCO's Fukushima Dai-ichi incident, and have not been sufficiently established. Accident sequences caused by other initiators have not been practically developed at all. With some of the events, PRA is difficult to carry out, or the results of assessment are assumed to be not highly reliable. A comprehensive assessment will be made on these events by conducting impact analysis using event trees and by referring to accumulated expertise through past accident sequence assessment on internal events.

Needless to say, the regulatory body is responsible to thoroughly examine and evaluate the details of the assessment and related accident management measures implemented by the operators, to confirm whether measures for preventing and minimizing consequences of core damage event are sufficiently established. For this purpose, the regulatory body needs to clearly determine safety goal or regulatory criteria on nuclear safety.

7.2 Internal Events that May Induce Severe Accidents

So far, the following internal events for BWR has been considered for sequences leading to core damage - large-break LOCA, off-site power failure, scram failure, water injection failure, small-break LOCA, transient events, failure of HPCI, LPCI and decay heat removal systems, etc.. For PWR, small-, medium- and large-break LOCA, steam generator heat exchanger tube rupture, secondary piping rupture, loss of off-site power supply, loss of primary feed-water system, transient events, ATWS, etc. The sequences branch off depending on the integrity of engineered safety components with progress of events; however, redundancy failures and common cause failures should also be given due consideration in the assessment.

With reference to the lessons learned from TEPCO's Fukushima Dai-ichi NPP accident, other initiators that should be given consideration are SBO, internal fire, multiple rod ejection accident, cyber terrorism, spent fuel pool loss of coolant accident, etc.

7.3 External Events that May Induce Severe Accidents

The following items related to natural phenomena and human events needs to be examined as external initiators leading to severe accidents.

1) Natural Phenomena

- Earthquake:
  Seismic events exceeding design seismic motion. Results of stress tests conducted at each plant should be examined to confirm plant integrity against severe accidents, and to develop mitigation measures.
- Tsunami:
  Tsunami events exceeding design tsunami height. Items such as the integrity of SSCs installed outdoors, inundation measures of key structures, etc., in the event of tsunami overflow of tide embankment should be assessed. Combined disaster of earthquake and tsunami should be given consideration.
- Meteorological Impacts
  Damage to SSCs installed outdoors caused by wind storms, hurricanes, fires, sand storms, tidal waves; river overflows, flooding, sediment flows, flashfloods, landslides, rock falls caused by intense rainfall including rain storms; load impacts, avalanches, snow storms caused by heavy snow; landslides caused by rapid snow melts; large electric currents and fires caused by lightening strikes; extreme heat or cold temperature (freezing point); and sudden changes in seawater level. The scope of consideration on the above weather conditions should not be limited.
- Volcanic Activities
  Volcanic bombs, volcanic lapilli, pyroclastic flows, lava flows, debris flows, volcanic blasts, volcanic ash falls, volcanic gas retention, in particular, ash fall impacts on the intake systems.
- Meteorite Falls
  Probabilities of harmful consequences caused by meteorite falling on, or in the vicinity of nuclear facilities, including shock wave impacts are extremely small. No agreement has been reached globally on including the item in external events.
- Biological Impact
  Influences on seawater intake facilities caused by massive outbreak of Nomura's Jellyfish blooms.

2) Human Events

- Fire and Explosion

  Influences of the outbreak of fire and explosion outside reactor buildings and offsite boundaries.

- Collision and Stranding of Ships

  Damage to underwater facilities and facilities located near seashore, loss of seawater intake due to crude oil spills

- Aircraft Crash

  Accidental – military, small- and large-size aircrafts; excluded from assessment target if occurrence probability is 10^{-7} times/reactor-year.

  Terrorism – large- and small-size aircrafts

  Corresponding fire outbreaks should be given consideration

- Sabotage (Terrorism)

  Acts of sabotage involving fire and explosion through the use of explosives and gasoline; physical impacts involving cable disconnection, destruction of central control room and other crucial safety facilities; human impacts associated with the use of toxic gas and viruses of infectious diseases, etc.

- Cyber Terrorism

  Influences on information network systems as cyber terrorism.

Events with very small occurrence probability should be clearly indicated. A variety of measures should be combined for preventing and mitigating consequences of severe accidents. Response measures should be founded on human competence and knowledge, utilization of existing facilities and offsite support, etc.

8. SPECIFIC EXAMPLES OF SEVERE ACCIDENT MANAGEMENT

Examples of measures against tsunami will be given for a better understanding of severe accident management. Fig. 8-1 shows accident management items that have been established, or in the implementation process by the regulatory body in the wake of TEPCO's Fukushima Dai-ichi NPP incident. The draft accident management measures have been formulated on the basis of scenarios on earthquake and tsunami as representing natural disasters in Japan.

8.1 Accident Management Reflecting Lessons Learned from TEPCO'S Fukushima Dai-ichi NPP Accident

In TEPCO's Fukushima Dai-ichi Nuclear Power Plant accident (hereinafter, "TEPCO's Fukushima Dai-ichi accident"), although all of the reactors automatically scrammed after the earthquake, reactor cooling could not be maintained due to SBO, which led to the extensive release of radioactive material to the environment. Maintaining reactor cooling capability is the fundamentals of nuclear reactor safety. Vent system (filtered vents[5]) with air clean-up unit is provided to prevent over-pressure of the containment and to control extensive radioactivity release, recommended from environment protection perspectives.

The primary concern in preventing severe accidents and mitigating their influences is ensuring fuel cooling capability, including integrity of the ultimate heat sink, and controlling radioactive material release to a socially acceptable level. In Japan, operating plants use seawater and atmosphere as ultimate heat sink, and SSCs need to be placed outdoors for this purpose. This is why external events directly influence ultimate heat sink capabilities. The integrity of safety facilities installed indoors may be maintained by enhancing strength, air-tightness and water-tightness of buildings against natural phenomena and human events recommended to be included in accident sequence assessment.

However, under the combined influences of multiple events, it is very difficult to ensure fuel cooling capability including the ultimate heat sink. Portable power source equipments, pumps and temporary piping systems, in addition to permanent SSCs should be utilized for cooling water intake from the rivers and the sea. Further, a flexible system of transporting cooling water, power sources and fuel via land, air and sea according to circumstances, together with provision of training on these arrangements should be established.

In the wake of TEPCO's Fukushima Dai-ichi NPP Accident, the Nuclear and Industrial Safety Agency investigated and made analysis of the accident, and developed 30 measures in five different areas which should be reflected in the regulatory requirements in February 2012, (refer Fig. 8-1) [22)], the areas of which are 1) measures for external power line (4 items); 2) measures for onsite electric facilities (7 items); 3) measures for cooling and water injection facilities (6 items); 4) measures for

containment rupture and hydrogen explosion (7 items); and 5) measures for instrumentation & management facilities (6 items).

Many of the items in the analysis may be effective in preventing and mitigating different types of severe accidents than the one that occurred in TEPCO's Fukushima Dai-ichi NPP. Technical expertise and safety measures in the analysis are those for BWR type reactors, however, may be applied to PWR and BWR reactors of different types than the ones in Fukushima Dai-ichi NPP. Key SSCs including power supply equipments for preventing and mitigating influences of severe accident should be permanent facilities with redundancy, diversity and independence for ensuring reliability as with existing safety facilities.

8.2 Application of Lessons Learned to Other Plants

Accident management measures implemented, or in the process of implementation at Hamaoka NPP of Chubu Electric Company, where the magnitude of earthquake and tsunami hazards are considered to be the severest, on the basis of lessons learned from TEPCO's Fukushima Dai-ichi NPP accident will be shown as an example.

1) Earthquake and tsunami assessment method

In the past, 1096 Eicho Earthquake (M. 8.3), 1498 Meio Earthquake (M. 8.3), and 1854 Ansei Tokai Earthquake (M. 8.4), with epicenter of 34 degrees north latitude off the coast of Enshunada have occurred in the vicinity of the Chubu Electric Company's Hamaoka NPP. From these experiences, Hamaoka NPP set forth maximum ground acceleration of 800G as design basis seismic ground motion Ss, larger than the design values of Fukushima Dai-ichi NPP. Hamaoka NPP further voluntarily determined seismic safety goal of 1,000G and completed seismic reinforcement work on Unit 3 to Unit 5 in March 2008.

On the basis of "Tsunami Assessment Method for Nuclear Power Plants in Japan", issued by the Japan Society of Civil Engineers in 2002, Fukushima Dai-ichi NPP established design tsunami height of 6.1 meters in 2009. However, after the Tohoku Region Off The Pacific Coast Earthquake, earthquake and tsunami experts conducted analysis on large-scale earthquake and tsunami simulating the observed records of the 3.11 event, and found that "the design scale tsunami generated by a number of fault movements along the Japan Sea Trench (small-scale earthquake) overlapped and expanded to an unanticipated height (15 meters) off the coast of Fukushima Dai-ichi NPP". In view of the analysis, the Examination Committee on Large-Scale Earthquake in the Nankai Trough of the Cabinet Office established under the Natural Disaster Council developed a new tsunami fault model simulating the largest tsunami induced by M. 9 consecutive earthquake on the basis of Tohoku Region Off The Pacific Coast Earthquake and Tsunami, larger than the previous simulation of M. 8.5 before Fukushima Dai-ichi NPP accident and released the results. The tsunami

model presented amplitude of the fault slip on the basis of surveys on analyses of 2011 Tohoku Region Off The Pacific Coast Earthquake, 2010 Chile Earthquake, 2004 Sumatra Earthquake, and further added conditions of larger tsunami height than was generally used, and made assessment on the largest scale tsunami. The tsunami model evaluated the maximum tsunami height in the vicinity of Hamaoka NPP at approximately T.P. (the mean sea level of Tokyo Bay) + 19 meters [23].

For reference, the tsunami of 1854 Ansei Tokai Earthquake in the Enshunada region facing Hamaoka NPP is assessed as the largest of previous tsunami events. The maximum tsunami trace height at high tide was reported as approximately 6 meters in the areas between Omaezaki, on the east side of the plant and Shirasuga, western region of Hamanako. The Central Disaster Council assessed tsunami trace height at T.P. + 7 meters in 2003, showing great variance with the results of the new simulation model.

If the new tsunami simulation induced by consecutive earthquake in the Nankai Trough predicted to occur in the near future is adopted by the central, prefectural and municipal governments, the cost on the development of infrastructure for disaster prevention and mitigation in the Tokai, Kinki, Shikoku, Kyushu regions facing the Nakai Trough will be tremendous. A more realistic and practical tsunami model with consideration given to the balance between benefit and cost should be developed for a reasonable infrastructure planning. The new tsunami model established by the National Disaster Council should be applied to safety assessment of nuclear plants in Japan.

2) Measures for Power Supply

Immediately after TEPCO's Fukushima Dai-ichi accident, NISA issued orders on measures for emergency onsite electric components, on the installment of large-size emergency power generators and power source cars on higher grounds for all plants with BWR and PWR reactors as measures for external events of not only earthquake and tsunami, but for fire, explosion, typhoons, etc., as well. Measures No. 1 to No. 4 (refer Fig. 8-1) - enhancing reliability of external power line; enhancing seismic resistance of substations and switching stations; and prompt recovery of external power facilities, have been formulated on the basis of technical assessment on TEPCO's Fukushima Dai-ichi accident, and are in the process of being implemented at each plant.

Chubu Electric Company's Hamaoka NPP is planning to construct a new seismic resistant building on high grounds (T.P. + 40 meters) which will not be affected by the tsunami for six 4,000 kVA high-capacity gas turbine generators (with redundancy, ensuring 2 weeks of fuel). Power cable via seismic resistant underground duct will be connected on a permanent basis to the power panel components (including switching panels) installed on the upper floors of the reactor building with inundation proofing (Measure No. 5: distributed arrangement of onsite electric facilities; No. 6: enhancing inundation measures; No. 7: enhancing redundancy & diversity of emergency AC). The power connection is remotely controlled and may be activated from the central control room. On

high grounds, a building/warehouse for stockpile of emergency electrical equipments and other materials and equipments will be established. Distributed arrangement of common power supply terminals (Measure No. 10: facilitation of external power supply; No. 11: stockpile of backup equipments for electric facilities) is in progress to ensure response to failure and damage of key emergency components and parts. In addition to the current AC power sources, backup AC (including exclusive battery charger) with the same capacity will be installed in each reactor (Measure No. 8: enhancing emergency DC; No. 9: deployment of independent, exclusive power line)[24]. Currently, installments of similar SSCs with consideration given to fire and explosion are making a rapid progress in all nuclear plants in Japan.

3) Specific Measures for Prevention and Mitigation of Severe Accidents

The directions on measures for preventing and mitigating accidents related to loss of cooling and water injection capabilities, containment vessel rupture, hydrogen incident (Measures No. 12 to No. 24) issued by NISA immediately after TEPCO's Fukushima Dai-ichi accident for BWR and PWR reactors included the arrangement of portable pumps, temporary underwater pump, alternative heat exchanger vehicle, hydrogen extraction equipment, etc., with consideration given to external events of not only earthquake and tsunami but fire, explosion, typhoon, etc., as well.

On the basis of the new tsunami assessment model developed after TEPCO's Fukushima Dai-ichi accident, Hamaoka NPP established bulkheads (3 meters in height) in the areas where outdoor seawater intake pumps are installed, in the event of overflow or flooding of water intake tanks over the existing breakwater (at sea level of 22 meters). Further, emergency seawater intake facilities were installed in each unit - specifically, two emergency seawater intake pumps were installed in newly established wave-proof, watertight reactor buildings of each unit to maintain integrity of seawater intake for core cooling. The seawater pumps can be quickly activated by remote operation from the central control room (Measure No. 13: ensuring inundation proofing, and distributed arrangement of cooling facilities).

Inundation measures at Hamaoka NPP for reactor buildings, etc. are double layers of watertight and reinforced doors on building exteriors; elevation of the ventilation terminals of emergency diesel generators with ventilation snorkels; application of water cut-offs and blank flange for waterproofing, water-pressure resistance and seismic resistance purposes to the penetrations of the buildings (as pipes). Measures against inundation of SSCs related to core cooling capabilities (including emergency diesel generator), and fuel pool cooling capabilities should be arranged (measure No. 13: ensuring inundation proofing, and distributed arrangement of cooling facilities).

Through the "enhancement of DC power supply and inundation measures", reliability of the RCIC "driven by steam generated from core decay heat" that played an active role during TEPCO's Fukushima Dai-ichi incident can be tripled. Hamaoka NPP has newly installed air-cooled heat

exchanger for high pressure pumps on the upper part of the mid-floor of the reactor building so that HPCI can be activated in the event of failure of both the existing seawater intake pump system and newly installed emergency seawater intake pump system. Also renovation on the reactor residual heat removal systems is planned to enable connection to alternative heat exchanger vehicles (measure No. 14: enhancement of final heat sink).

In addition to the current condenser tank and mass condensate water storage tank, a large-capacity emergency freshwater tank will be installed on high grounds (T.P + 30 meters) of the plant site for injecting water into the reactor and the fuel pool. The freshwater tank component comprises of 9,000m^3 concrete water tank and a pump room, with a motor-driven pump via gas turbine generator and a diesel-driven pump (diversity) that will feed water to the reactors of each unit and the spent fuel pool. In addition, approximately two weeks of water supply for cooling the reactor core and spent fuel pool may be ensured by filling the pilot tunnel (established to obtain bedrock data required for reactor establishment application) of Unit 3 with fresh water (Measure No. 16: enhancement of alternative water injection capabilities, No. 17: enhancing reliability of cooling & water make-up capabilities of spent fuel pool).

As such, reactor containment spray capabilities (Measure No. 18) and containment top flange cooling capabilities (Measure No. 19), which are provisions for mitigating core damage may be enhanced through redundant and diverse provisions of emergency power sources, water injection and final heat sink. Further, alternative heat exchanger vehicle may also be used for mitigation measures (Measure No. 18: diversification of containment heat removal capabilities). Through these measures, containment vessel damage experienced in TEPCO's Fukushima Dai-ichi accident may be prevented (or minimized significantly). Even in the event of a core damage causing hydrogen generation, leading to radioactive material release to the containment vessel, by combining hydrogen venting, containment spray system, dilution in containment pool, filtered containment vent can significantly mitigate impacts of radioactive material release to the environment (Measure No. 22: mitigating influence to the environment by venting; No. 24: prevention of hydrogen explosion). [24)]

Measure No. 25 to No. 30 have been developed as management measures and instrumentation equipment measures under emergencies including mitigation of core damage incident. These have been formulated on the basis of the lessons learned on the significance of collecting important data such as conditions of the reactor and the core, radiation dose monitoring of the site and the surrounding areas; and sharing of information and establishing communication onsite and between related parties and organizations. All plants have initiated implementing these measures.. However, many issues remain to be addressed including the development of instrumentation method.

Needless to say, the preconditions for the effectiveness of measures for preventing and mitigating severe accidents is that the operating staff have been provided sufficient education and training, have competencies to understand accurately conditions under emergencies, and the ability to flexibly

combine the uses of portable and permanent SSCs.

8.3 Safety Margin and Filtered Vents

With the failure of all motor-driven pumps and loss of heat transfer from reactor core to final heat sink for core cooling under an SBO, the only mitigation measure left was the discharge of steam generated in the containment by alternative water injection for heat removal of the reactor core, which was what happened at TEPCO's Fukushima Dai-ichi accident.

The most difficult decision that had to be made in the accident was injecting seawater into the reactor for cooling and releasing containment atmosphere (containment vessel vent) that contained steam (radioactive material) generated by core cooling.

The initiators of TEPCO's Fukushima Dai-ichi accident were SBO and the loss of core cooling.

Since the reactor system (primary containment system) and the turbine system (secondary containment system) of PWRs have independent cooling systems separated by the steam generator, the operating staff will not feel pressured in releasing steam relieve valve of the secondary containment for reactor cooling because it does not contain radiation. Water to the steam generator will be injected via secondary steam-driven pump from the secondary make-up water line. Release of the steam into the atmosphere via the steam generator will generate natural circulation for core cooling. Analyses have shown that it will take a half to one day for reactor pressure and coolant temperature to stabilize through this process [25)]. For a subsequent cold shutdown, means such as fire engine pumps will be used to inject water for cooling. In the event of loss of DC power source, due care should be taken so that the valve manipulation of the steam-driven pumps and that for controlling nitrogen injection of the accumulator system maintain integrity.

RHR needs to be functional for a stable maintenance of long-term decay heat removal. Repair and replacement of the RHR heat exchanger and pumps in the event of a failure is not difficult since the components are located in the reactor auxiliary building outside the containment vessel, and are accessible.

There are different designs in the containment vessels of BWR reactors. Mark I type containment of Units 1 to 4 of TEPCO's Fukushima Dai-ichi are small in capacity compared against the power output. Capacity was enhanced on advanced Mark I type through modification of the drywell from a flasco-shape to egg-shape, as well as by lowering the location of the vessel to improve seismic resistance. The drywell and wet well (SC) was integrated for Mark II type and ABWR, enlarging and changing the containment structure and enhancing safety margin. In particular, with ABWR, the recirculation pump was mounted in the reactor vessel, eliminating likelihood of large break LOCA; the integrated structure of the containment and the reactor building which reinforced seismic resistance; separation and independence of emergency cooling systems; simplification and

enlargement of control panel boards for information sharing by the operating staff, and other safety improvements were made, enhancing the overall safety margin.

Since discharge of heat into seawater is hampered by SBO conditions for both BWRs and PWRs, “feed and bleed”, a method of cooling by venting vapor that is generated by feeding water into the reactor vessel should be recognized as the last resort and the sole success scenario for cooling.

Because transient response under beyond design basis conditions differ depending on the type and the design of each plant, the arrangement of additional safety SSCs should be made in accordance with specific conditions of each plant. In addition, the safety measures should not only focus on SSCs, but should include the organization of appropriate procedure manuals and provision of adequate education and trainings.

The filtered vent that was highlighted in the 3.11 incident is included in the new safety standards by the Nuclear Regulation Authority.

Hydrogen generated in the containment will not explode or combust in BWR reactors because of inert nitrogen atmosphere. However, due to the relatively small free volume of the containment, the filtered vent is provided to relieve excess pressure induced by the generation of large volume non-condensable gases as steam and hydrogen. Early venting prior to the onset of core damage was considered as an effective severe accident management measure in reducing risk of containment over-pressure, since the wet well water (suppression chamber) would filter and cleanse containment atmosphere, assumed to reduce Iodine and Cesium radioactivity to 1/100. Literature research conducted on BWR containment venting systems in 2000 showed that the US and Germany emphasized the significance of containment venting during severe incidents. Improvements were made on BWRs by adding reinforced pressure resistant vent lines with rupture disks to enable releasing large quantities of the steam generated in the event of delayed venting. Since reduction of both the reactor and containment pressure by venting enables coolant injection into the core, alternative injection via the fire extinguisher system was provided. Water injection into the reactor core is essential for preventing fuel damage, as well as controlling steam and hydrogen generation. Enhancement was also made on RPV pedestal injection.

It was unfortunate that although accident management measures related to SSCs have been developed, the inadequacies in the understanding on the design goals, procedures and trainings provided led to the severe accident. With view to these circumstances, the significance on the arrangement of filtered containment vent is not only as a final means against the onset of a severe accident, but in early venting – for reducing containment pressure to enable water injection and prevent core melting.

Due to the large free volume of PWR containment, excess hydrogen generated under severe

accident conditions will not induce containment pressure to exceed design pressure. Large quantities of steam is released with hydrogen generation during severe accidents. With PWRs, the generated steam will give rise to containment pressure loading, as well as hamper hydrogen combustion. However, for PWRs with ice condenser type containments that have a relatively small free volume, combustible gas control equipment (included from the first version of Safety Design Review Guidelines) is provided to prevent hydrogen explosion (accident management) in view of its characteristics of the reduction of partial pressure during containment cooling process, the gradual increase of hydrogen and oxygen concentration caused by radiolysis of water, which will eventually reach combustible domain, or explosion.

The most effective means for controlling containment over-pressure caused by excess steam released under accident conditions for PWR containments with large free volume is steam condensation. This is why alternative injection is included in the severe accident management of PWR reactors instead of the containment vent systems. Containment atmosphere recirculation systems that reduce pressure and condense steam by re-circulating and cooling the atmosphere instead of venting is effective for mitigating consequences to the environment as well.

As witnessed by TEPCO's Fukushima Dai-ichi accident, the enormous quantities of non-condensable gases as steam and hydrogen released by extensive fuel damage and core melt induced containment internal pressure to exceed design pressure which consequently led to containment rupture. An extensive containment rupture within a short time period will give rise to significant release of radioactive material (Iodine, Cesium, Strontium) combined with steam and non-combustible gases, adversely affecting the environment and the local public. The purpose of the filtered vent is in filtering or reducing radioactivity, and to vent containment atmosphere to prevent pressure overload and containment rupture. The filtered vent is the final means against containment damage (refer Section 5.3 "Considerations for Environment Pollution"),and should be provided in accordance with environment goals. A filter designed by combining chemicals and metal filter is shown as an example.

The adoption of the filtered vent is currently being examined as a back-up for various preventive measures. In utilizing filtered vents as a final means in accident management, necessity, effectiveness and influences on safety in early venting should be thoroughly examined with consideration given to the details of the examination process so far, and the variations in the type and the design of the containment. The key point is the establishment of a highly reliable, enhanced safety system for protecting the local public and the environment from harmful consequences of radiation under all circumstances.

8.4 Cases in Other Countries

1) Design Basis External Events

IAEA: Minimizing occurrence frequencies and likelihood of harmful consequences

The elimination of cliff edge effect in seismic resistance design

US: The severest incident extracted from historical record

Not considered for typhoon events of design wind speed below 10^{-7}/year

UK: Not considered for all events with occurrence frequencies below 10^{-7}/year

France: Order issued on the organization of hardened core requirements (June 2012)

Establishment of framework and structures to maintain integrity, in preventing and controlling progression of events involving core melts and extensive release of radioactive material caused by the onset of rare extreme natural phenomena and extended SBO.

2) Aircraft Crash Incidents (Accidental)

US: Reactor Site Criteria (10CFR Part 100) specifies design considerations for occurrence frequencies exceeding 10^{-7}/reactor-year.

However, after the 9.11 terrorist event, an order (ICM Order B.5.b, later, federal register) which requires licensees to adopt mitigation strategies using readily available resources to maintain or restore core cooling, containment and SFP cooling capabilities to cope with the loss of large areas of the facility due to large fires and explosions from any cause, was issued.

Requirements to large aircraft crashes for new plants.

Germany, Switzerland: In response to military craft incident in late 1970's, evaluated military plane incidents as design basis accident. Measures against aircraft crash include redundant arrangements and physical separation of safety SSCs as emergency power sources and RHR systems, which have been back-fitted to operating plants.

France, Finland: Consideration given to large aircraft crashes regarding new plants (EPR)

3) Aircraft Crash Incidents (Including Terrorist Events)

UK: Sizewell B NPP (start of construction in 1988) established provisions against aviation incidents including those by malevolent intent, such as enhancement of heat removal system (e.g., alternative air cooled ultimate heat removal components); arrangement of multiple containments; distributed arrangement of power supply equipments in separate buildings; establishment of the secondary control room, etc.

US: After 9.11 terrorist incident, issued an order (ICM Order B.5.b, later federal register) which requires licensees to adopt mitigation strategies using readily available resources to maintain or restore core cooling, containment and SFP cooling capabilities to cope with the loss of large areas of the facility due to large fires and explosions from any cause, including beyond design basis aircraft

impacts for new plants.

8.5 Effectiveness of Measures Established by NISA for Events Other than Tsunami Events and Future Issues

As shown in Section 8.1, the 30 measures established by NISA may be applied in preventing and mitigating severe accidents induced by natural phenomena other than the tsunami and by human events shown in Section 7. For example, measures against terrorism included in the new requirements by the Nuclear Regulation Authority are in equivalence to the terrorist measures determined by US. Immediately after 9.11, 2001 incident, the US Nuclear Regulatory Commission (NRC) conducted assessment using state-of-the-art structural analysis and fire analysis on nuclear incidents caused by a terrorist attack on large passenger airliners involving physical impacts as large fires and radioactive material release to the environment, and confirmed small possibilities of reactor damage and subsequent release of radioactivity. Further, NRC issued ICM Order B.5.b for mitigation of terrorist threats, requiring licensees to adopt mitigation strategies using readily available resources to maintain or restore core cooling, containment and SFP cooling capabilities to cope with the loss of large areas of the facility due to large fires and explosions from any cause, including beyond design basis aircraft impacts.

The B.5.b of ICM Orders correspond to the 30 measures established by NISA – “to adopt flexible and practical mitigation strategies using readily available resources as portable power supply, water make-up and heat exchanger systems and trained personnel, to maintain or restore cooling capabilities”. Around 2003, NRC made confidential notification on the orders (confidential due to being a terrorist measure) to member countries of the nuclear community including Japan. Finally after Fukushima Dai-ichi accident, Japan has established emergency preparedness and response measures, including those relevant to B.5.b of ICM Orders, against events with likelihood of inducing fires, explosions, functional failures, etc.

NISA at the time did not see the notification by US NRC on B.5.b as an opportunity in breaking away from the long-pitched “absolute safety myth” that the government and the nuclear community in Japan had been seeped in and in comprehensively reducing risk involving nuclear plants. Had NISA, for example, issued orders to all nuclear plants in Japan to implement portable safety components that are inexpensive and may flexibly deal with anticipated events and event progression, in mitigating risk caused by external events as tsunami, fires, explosions, etc., on the basis of 2004 Off the Coast Sumatra Earthquake, killing over 220,000 people and 2007 Chuetsu-oki Earthquake involving extended fire caused by transformers, the outcome of TEPCO’s Fukushima Dai-ichi accident would have taken a completely different turn. In contrast, the earthquake-prone

Taiwan having experienced M.7.6 earthquake in 1999 was receptive to the US notification and established comprehensive risk mitigation strategies including the arrangement of large capacity water supply system. The Nuclear Regulation Authority will include measures against tsunami events and terrorist threats equivalent to those specified in B.5.b.

Japan with high exposure to natural disasters should establish a framework of measures against not only earthquake and tsunami, but aircraft crash, tornados, meteorite events, large-scale volcanic eruptions, acts of sabotage by terrorism, etc.

The 30 measures on severe accident management are requirements related to the hardware, or tangible aspects, such as key safety SSCs.

Even after TEPCO's Fukushima Dai-ichi NPP accident, the key issue on radiation risk has not been highlighted in Japan, while discussions continue over the tangible aspects of safety measures that are easily understood by the public in general. These circumstances after Fukushima accident mirror Japan's stance that has not changed in emphasizing "low unplanned shutdown frequency compared to those of other nuclear states" as criteria for safety performance of nuclear plants described in Section 5.1. Section 4 to 6 describes the extreme importance of both the government and the nuclear community to recognize responsibilities commensurate with the assigned roles for achieving safety enhancements of both tangible and intangible aspects in nuclear power generation. The fundamentals of severe accident management is "management" (intangible aspects), as shown by the experiences of TEPCO's Fukushima Dai-ichi accident and as pointed out in various reports and analyses on the accident. Regulating "management" and the operators' efforts in incorporating "management" in plant operations are crucial issues that must be addressed The operating staff must ensure safety in daily plant operations, and in the event of an accident, the operating staff and the director responsible for site supervision must deal with the circumstances. From nuclear regulatory and policy perspectives, a system for maintaining and enhancing competencies of the site personnel in handling accident conditions should be established. Further, a transparent and effective liaison between related organizations and professional fields beyond the domains of science and engineering should be established for ensuring nuclear safety, including cooperation by the Self-Defense Force for disaster preparedness and response.

Footnote5: Filtered vents

In the event of severe accident involving extensive release of core damage, measures must be taken to maintain pressure and temperature of the containment atmosphere within design values, where containment vent systems and atmosphere recirculation systems may be effective means.

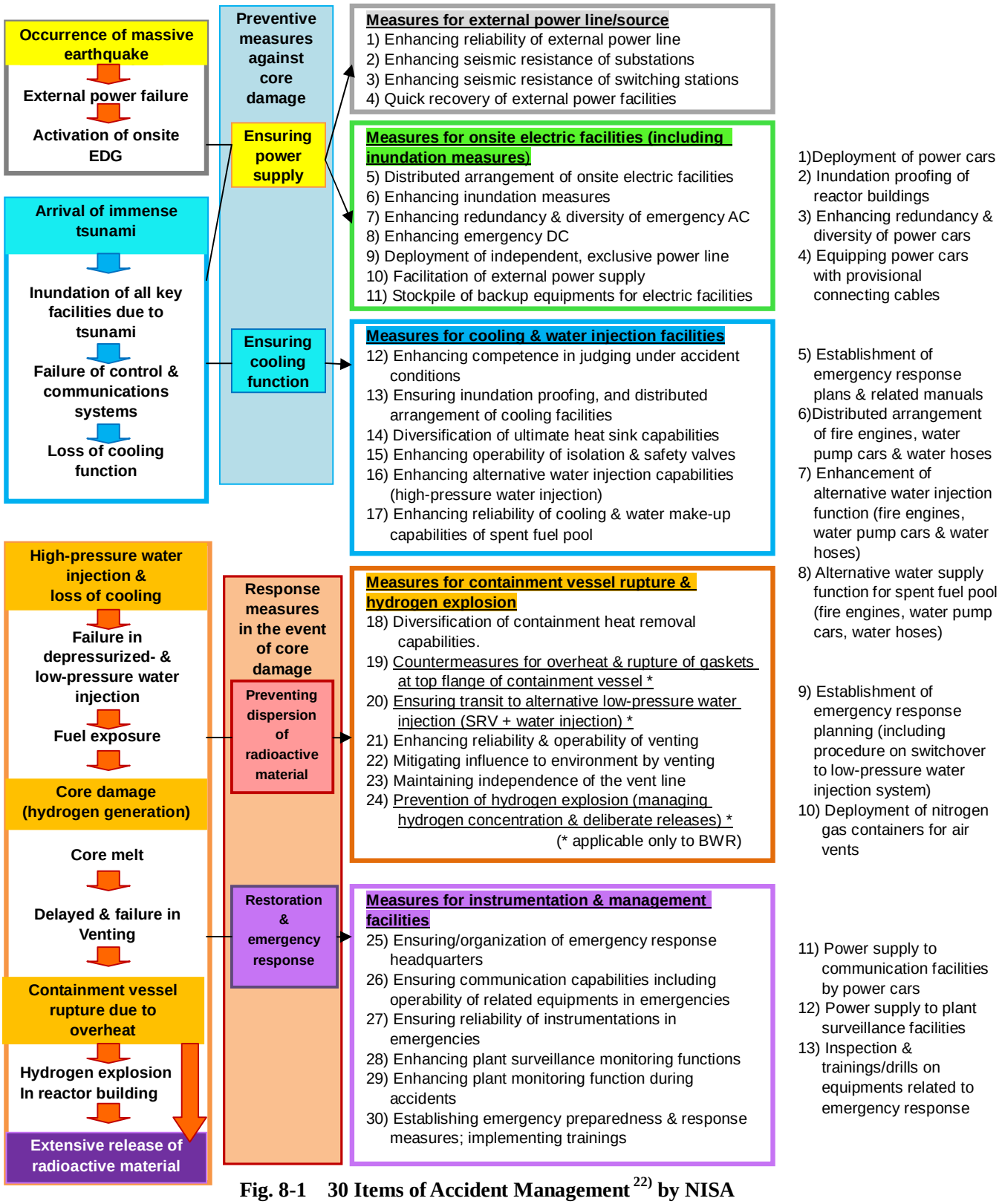
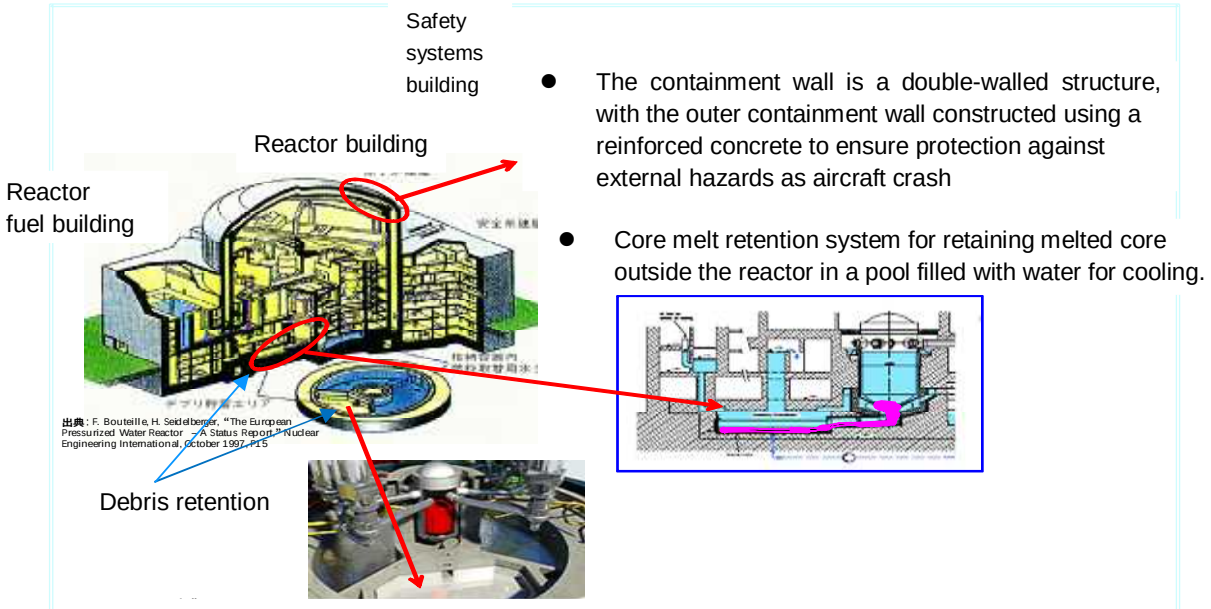


Fig. 8-1 30 Items of Accident Management [22)] by NISA

9. SPECIFICATIONS AND MANAGEMENT OF NEXT GENERATION REACTORS

During the period construction of many of the currently operating plants was being promoted, “the concept of next generation reactors” was proposed.

European countries, particularly, Germany and France promptly addressed the issue in the wake of the Three-Mile Island accident (PWR). No one in the nuclear community at the time anticipated the occurrence of a large-scale core melt incident, and watched event sequences in dismay, wondering whether a melt-through of the pressure vessel and extensive release of radioactive material through the ruptured containment will take place, or not. The result was outflow of the molten fuel to the base of the core and partially into the reactor structure and lower core structure, but the fuel did not come in direct contact with the reactor containment vessel. However, the molten fuel assembly components and structural parts of the reactor (corium debris) penetrated through the lower core grid, damaging instrumentation tube located in the lower plenum, further corroding and rupturing stainless panel at the base of the pressure vessel. Fortunately, melt-through of the pressure vessel did not occur. However, there were minor releases of radioactive iodine, and evacuation recommendations for pregnant women and children was issued, which eventually led to the evacuation of the local residents under a great deal of confusion caused by poor emergency communications.



Reference: http://www.areva.com/EN/operations-1740/epr-reactor-safety-as-a-priority.html

Reference: F. Bouteille, H. Seidelberger, “The European Pressurized Water Reactor – A Status Report”, Nuclear Engineering International, Oct. 1997, P.15

Fig. 9-1 Conceptual Diagram of EPR

Figure 9-1 illustrates EPR, or European Pressurized Reactor, an “evacuation free” third-generation PWR reactor, designed and developed jointly by Framatome (currently, Areva) of France and Siemens (the nuclear business division was later merged into Framatome) of Germany. The EPR was designed with enhanced core cooling capabilities to withstand beyond design basis events, to retain and cool corium debris in the containment in the event of a melt through and retain radioactive material in the containment. The containment wall is in two layers for protection against aircraft crashes. EPR is currently under construction in Finland, France and China.

Whereas, there was about a 30-year blank period in the construction of light-water reactors due to the Three-Mile Island accident in the US. During the period, US focused on enhancing operating rate and power output rate of existing plants, and efforts were made on the development of natural circulation cooling system like in AP-1000. Enhancement on AP-1000 was made in response to 9.11 terrorist attack, with a new design on protection against aircraft impacts. The AP-1000 is shown in Fig. 9-2. The cost on the construction of AP-1000 was reduced because of the smaller number of components required in the new design. It is currently under construction in the US and China.

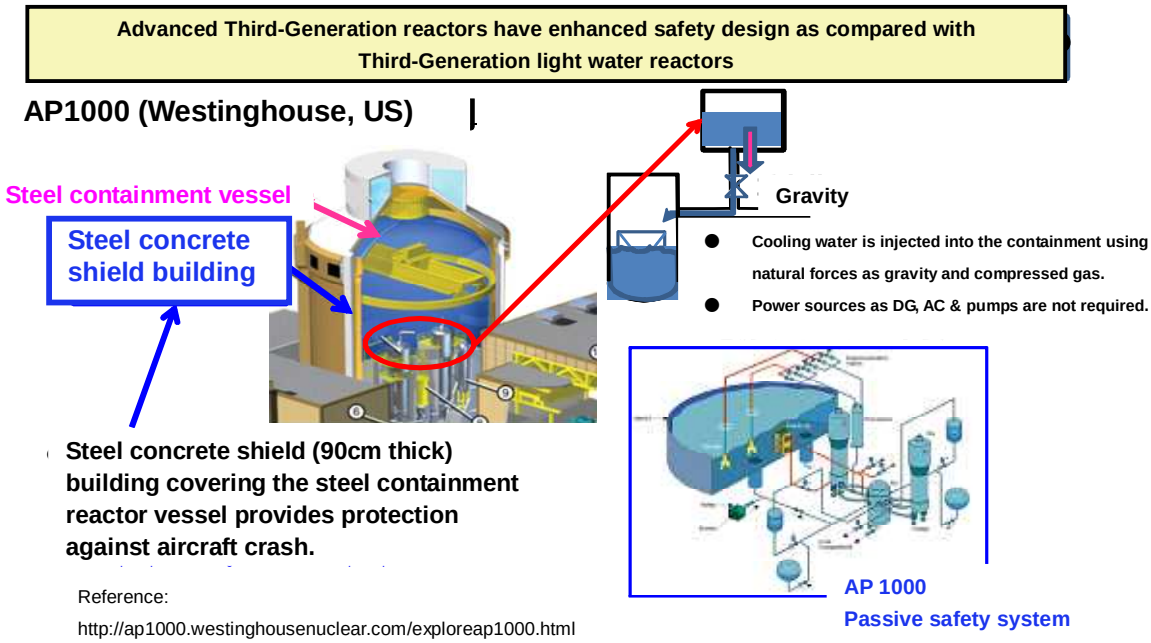


Fig. 9-2 Conceptual Diagram of AP-1000 [6]

The introduction of next-generation reactors in Japan has been delayed with its development program initiated in 2008 under the collaboration by the government and nuclear industries targeting introduction in 2030’s, with the termination of 60 years service life of light-water reactors that began operations in early 1970’s.

The development of next-generation reactors has been initiated as an international project, in light of the increasing demands – demand for replacements not only in Japan, but also in Europe and US, and demand for new constructions in countries in Asia and the Middle East that are newly introducing nuclear plants. The purpose of the project was also in maintaining and enhancing technical and personnel aspects in the nuclear technology field. The development targets for the next-generation light water reactor in Japan is shown in Table 9-1[26].

Item	Key Requirements
1.Basic Conditions	Power output: 1,700,000 to 1,800,000 kW Utilization of common technical expertise, applicable to 800,000 to 1,000,000kW without comprising effectiveness of standardization.
2.Safety	CDF and CFF that meet global standards Design consideration on severe accident management measures
3.Economic viability	Construction cost (matured reactors): JPY0.13M/kW Construction period: Less than 30 Mo (from bedrock inspection to start of operation) with preconditions on adherence to planned construction period. Operating rate/hr: 97% (average life cycle), 24-months operating cycle Design service life: 80 years Power generating cost is competitive against other power sources.
4.Societal acceptance	Design must sufficiently reduce extensive release of radioactive material to the environment Sufficient margin over residual risk involving earthquake and tsunami. Applicable to provisions against aircraft impacts and other security measures by European countries and US. Workers' dose level: sufficiently below current standards
5.Management, Operation, maintenance	Quantity of maintenance resources: 50% of the newest plant volume Maintenance enhancement and evening-out of maintenance load Core design: average fuel burn-up of 70GWd/t, applicable to complete fuel loading of MOX. New technologies to be applied should be sufficiently matured.
6.International standards	Applicable to licensing requirements and regulatory standards in US and Europe. Standardized design not reliant on siting conditions.

Table 9-1 Development Targets for Next-Generation Light Water Reactors

Design safety goal includes considerations on severe accident management as with standard next-generation reactors in Europe and US. Further, enhanced seismic resistance components for broader application of standard design regardless of siting conditions has been developed and introduced. Conceptual diagram of the next-generation reactor is shown in Fig. 9-3.

Next-generation reactors have the following design features: 1) enhanced core cooling capabilities against severe accidents on the basis of extensive scenarios; 2) integrity of cooling capabilities in the event of a core melt and corium debris falling on the containment, against external release of radioactivity (no residents evacuation required); 3) structural integrity of the containment vessel and reactor building against aircraft impacts; 4) design specifications not reliant on seismic conditions of the construction point, and the arrangement of enhanced seismic resistance components (Japan), etc.

However, in view of the broad spectrum of extreme natural disasters as earthquakes, tsunamis, flooding, landslides, volcano eruptions, typhoons, tornados, etc., that Japan has experienced in the past, and significant vulnerabilities in Japan's geo-political circumstances against nuclear terrorism and aircraft crashes (NPPs as military targets), efforts should be extended to pursuing safety enhancement that surpass the global standards of safety.

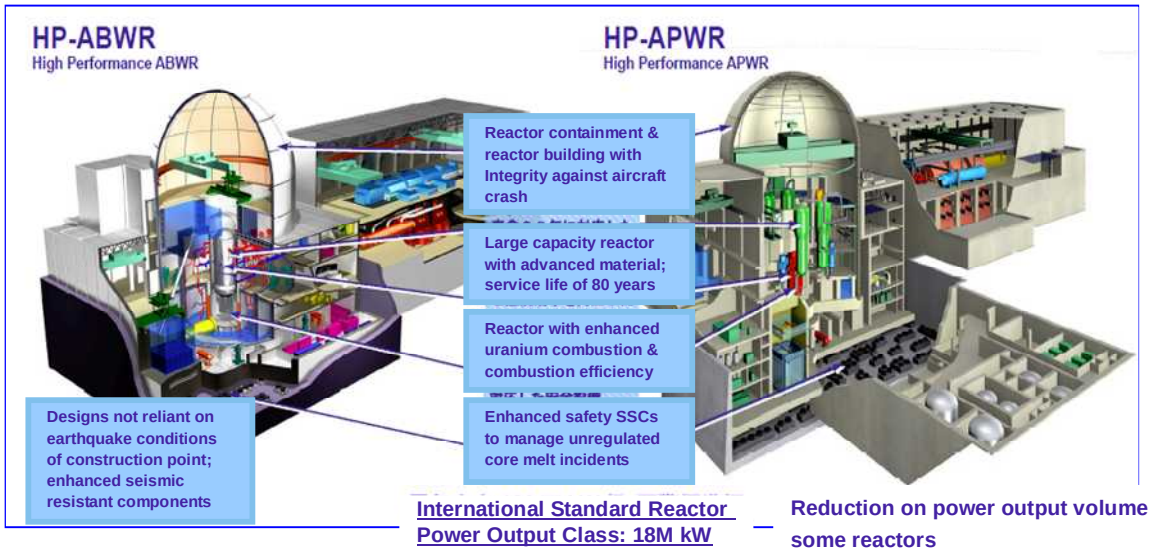


Fig. 9-3 Conceptual Diagram of Next-Generation Reactors in Japan

As well, under the defense-in-depth concept described in Section 4 (the preconditions for multiple defense barriers), plant, building and component configurations should be flexibly designed and reasonably applicable to site-specific safety goals, with consideration given to the occurrence probability of beyond design basis conditions. In addition to measures against core melt conditions the next–generation reactor should include the following design - maintenance of reactor core integrity for some period without operator actions in the event of loss of all power sources; extended containment capabilities that may control radioactivity release in the event of extensive core damage. Public consensus and acceptance should be established through these enhancements, as well as realizing a cost effective, economically viable next-generation reactors.

Footnote6: Next Generation Reactors

Next generation nuclear plants (reactors) was defined by the US Department of Energy (DOE). Prototype and power reactors established between 1950's to 1960's is defined as Generation I reactors, followed by Generation II commercial pressurized water and boiling water reactors, CANDU (CANada Deuterium Uranium) reactors, etc., established between 1960's to 1990's with design operating lifetime of 30 to 40 years, most of which are in operation. Design improvements

were made on Generation II reactors after the Three-Mile Island incident, as enhanced safety systems, thermal efficiency, extension of operational life period, cost reduction, etc., which is the Generation III reactors. The majority of reactors established in 1990's are Generation III reactors. Significant improvements on safety such as passive safety features of emergency core cooling and containment cooling systems, and for preventing LOCA events, were made over Generation III, which are Generation III + reactors. The designs include AP-1000, EPR (European Pressurized Reactor), ABWR (Advanced Boiling Water Reactor), APWR (Advanced Pressurized Water Reactor) reactors. A joint research development on 6 new type of reactors including FBR has been initiated by Japan, US and Europe targeting commercial operation in 2030's, which are called Generation IV reactors.

10. SUMMARY AND RECOMMENDATIONS

First, the fundamental concept on nuclear safety should be determined by referring to “Fundamental Concept on Nuclear Safety - Part I: Nuclear Safety Objectives and Fundamental Safety Principles” established by the Atomic Energy Society of Japan, to be shared by all parties in the nuclear community (including local governments), and to fulfill responsibility commensurate with the assigned roles.

Secondly, on the basis of understanding of the defense-in-depth concept, emphasis should be placed on defense-in-depth design that focuses on functional aspects for establishing a framework on ensuring safety through the integrity of the systems, and its application to operating plants. For beyond design basis conditions, accident management framework should be established to deal with the various scenarios on accident sequences with a continuous process of applying state-of-the-art-technologies to ensure accident management.

Thirdly, incorporating resilience engineering concept essential for post-accident management and creating scenarios on the recovery process for which various procedures may be developed. To ensure the effectiveness of the procedures expected to be complex, digitalization of the procedures, along with a documented procedures manual should be developed.

Fourthly, various measures related to human resources development should be considered, such as fostering of safety culture, personnel exchange and enhancement of qualification systems, etc. In view of the vulnerabilities shown in the accident management at Fukushima, a person responsible for plant operations, Chief Engineer of Reactors should be assigned to each plant. At the same time, to enhance quality of all operating staff, a qualification program with requirements corresponding to the job level in terms of skills and knowledge, responsibilities, work conditions including compensation, should be established.

Japan has been recognized as the global leader with technical competencies in the production of SSCs, design and construction of nuclear facilities, not to say the least of the pre-eminence in parts manufacturing. Of the few nuclear plant manufacturers in the world, three are Japanese manufacturers. It is no exaggeration to say that the Japanese manufacturing process management governs nuclear plant construction worldwide. Nuclear industry in Japan has made progress to be ranked as number one globally in terms of reliability of its key components such as the pressure vessel. However, the regulatory aspects of nuclear safety in Japan has been stagnant. From the moment government decision was made on the introduction of nuclear power generation into Japan for energy security, safety control, or ensuring safety against nuclear risk was the first issue that should have been addressed by all parties including the authorizing government, the regulatory body, related organizations and corporations and the public. Instead, regulatory control on nuclear safety

was used for political negotiations and trade-offs between the government and the industries. Consequently, this led to the deviation from global standards on “nuclear safety”, including the establishment of public consensus and communications. The root cause for Fukushima Dai-ichi accident was the the total lack of efforts and a sincere attitude in ensuring “nuclear safety” by all those involved, by all people of Japan.

On such basis, the first step that should be taken is for the engineers and experts engaged in nuclear power generation to reflect and to make commitment to the new endeavors in ensuring “nuclear safety”.

Now is the time that the lessons learned from the Fukushima accident is reflected on, and incorporated into post-accident management, disaster recovery and clean-up by liaising with the international community, as well as in establishing “nuclear safety” in achieving Japan’s goal on the highest standards of safety in nuclear power generation.

Of the nuclear plants in Asia, including those in planning phase, 28 plants have been established in South Korea, 27 in China (the number of planned construction is 100 to 200), 31 in India, and is expected to grow to more than 500 plants worldwide.

TEPCO’s Fukushima Dai-ichi accident is not only the experiences of Japan. Japan, by reflecting on, and making the best of the experiences is creating the pathway to ensuring and establishing “nuclear safety”, sought by all parties worldwide. Japan has a commitment to sharing, and reflecting on these experiences with the global community.

In view of the dire consequences of TEPCO’s Fukushima Dai-ichi accident, measures for defense-in-depth level 4 or beyond design basis events in the severe accident spectrum should be established and applied for the operation of existing plants. To this end, continuous enhancement of measures against extreme natural disasters as large-scale earthquakes and tsunami as well as measures for level 4 events induced by other causes with consideration given to site-specific conditions as design and siting, should be established appropriately and in good sequence.

Regardless of whatever measures taken, no measure for any activity in any industry including those for nuclear power generation will guarantee 100% safety. All activities involve risk and measures are taken to minimize such risk to the extent possible. This must be communicated to the public together with the benefits gained by nuclear power generation.

Followings are the recommendations with commentaries.

RECOMMENDATIONS

Recommendation 1

Anticipating 'unforeseen' natural disasters or human events associated with nuclear incident is imperative. A fundamental approach in anticipating the 'unforeseen' (events) is critical for ensuring nuclear safety, and should be developed.

(Commentary)

"Unforeseen" is unacceptable in ensuring safety of nuclear power facilities. The regulatory body and licensees should establish a framework on emergency preparedness and response on all credible natural disasters, human-induced and internal events, etc. The best approach in eradicating the "unforeseen" is to thoroughly examine and identify as many severe accident scenarios as possible from a broad spectrum of events, develop relevant measures, and provide drills and trainings on these measures.

Recommendation 2

A framework for ensuring nuclear safety should be established, whereby, safety review guidelines and standards on safety should be reevaluated without being subject to preconceptions for developing a globally established framework on nuclear safety.

(Commentary)

By referring to the IAEA Safety Standards, "Fundamental Concept on Nuclear Safety" for ensuring safety of nuclear power plants that is tailored to Japan circumstances should be established. On the basis of the concept, a framework that embodies safety objectives, performance objectives and fundamental policy on safety regulations should be developed in the early stage.

A thorough review and modifications should be made on the Safety Design Review Guidelines on the basis of the lessons learned from the Fukushima accident. Safety assessment method associated with severe accident management in the spectrum of defense-in-depth level 4 should be newly developed and incorporated into the Safety Evaluation Review Guidelines.

Recommendation 3

All related parties in the nuclear community must recognize responsibilities commensurate with assigned roles, and establish the top priority in ensuring safety. The regulatory body, in particular, must determine fundamental principles for the prevention of, and mitigation of consequences of severe accidents by hearing the opinions of a broad spectrum of experts. The licensees must determine severe accident measures and effectively implement them with a sense of vigilance.

(Commentary)

In the event of a severe accident, the local public in the vicinity of the site boundary and the environment must be protected from harmful effect of radiation. This must be kept in the minds of all those involved in the design, construction, operation of nuclear power plants and associated facilities. A continuous process of incorporating state-of-the-art technologies and results of safety researches should be maintained under the proactive support of nuclear experts.

Recommendation 4

The State and the licensees should independently and/or jointly – along with scientists and experts in nuclear technology field through professional societies - establish risk communication on nuclear power generation with the public as well as promote activities in establishing public consensus on the benefit and risk of nuclear power generation.

(Commentary)

The government and the licensees are responsible for the continuous process in building consensus and gaining public confidence on the benefits and risks of nuclear power generation. Scientists and experts in nuclear technology field must also establish and maintain dialogue with the public on the benefits of nuclear power generation, which do not necessarily guarantee absolute safety and should be balanced against the risks.

Recommendation 5

The regulatory body shall regulate plans and inspections on severe accident prevention and mitigation measures proposed and prepared by the operators. In the examination of measures, all internal events (including human error events, etc.), natural phenomena and human-induced events associated with severe accident should be included. By cooperating with experts and operators, the regulatory body should develop effective accident management by combining measures, including the use of a variety of components and equipments for preventing and mitigating severe accidents.

(Commentary)

The regulatory body shall conduct inspection and surveillance on severe accident management without omission to ensure that the combination of existing and new components and equipments for preventing and mitigating consequences of severe accidents fulfill requirements. Regular inspection on these facilities should be maintained.

Recommendation 6

Reliability of safety functions for preventing and mitigating severe accidents shall be ensured through elimination of common cause failures, by ensuring independent effectiveness through distributed arrangement and diversification of safety functions.

(Commentary)

For the elimination of the likelihood of common cause failures of key safety components and equipments, diversified and distributed arrangement of these components and equipments with different operating principles should be provided.

Recommendation 7

Specific measures for accident management should be flexible as to address unanticipated conditions which may not be dealt with by permanent facilities. Thus, transportable and mobile facilities (fixed on vehicles), and redundant connections should be provided for flexibly coping with all circumstances.

(Commentary)

For preventing and mitigating severe accidents, effective accident management should be developed by combining measures, including use of a variety of components and equipments. In general, by identifying scenarios leading to severe accidents, design and arrangement of permanent facilities for preventing severe accidents could be implemented. However, in the event of an unanticipated event, or the unexpected sequence of events leading to a severe accident, the provision of transportable and mobile facilities is extremely effective, and recommended. It goes without saying that the effectiveness of these measures should be confirmed and ensured in advance.

Recommendation 8

Operators must assign onsite accident management specialist(s) with a thorough understanding of the nuclear power generation system, having competence to understand or assume likely circumstances of the nuclear reactor under accident conditions, and the ability to make appropriate judgment in providing necessary directions to onsite staff.

(Commentary)

The accident management specialist must possess professional expertise and competence in accident management, supervise education and training on accident management, and advise the site director on matters such as installment of necessary facilities and allocation of staff required for implementing accident management. In emergencies, the accident management specialist must support the site director on the deliberations, decision-making and authorizing accident management operations.

Recommendation 9

Operators shall prepare an accident management procedure manual by confirming each item of the manual at the site, on the basis of which education, drills and exercises under all credible conditions shall be fully provided to the staff.

(Commentary)

Staff involved in a nuclear power plant should not only understand the basics of nuclear power generation, particularly, reactor physics, reactor behavior under accident conditions, nuclear safety, but also have a thorough knowledge of overall plant characteristics. Education and training on severe accident measures on the basis of a broad range of scenario sequences leading to severe accident should be provided.

Recommendation 10

The regulatory body shall conduct inspection and surveillance on accident management without omission. Operators and the regulatory body should independently, or in cooperation, carry out reassessment for continuous enhancement of accident management.

(Commentary)

The regulatory body shall regularly require operators to submit 'Severe Accident Prevention Plans', that contains reporting on credible accident scenarios on natural disasters, human events and internal events; response management plan; and implementation status of emergency response training, etc., for each plant. Subsequently, the report shall be reviewed and evaluated for approval by the regulatory body. Notwithstanding the regulatory requirements, the operators should be alert for any events with likelihood of severe consequences, and develop preparedness and response measures against these events.

In addition, the regulatory body is recommended to convene an annual meeting for the briefings on activities by accident management specialists, which includes a session comprising of the regulatory body, operators, manufacturers, and experts in the nuclear field to provide advice to the accident management specialists and for sharing good examples and role model cases.

11. EPILOGUE

It was Hiroyuki Abe, who showed strong concern over the various issues presented in TEPCO's Fukushima Dai-ichi accident, and over similar issues related to the safety of other nuclear plants.

In view of the unforeseen event of Tohoku Region Off The Pacific Coast Earthquake, the Headquarters for Earthquake Research Promotion (under the Ministry of Education, Culture, Sports, Science and Technology) developed and presented the worst case earthquake and tsunami scenario likely to occur. The scenario was modeled on simulation of simultaneous and consecutive earthquake in the region of the Tokai, To-Nankai, and the Nankai Troughs with a magnitude of M.9 showing corresponding tsunami heights under the scenario in each areas of Japan. Nuclear plant operators in Japan have initiated the reevaluation of design seismic motion and design tsunami height on the basis of worst case scenarios, and to re-examine and develop additional safety measures.

Why TEPCO's Fukushima Dai-ichi accident was not preventable, and what measures should have been in place have been addressed in Section 2 and 3, with details on preparedness for severe accidents, judgment and actions required in the course of events, fundamental issues of accident management. The three fundamental principles, "shutting down", "cooling", and "containment" in the accident management of up to defense-in-depth level 3 are applicable to severe accident prevention. The shutdown of both Kashiwazaki Kariya NPP during Chuetsu-oki Earthquake and that of Fukushima Dai-ichi NPP has been successful. The initiator of Fukushima Dai-ichi NPP accident was the loss of "cooling". Sequence of events following the coolant failure, and measures required for preventing and mitigating severe accident consequences based on fundamental nuclear safety principles are outlined in Section 4 to Section 8. In Section 9, the overview on the next-generation reactors that take into account severe accident management have been presented.

As conclusion, recommendations on preventing recurrence of the disaster have been formulated, with emphasis on the significance of not only hardware (e.g., SSCs), or tangible aspects but intangible aspects as commitment to safety by all individuals involved in the operation of nuclear plants.

In the recommendations, the importance of all organizations and individuals, including the regulatory body, operators, manufacturers, etc., involved in the operation of nuclear facilities and activities to strive to make the best achievable efforts to ensure safety and to minimize risk commensurate with assigned roles and responsibilities, based on the principles of nuclear safety culture has been clearly shown. Subsequently, dialogue should be established with the public on the benefits and the risks of nuclear power generation to reach on a consensus over the use of nuclear power.

Due to shortage of time and limited number of experts, some parts of the report may require further information and details. However, it is with hope that the document will provide the basis on

“nuclear safety” for the future operation of nuclear plants for all parties engaged in the activities of nuclear facilities.

ACKNOWLEDGEMENT

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The nuclear community must invest full efforts so that the recommendations may be utilized to gain broader understanding of the public on nuclear safety and the use of nuclear power. We look forward to the continued support.

REFERENCES

1) "The Official Report of The National Diet of Japan Fukushima Nuclear Accident Independent Investigation Commission", The National Diet of Japan Fukushima Nuclear Accident Independent Investigation Commission July 5, 2012
2) "Report on the Accident at Fukushima Nuclear Power Stations of Tokyo Electric Power Company", Investigation Committee on the Accident at Fukushima Nuclear Power Stations of Tokyo Electric Power Company, July 23, 2012
3) "Research Investigation Report by The Independent Investigation Commission on the Fukushima Nuclear Accident", Rebuild Japan Initiative Foundation (private investigatory committee), February 28, 2012
4) "Fukushima Nuclear Accident Analysis Report", Fukushima Nuclear Accident Investigation Committee, Tokyo Electric Power Company, June 20, 2012
5) "Fukushima Dai-ichi NPP: Accidents and Disasters Caused by the Pacific Coast Tsunami of Tohoku Earthquake: Lessons from Evaluation of Fukushima Dai-ichi NPP Accidents", *Atmos*, Atomic Society of Japan journal, P. 1 – 14, No. 6, Vol. 53, June 2011
6) "Accident Management for Severe Accidents at Light Water Power Reactor Installations", The Nuclear Safety Commission, May 28, 1992
7) References prepared by Agency for Natural Resources & Energy, MITI, Japan
8) "Report on Development of Accident Management for Fukushima Dai-ichi NPS", Tokyo Electric Power Company, May 2002
9) "Technical Findings on Accident at TEPCO's Fukushima Daiichi NPP (ad interim report)", NISA, February, 2012
10) "Additional Report of the Japanese Government to the IAEA – The Accident at TEPCO's Fukushima Nuclear Power Stations – (Second Report)", Nuclear Emergency Response Headquarters, Government of Japan, September, 2011
11) *Atmos*, Atomic Society of Japan journal, "Probabilistic Risk Assessment for Nuclear Facilities and Future Trends", P. 45 - 50, No. 1, P. 52 -56, No. 2, P. 36 - 42, No.3, Vol. 54, January, February, March 2012
12) *Atmos*, Atomic Society of Japan journal, P. 23- 27
13) "Reassessment of Fukushima Nuclear Accident and Outline of Nuclear Safety Reform Plan (Interim Report)", Nuclear Reform Special Task Force, Tokyo Electric Power Company, December 14, 2012
14) "AESJ-SC-TR005, "Fundamental Concept on Nuclear Safety – Nuclear Safety Objectives and Fundamental Safety Principles", Technical Report by Standards Committee, Atomic Energy Society of Japan, April 2012

15) “Concept on Seismic Safety in the Design and Evaluation of Nuclear Power Plants”, July 2010, Special Advisory Committee on Nuclear Power Plant Earthquake Safety, Atomic Energy Society of Japan
16) “2012 Report on Japan Ageing Management Program on System Safety”, Nuclear Regulation Authority, 2012
17) “Issues and Recommendations on Nuclear Power Generation: Is Service Limit on Nuclear Power Plants Really 40 Years?” *Energy Review,* November, 2012
18) “Reassessment of Fukushima Nuclear Accident and Outline of Nuclear Safety Reform Plan”, Tokyo Electric Power Company, March 2013
19) “Interim Report on the Investigation and Review on Safety Goals", Special Committee on Safety Goals, Nuclear Safety Commission, December, 2003
20) “Guideline on Severe Accident Consideration in Containment Design of Next-Generation Reactors” Special Investigation Committee on Containment Vessel Design, Nuclear Safety Research Association, April 1999
21) “Issues and Recommendations on Nuclear Power Generation: Seismic Resistance and Tsunami Resistance” *Energy Review,* December, 2012,
22) “Lessons Learned from Fukushima Daiichi Nuclear Power Plant Accident and 30 Measures for Accident Management”, Tadashi Narabayashi, Global Energy Policy Resarch, November, 19, 2012
23) “Results of Assessment on Effects of Tsunami on Hamaoka Nuclear Power Plant Using Tsunami Fault Model of the Cabinet Office, Japanese Government”, Chubu Electric Company, December 2012
24) “Anti-Tsunami Construction at Hamaoka Nuclear Power Plant”, Japan Society of Maintenology, P. 24 -29, No. 4, Vol. 11, 2013
25) "Implementation Status of Emergency Preparedness Measures (Revised; Ohi NPP)”, Kansai Electric Power Company, April 2011
26) “Interim Report on Technology Development of Next-Generation Light-Water Reactors”, Institute of Applied Energy, July 29, 2010

GLOSSARY

B.5.b

Immediately after 9.11 incident, ICM order was issued by US Nuclear Regulatory Commission requiring licensees in US to adopt mitigation strategies against terrorism, etc. B.5.b is a section of the order that addresses damage from fire or explosion such as could occur from impact of large commercial aircraft. B.5.b also requires provision of safeguards (e.g., portable power supply components) and trainings for SBO.

Cs-137

Radioactive cesium isotope with mass number 137 and a half-life of 30.1 years, is generated through nuclear fission of Uranium235. Whereas Cs-134, the radioactive cesium isotope with mass number 134 has a half-life of 2.1 years.

TMI Accident

A severe nuclear accident that occurred in Three-Mile Island Nuclear Power Plant in Pennsvlvania, in the northeastern part of US on March 28, 1979. The loss of core cooling led to core melting, with minor release of radioactivity in the atmosphere. The accident was evaluated as a Level 5 incident under the INES (International Nuclear Event Scale) standard.

Accident Management

Actions taken during the evolution of beyond design basis events to prevent escalation to severe accidents, and to mitigate influences of severe accidents by effectively utilizing SSCs arranged as part of accident management combined with the safety functions incorporated in the safety design and safety margin and other function that may be expected to prevent or mitigate accident conditions.

Safety Culture

The assembly of characteristics and attitudes in organizations and individuals which establishes that, “as an overriding priority, protection and safety issues receive the attention warranted by their significance.”

Event Tree

A graphical presentation of the alternative outcomes, or the success and failure of each safety measures in compensating failure of the systems.

Emergency Response Center

A facility to be used as a base for nuclear emergency response. In the wake of the 1999 JCO criticality accident at Tokai-mura, Ibaraki Prefecture, the Act on Special Measures Concerning Nuclear Emergency Preparedness was set forth which included the establishment of emergency response center as a base for the related parties including the government, local governments, the operators, nuclear experts, to effectively carry out emergency response for ensuring safety of the

local public.

Probabilistic Safety Assessment

A comprehensive quantitative assessment on the occurrence frequency and influences of credible accidents and failures in nuclear facilities.

Probabilistic Hazard

The occurrence frequency of a hazard presented in accordance with the magnitude of a hazard, or the magnitude of a hazard presented in accordance with the occurrence frequency. With regard to all initiating events in general, the magnitude of an event is proportional to its occurrence frequency; i.e., the greater its magnitude, the smaller the frequency of occurring.

Severe Accident

Previously called "severe accident" by former regulatory regime, it has been renamed, "*judai-jiko*" (serious accident) in the Nuclear Regulation Authority Establishment Act. This report uses the term, "severe accident" for ease of understanding. Severe accident has been defined as "the conditions exceeding design basis where measures developed on the basis of design safety criteria will not effectively maintain core cooling capabilities or control reactivity, resulting in extensive core damage, as core melting". Severe accident conditions includes extensive release of radioactive material due to significant loss of containment isolation capabilities. Design basis event refers to accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

Hypothetical Accident

Under "Regulatory Guide for Reviewing Nuclear Reactor Site Evaluation and. Application Criteria", the basis in the safety assessment preceding construction of nuclear power plants for evaluating the adequacies of siting conditions, "hypothetical accident" is defined as an accident exceeding the conditions of a serious accident, improbable from technical perspectives. In the event of a hypothetical accident, the guideline stipulates controlling radiological consequences to the public in the vicinity. The assessment criteria has been set forth on the ratio of radioactivity release to the containment against quantity of radioactive material contained in the reactor, for noble gases at 100% and for Iodine at 50%.

Gal

Derived unit of acceleration, defined in terms of CGS (centimeter-gram-second); 1 Gal is defined as the acceleration of 1 centimeter per second squared (cm/s).

Design Basis Seismic Motion

Seismic motion considered in the design safety of nuclear facilities.

Performance-based Criteria

Establishment of performance-based technical standards by the regulatory body; or technical

standards (codes & regulations) focusing on performance-based criteria, with flexibility in the selection of specifications for achieving required performance.

Cliff Edge Effect

Abnormal plant behavior caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter.

Preclusion of Succeeding Defense Levels

Defense-in-depth concept in precluding the succeeding level of defense barrier

Consensus

General agreement or accord; an opinion or position reached by a group as a whole.

Sabotage

Deliberate action directed at destruction, subversion, obstruction, etc., of nuclear facilities.

Residual Risk

The risk, or the likelihood of beyond design basis seismic motion affecting nuclear facilities, inducing severe damages and release of extensive radioactivity from the facilities, resulting in radiological consequences to the public in the vicinity. Residual risk must be minimized.

Sequence

A systematic and logical modeling of accident progression.

Manual Scram

Scram refers to the rapid emergency shutdown of the reactor (generally, insertion of control rods into the core at once by means of gravity). Control rods are automatically inserted into the core when detecting excessive reactor output increase due to causes as significant earthquake magnitude. In the event automatic scram fails or operating staff judgment to scram, all control rods will be inserted immediately by pushing the shutdown mode button.

Defense-in-Depth

Defense-in-depth of nuclear reactor refers to:

Level 1: Prevention of abnormal operation and failures

Level 2: Control of abnormal operation and detection of failures

Level 3: Control of accidents within the design basis

Level 4: Control of severe plant conditions including prevention of accident progression and mitigation of severe accident consequences

Level 5: Mitigation of radiological consequences incurred by significant radioactivity release.

Scram

A rapid emergency shutdown of the reactor (generally, insertion of control rods into the core at once by means of gravity). Control rods are automatically inserted into the core when detecting excessive reactor output increase due to causes as significant earthquake magnitude.

Stakeholder

Interested party; concerned party.

Control Rod

A rod or a plate that can be inserted into, or retracted from the core of a nuclear reactor to control its power; contains materials that absorb neutrons.

DEC (Design Extension Conditions)

Refers to accident conditions exceeding design basis and conditions involving radioactive material release but controlled within the acceptable limit; given consideration via best estimation technique in the design process. DEC includes severe accident conditions.

Design Basis Accident (DBA)

Same definition as “accident” defined in the Evaluation Guidelines. In the Safety Standards, the Nuclear Regulation Authority has renamed “accidents” to “design basis accidents”.

However, there is not necessarily a unified, coherent definition of design basis accident. As cited in “Design Basis Event”, different design basis events are determined for each facility; whereas, in ensuring safety of the entire plant system, points of issue evolve over whether the event is an internal or an external event.

Design Basis Hazard (DBH)

A postulated hazard in benchmarking safety design of nuclear facilities; that the facilities must be designed to withstand without loss to SSCs. DBH are named differently in accordance with the initiating event, such as “design basis seismic motion” for earthquakes, “design basis tsunami height” for tsunami, and “design basis threat” for terrorist events.

An initiating event that exceed design basis hazard conditions does not imply that it falls in the region of defense-in-depth level 4. For example, in the event of beyond design seismic motion, if plant operation is not suspended via the seismometer and SSCs maintain integrity because of safety margin, then the event would be evaluated as a level 2 event.

Preclusion of Preceding Defense Levels

Defense-in-depth concept in precluding the preceding level of defense barrier.

Scenario Tsunami

Anticipated tsunami height taken into account in the design safety criteria; in the new Safety Standards compiled by the Nuclear Regulation Authority, “scenario tsunami” is called “design basis tsunami”.

Redundancy

Refers to the presence of alternative equipments, systems so that any of the equipment/system will perform a function even in the event of failure of the other, for example, provision of multiple, identical EDGs exceeding sufficient volume.

Diversity

Diversity refers to the provision of two or more redundant components or methods to perform an identical function, for example, diversification of reactor shutdown by control rod insertion and injection of boric acid solution.

Chernobyl Accident

The worst nuclear accident so far occurred on April 26, 1986 at the Chernobyl Nuclear Power Plant in Unkraine. It was rates as level 7 incident (major accident) on the International Nuclear Event Scale (INES) standard.

Independence

Refers to conditions where more than two systems or components under design operating conditions, are unaffected by common cause or dependent cause, by the operation or failure of other systems or components. Systems for the operation and systems for ensuring safety must be designed to function independently, so that failure of one system does not hamper performance of the other system.

Hazards

The magnitude of each initiating events (or capacity, strength, height).

Back-checking

To review; systematic reassessment of plants, and existing SSCs (systems, structures, components) against current standards.

Back-fitting

The modification or addition to SSCs (systems, structures, components) or design of a plant or facilities to meet current safety standards.

Heat Sink

A medium in which the heat is absorbed, or removed.

Filtered Vents

Containment venting component with filters, which can reduce radioactivity to $1/100^{th}$ to $1/1000^{th}$. The reduction rate of radioactivity with wet filter systems is $1/10^{th}$ the $1/100^{th}$.

Blowout Panel

A temporary cover over an opening in the reactor building which will automatically blow out (open) to relieve internal building excess pressure and to protect against explosion.

Becquerel (Bq)

SI-derived unit of radioactivity; 1 Bq is defined as the activity of a quantity of radioactive material in which one nucleus decays per second.

Pedestal

Vacant space at the base of the reactor pressure vessel. Different from the reactor pedestal.

Venting

Venting strategies to reduce pressure and temperature build-up in the containment are wet vents, or

wet filters, in venting via suppression pool located at the base of the containment structure and dry vents, in venting by opening vents to release radioactive material directly into the atmosphere.

Melt-through

The condition in which melting of the fuel core (meltdown) leads to runaway melting of the fuel out of the pressure vessel or the containment vessel.

Corium (Fuel Debris)

Lava-like molten mixture of portions of molten fuel, fuel cladding, fuel assembly, structural materials from the affected parts of the reactor and the core.

Risk

Danger; possibility of suffering harm; the likelihood of unanticipated conditions.

Core Spray System

Of the emergency cooling systems classified as ECCS (emergency core cooling system), the core spray system, comprising of HPCS (high-pressure core spray system) and LPCS (low-pressure core spray system), delivers water from the upper part of the pressure vessel to cool the core.

Resilience

The ability to recover or resume.

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