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Severe Accident and Accident Management

Incorporated administration agency

Japan Nuclear Energy Safety Organization

JNES

--- Outline ---

The "severe accident" is an accident of which consequences exceed largely the extent of accidents having been considered up to now in the traditional design of nuclear power plants, and which would result in core damage.

The "accident management" (AM) means measures to prevent the occurrence of a severe accident by taking full advantage of capacities of equipment in a nuclear power plant, and mitigate its consequences even if a severe accident occurs.

Nuclear power plants have adopted a "defense in depth" principle to ensure safety, for which the following three levels of measures are taken:

(1) For measures of "preventing the occurrence of anomalies", the safety design taking into account a margin to the temperature and pressure of each component is incorporated. Systems to prevent malfunctions or inadvertent activation are also adopted.

(2) As measures to "detect an anomaly at an early stage and prevent it from escalation so as not to lead to an accident," provisions for early detection by automatic monitoring systems are made. Furthermore, design arrangements for shutting down the reactor are made in the case when an anomaly is detected.

(3) As measures to "prevent the escalation of an accident even if it occurs" and "mitigate its consequences," for example, for the case of a pipe rupture accident, systems to inject a large amount of water into the core for cooling (emergency core cooling systems) and a steel-made reactor containment to confine radioactive materials are provided.

It is needless to say that ensuring safety based on the principle of "defense in depth" is a responsibility of licensees (electric utilities), but the Government has a responsibility to examine the adequacy of safety management performed by the licensees in order to ensure public safety. Therefore, prior to the operation of a nuclear power plant, the following safety regulations are implemented:

(1) When an electric utility intends to install a nuclear power plant, it must apply for a reactor establishment permit to the Minister of Economy, Trade and Industry, in accordance with the "Act for the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors" (Law No. 166, 1957). The National Government reviews the conformity of the application contents to the licensing criteria. Adequacy of the design, safety margins, etc. are reviewed in the licensing process by assuming events (design-basis events) which are caused by internal factors at the power plant, such as component failures and human errors.

(2) An electric utility, after obtaining a establishment permit, must obtain an approval for the design and construction plan of the nuclear power plant from the Minister of Economy, Trade and Industry (Article 47, the Electricity Utility Act, 1964) .

(3) Even after obtaining the approval for the design and construction plan, it is required to

undergo pre-service inspections at various construction stages to examine the conformity to the contents approved by the Minister of Economy, Trade and Industry (Article 49, the Electric Utility Act).

In the subsequent operation phase, various kinds of inspections and safety regulation such as aging management are implemented by the Government. With these safety measures, the risk of nuclear reactor facilities to the accident called "severe accident" which would cause severe core damage greater than design-basis events is "judged to be adequately low" since the occurrence of the severe accident is practically unlikely from an engineering viewpoint." However, the probability of the severe accident is not zero. There could be a case that multiple human errors would render safety systems inoperable, or multiple simultaneous initiating events would progress to an unpredictable situation. In 1979, at the Three Mile Island Nuclear Power Station in the U.S., a serious accident involving core damage occurred. It was the first of its kind in commercial nuclear power plants. Since most radioactive materials released from the nuclear reactor were confined in the containment, the effects of this accident on the health of the residents in the vicinity were negligible. On the other hand, the accident of the Chernobyl Nuclear Power Station in the former Soviet Union in 1986 released a large amount of radioactive material from the nuclear reactor to the atmosphere as a result of the core meltdown and explosion, and contaminated very large areas. According to the International Atomic Energy Agency (IAEA) and the World Health Organization (WHO), a large number of deaths directly associated with the accident would occur, and it would cause much human suffering due to cancer induced by radiation. The Nuclear Safety Commission of Japan pointed out that the safety of the Chernobyl Nuclear Power Station was not adequate, since the Chernobyl reactor facility had not been provided with a reactor containment vessel as opposed to the Japanese reactor design.

Taking these accidents as lessons learned, the importance of risk management for a severe accident came to be recognized, and the Nuclear Safety Commission of Japan recommended development of the accident management, and accepting this recommendation, the Nuclear and Industrial Safety Agency (NISA) requested electric utilities to prepare their accident management by conducting the probabilistic safety assessment (PSA) and taking into consideration the safety characteristics of each nuclear reactor facility. While the Japanese National Government recognized that further safety regulations were unnecessary as the safety of nuclear power plant in Japan was fully ensured by the present safety measures, it recommended that electric utilities should perform self-disciplined safety efforts in order to reduce a risk of accident and to further enhance safety. The electric utilities performed PSA-based studies during periodic safety review performed every ten years for nuclear reactor facilities in operation, and prepared their accident management measures for all existing nuclear reactor facilities in Japan by 2002.

(Refer to the text "Probabilistic Safety Assessment" (PSA) for the PSA.)

The Government made requests to the licensee regarding the accident management as follows:

(1) For new nuclear reactor facilities, the arrangements for implementing accident management including specific plans on equipment, preparation of procedures, personnel training, implementing and supporting organizations shall be established in the detailed design phase or in the subsequent phase.

(2) For nuclear reactor facilities under construction or in operation, the arrangements for implementing accident management shall be gradually established.

(3) The probabilistic safety assessment (PSA) methodology shall be used for establishment of accident management.

Since PSA demonstrates what kinds of accident scenario contribute to the risk, it makes it clear how to establish much more effective measures for ensuring safety. PSA was performed on all nuclear power reactor facilities by 2002, and the results showed that the frequency of occurrence of a core damage accident is 1/100,000 or less per one year for one reactor and the frequency of occurrence of an accident leading to containment damage is 1/1,000,000 or less per one year for one reactor.

Contents

1. Severe Accident and Accident Management

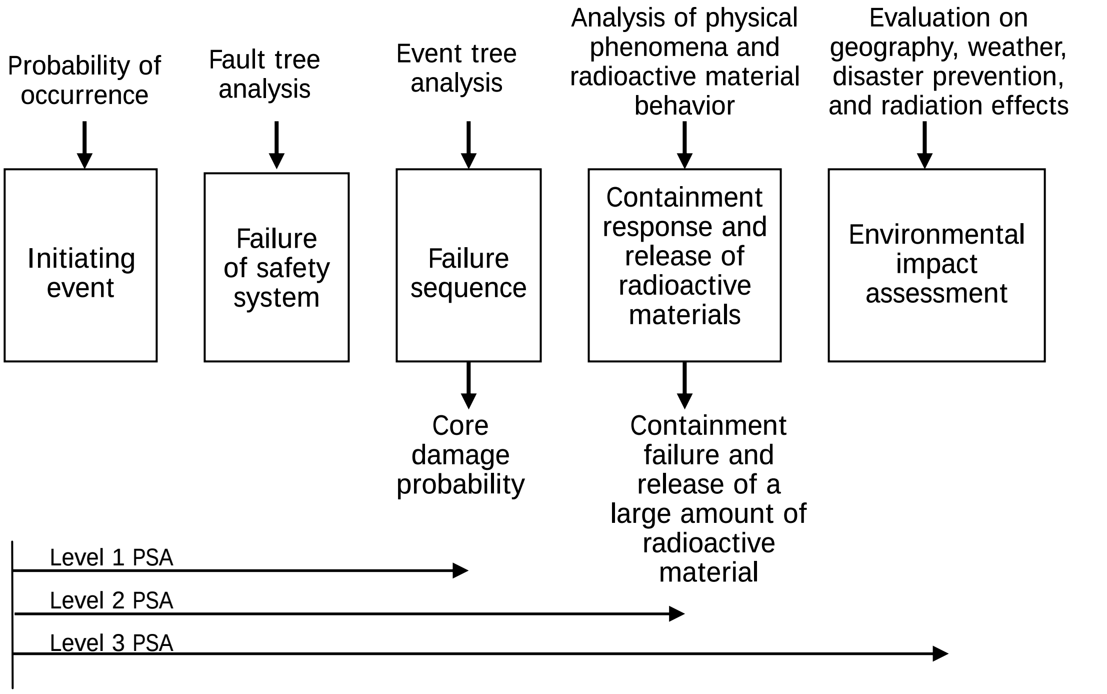
When an anticipated operational occurrence or a design basis accident occurs at an LWR, its safety is ensured by confining radioactive materials within the facility by structures, systems and components which have functions of "prevention" and "mitigation." However, the Three Mile Island (TMI) Accident in the U.S. (1979) and Chernobyl Accident in the former Soviet Union (1986) were events exceeding the design-basis accident. This indicates that there could be an event that neither appropriate reactivity control nor core cooling can be achieved by means assumed in safety design, resulting in significant core damage.

As preconditions for establishing the accident management (AM) assuming a severe accident exceeding design-basis events, it is assumed that the safety protection system, reactor shutdown system, reactor containment system, emergency core cooling system, and emergency power supply system, etc. with protection and mitigation functions fail to function in addition to an occurrence of an anticipated operational occurrence or a design basis accident. The anomalies of Protection System or Mitigation System functions could lead to a severe accident (SA), resulting in a release of a significant amount of radioactive material.

The accident management is to take measures, in the event of a severe accident, to prevent the release of a large amount of radioactive material and mitigate its consequences by operational actions. It also includes design changes in the "non-safety but effective for accident mitigation" equipment to be made available as a protection system or a mitigation system during the accident.

2. PSA

The PSA is a method to evaluate safety by assuming that the safety protection system, reactor shutdown system, reactor containment system, emergency core cooling system, and emergency power supply system, etc. with protection and mitigation functions fail to function during anticipated operational occurrences or design basis accidents. The anomalies of Protection System or Mitigation System functions could lead to a severe accident (SA), resulting in a release of a significant amount of radioactive material. The basic PSA flow of a severe accident is shown in Figure 2-1. Refer to the text "Probabilistic Safety Assessment (PSA)" for PSA details.



Level 1 PSA; Accident sequence frequency leading to core damage
Level 2 PSA; Accident sequences that release a large amount of radioactive material to the environment
Level 3 PSA; Evaluation from the source term to the atmospheric diffusion and exposure risk assessment of the general public in the vicinity of nuclear reactor facility

Figure 2-1 Basic PSA flow of a severe accident

An initiating event of an anticipated operational transient or an accident could develop into a severe accident when the safety system consisting of multiple systems does not perform its function as designed due to a failure of equipment or erroneous operator actions. For instance, in evaluating the design-basis accident, it is verified that safety is ensured even assuming a failure of one ECCS system and a loss of offsite power during a loss of coolant accident for BWRs. In the event tree analysis, the subsequent developments after an occurrence of a loss of coolant are described in an event-tree format to ensure that there is no omission of the developments in the situation. In the event tree, all safety systems to be activated during a loss of coolant accident, and paths of success or failure of the safety system functions are shown. And the "ultimate conditions brought on the nuclear reactor facility" regardless of success or failure of the safety system functions are shown. The "ultimate conditions of the nuclear reactor facility" include core damage and containment failure.

On the other hand, a fault tree analysis is performed on one safety system to determine its failure probability. In the fault tree, factors for mechanical component failures, electric component failures, power supply failures, utility failures and erroneous operator actions to determine a failure rate of one safety system are represented in a tree format, and the failure probability of the safety system is derived from the failure rates of mechanical components and others.

The probability to result in "ultimate conditions of the nuclear reactor facility" is derived from the product of a probability of an initiating event of "loss of coolant" and failure probabilities of multiple safety systems on the paths of the event tree. When this probability exceeds the following "safety goals," it is necessary to implement the accident management (AM) measures.

Safety goals: "Occurrence frequency of a core damage accident shall be 1/100,000 or less per one nuclear reactor for one reactor and that of an accident which results in a containment failure shall be 1/1,000,000 or less per one nuclear reactor for one reactor.

PSA results provide relative importance of the components, systems and structures, and reveal weak points for ensuring nuclear reactor safety. These results are also used to improve the components, systems and structures, as well as operation management, and promote the efficiency of the measures for ensuring safety.

The information obtained by PSA is used to improve the components, systems, structures and management for ensuring safety, and numerically indicate risks to serious accidents (to put it differently, a safety level) of a nuclear reactor facility. The risks should be made as small as possible. But it will be an effective method to set out negligible risk levels (safety goals) and to improve safety in terms of these levels. Nowadays, many nuclear developed nations have set out safety goals to avoid a large risk to individual residents in the vicinity of nuclear power stations. Generally speaking, the negligible risk to the individuals is smaller than 10^{-6}/year (once every million years). There are many concepts regarding the negligible individual risk and its basis, and its acceptance is to be determined by the society. The individual risk 10^{-6} per year is equivalent to several occurrences in a long history of human beings (about 4 or 5 million years after the first human appeared on the earth).

Japanese electric utilities completed PSAs for all existing nuclear reactor facilities (52 units) by 2002 and prepared the accident management measures. The electric utilities conducted a PSA for every type of their nuclear reactor facilities, and the PSA results produced an enormous amount of data. The accident management measures taken are provided below.

3. Accident Management of LWR-Type Nuclear Power Reactor Facilities in Japan

3.1 Accident Management (AM)

As the results of extensive implementation of PSAs, measures to prevent an occurrence and escalation of a severe accident and to mitigate its consequences have been taken, focusing on:

(1) To shutdown a nuclear reactor;

(2) To cool a core;

(3) To remove heat from and confine radioactive materials in the containment; and

(4) To enhance safety functions as the accident management.

Details of accident management for BWRs and PWRs are shown in Table 3-1 and 3-2, respectively, and examples of accident management for BWRs and PWRs are illustrated in Figure 3-1 and 3-2, respectively.

Electric utilities issued the results of PSAs after AM measures were established. The results satisfy the safety goals and some of those are as follows:

- Frequency of core damage (/reactor year): 1.6×10^{-7} (an example for existing BWR-4)
- Frequency of containment failure (/reactor year): 1.2×10^{-8} (an example for existing BWR-4)
- Frequency of core damage (/reactor year): 2.4×10^{-8} (an example for existing BWR-5)
- Frequency of containment failure (/reactor year): 5.5×10^{-9} (an example for existing BWR-5)

Table 3-1 Accident management for BWRs

Classification	Measures	Details
(1) Alternative reactivity control function	Alternative control rod insertion	The instrumentation and control system provided separately from the existing reactor shut-down system detects reactor pressure high and reactor water level low, and shuts down a reactor by automatic control rod insertion.
	Alternative recirculation pump trip	The instrumentation and control system provided separately from the existing reactor emergency shutdown system detects reactor pressure high and reactor water level low, and shuts down automatically reactor recirculation pumps to reduce reactor power.
(2) Water injection function to the nuclear reactor and containment	Alternative water injection	When operation of the low pressure coolant injection system goes wrong, the water is injected to a nuclear reactor from the existing makeup water system and the fire-extinguishing basin system.
	Automation of reactor depressurization	When the high-pressure water injection system cannot function because of high reactor pressure, the reactor depressurization system is automatically actuated on reactor water level low to inject water by the low pressure coolant injection system.
(3) Heat removal function from the containment	Alternative heat removal	When the residual heat removal system gets out of order, while getting it restored, alternative heat removal by the existing drywell coolers and the reactor coolant cleanup system is used to remove heat from the containment.
	Restoration of failed components of the residual heat removal system	Getting the residual heat removal system restored using the time allowed in the course of the accident, it will be used for containment heat removal.
	Containment venting	Providing a containment vent piping with enhanced resistance to pressure to prevent containment over-pressurization for containment heat removal
(4) Enhancement of safety functions	Means for power supply availability	The AC power supplies for driving equipment are shared between adjacent nuclear reactor facilities. Moreover, for a stand-alone plant, the power supply from a diesel generator dedicated for the high pressure core spray system is made available within the nuclear reactor facilities.

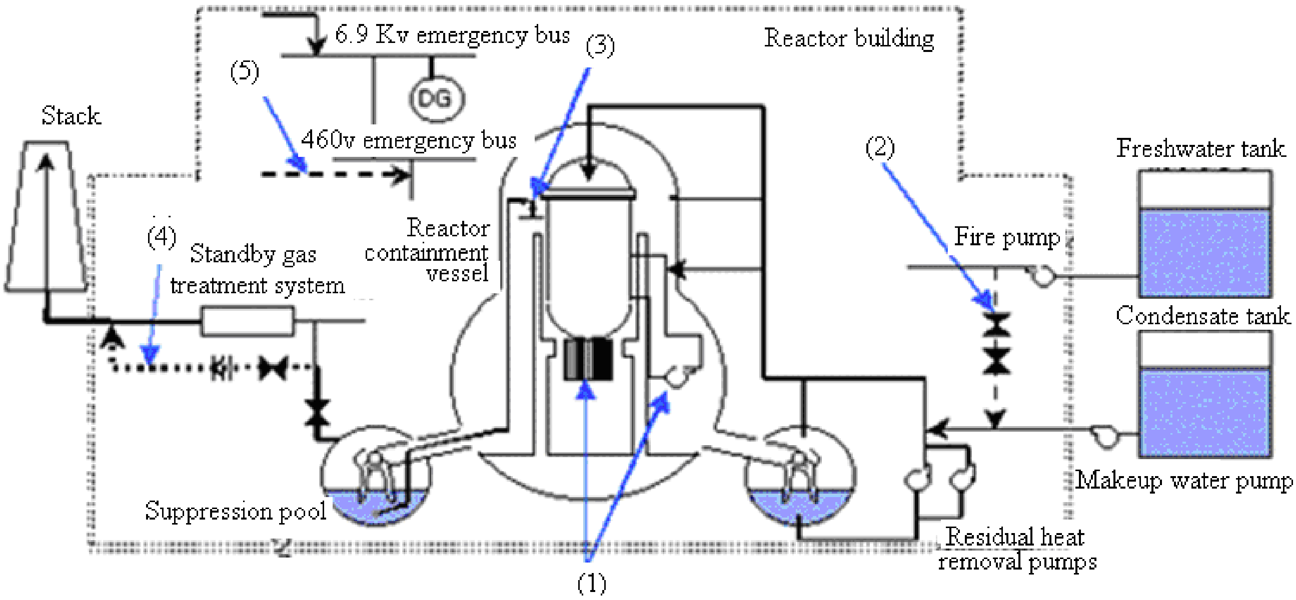


Figure 3-1 Illustration of examples of accident management for BWRs

The following are the supplementary descriptions on accident management measures currently implemented in BWRs:

(1) Alternative reactivity control function

Detectors and the circuits for reactor pressure and reactor water level completely independent of the existing reactor shut-down system are provided to add a function to insert automatically control rods and to shutdown a nuclear reactor. When the existing reactor shut-down system does not function due to a failure of its detectors or circuits, this function can control reactivity. And, providing completely independent detectors and their circuit for reactor pressure high and water level low, functions to automatically stop the reactor recirculation pumps and to reduce the reactor power are added. These functions use reactor characteristics that a reduction in core cooling water flow due to the shutdown of the reactor recirculation pumps increases core void fraction, and then, increased void fraction lowers reactivity due to its negative reactivity characteristic.

(2) Water injection function to the nuclear reactor and reactor containment vessel

This is a measure to use existing equipment capable of injecting cooling water into the nuclear reactor when the low pressure coolant injection system fails to function. This measure is to obtain an ability to inject water into the nuclear reactor and containment using the pumps of existing condensate makeup water system or the fire protection system by changing their piping connections etc.

Furthermore, when the high pressure water injection system does not function because of high reactor pressure, the reactor depressurization system is automatically actuated on the reactor water level low to reduce reactor pressure and enable the low pressure coolant injection system to inject water. The reactor depressurization system, using the function of safety relief valves, opens those valves and depressurizes the reactor vessel.

(3) Function to remove heat from the reactor containment vessel

This is based on a concept that when the residual heat removal system fails, in order to obtain its restoration time, the alternative heat removal is performed using the existing drywell coolers and the reactor coolant cleanup system, although their heat-removal capacity is small, to wait for a recovery of the residual heat removal system from its failure. In addition, the containment decay heat removal function is improved by newly providing the containment vent piping with enhanced resistance to relieve the containment pressure to a stack.

(4) Enhancement of safety functions (means for power supply availability)

Circuits and breakers are provided so as to make a high-pressure AC power supply available from adjacent nuclear reactor facilities. There are many cases that multiple nuclear reactor units are installed on the same site, so the nuclear reactor facility of Unit 1 are so designed to be able to receive its power supply from Unit 2 or Unit 3. Moreover, for a stand-alone plant without an adjacent nuclear reactor facility, the power supply from the diesel generator dedicated for its high pressure core spray system is made available within its nuclear reactor facility.

Table 3-2 Accident management for PWR

Classification	Measures	Details
(1) Reactor shutdown function	Diversification of emergency secondary system cooling	In the case of failures to shutdown a nuclear reactor and to start the auxiliary water supply system, the core is cooled using the secondary system via steam generators by manual start-up of the main feed water system. (Measures taken by improvement of procedures using existing equipment.)
(2) Core cooling function	Utilization of the turbine bypass system	In the case of multiple failures of the high pressure injection system and the containment spray system, the heat removal is performed using the secondary system by manual opening of turbine bypass valves.
	Alternative recirculation	In the case of a re-circulation failure of the emergency core cooling system, the residual heat removal system and the containment spray system are connected to inject water from the containment spray system into the core.
	Alternative auxiliary equipment cooling	In the case of a loss of functions of the reactor auxiliary equipment cooling water system, the cooling water for HVAC is fed to the auxiliary equipment cooling water system of the residual-heat-removal pumps to make them operable for core cooling.
	Cooling-down & recirculation	In the case of a failure to isolate leaking locations during a steam generator tube rupture accident, water is injected into the nuclear reactor using the emergency core cooling system etc., and while the leakage is controlled by heat removal with steam generators by operation of main steam relief valves etc., and by depressurization of the nuclear reactor by operation of pressurizer bypass valves,

		etc., the core is cooled by the residual heat removal system.
(3) Radioactive material confinement function	Natural convection cooling within the containment	In the case of a containment atmospheric pressure rise due to a failure to actuate the containment spray system, the nuclear reactor auxiliary equipment cooling water is fed to the normal containment recirculation units to lower the temperature and pressure in the containment by natural circulation in the containment and to thereby maintain the containment integrity.
	Water injection into the containment	In the case of a failure to actuate the containment spray system, water of the fresh water tank is sprayed from the spray header using fire pumps by changing the piping connections.
	Forced depressurization of the primary system	When the nuclear reactor is in a high-pressure condition due to a loss of function of the high pressure injection system and a loss of the heat-removal function of the secondary system, the nuclear reactor is depressurized by manual opening of pressurizer bypass valves to prevent the direct heating phenomenon of the containment atmosphere.
	Planned hydrogen combustion	Hydrogen generated by the metal water reaction is burned using the igniter to prevent a large-scale combustion in the hydrogen high-concentration condition (this measure is only taken at some nuclear reactor facilities).
(4) Enhancement of safety functions	Alternative auxiliary equipment cooling	In the case of a loss of functions of the reactor auxiliary equipment cooling water system, the cooling water for HVAC is fed to the auxiliary equipment cooling water system of the residual-heat-removal pumps to make them operable.
	Power supply sharing between units	In the case of a loss of all AC power supplies, using the high-pressure power supply line from adjacent nuclear reactor facilities, the emergency power is fed to the emergency bus line on the side with a loss of AC power supply.

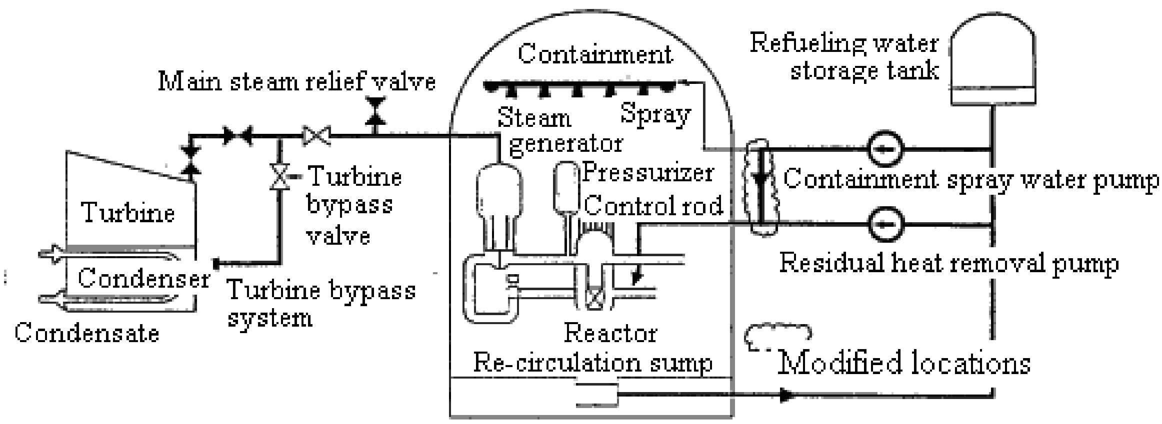


Figure 3-2 Illustration of examples of accident management for PWRs

The following are the supplementary descriptions of currently implemented accident management measures in PWRs:

(1) Reactor shutdown function

In the case of a failure to shutdown a reactor and start the auxiliary feedwater supply system, in order to remove the heat from the nuclear reactor, the main feed water system that has been automatically shutdown at the beginning of the accident is started manually to cool the core through the steam generators. This is a measure adopted by improvement of procedures using existing equipment.

(2) Core cooling function

Utilization of the turbine bypass system:

In the case of multiple failures of the high pressure injection system and the containment spray system, it is possible to remove the heat of nuclear reactor by means of steam evaporation heat using the existing equipment, i.e. opening the main steam relief valves to release the steam of the secondary side of steam generators, but assuming the case that the main steam relief valves fail to open, the steam is led to the condenser through the turbine bypass system to release the heat to the sea water system. This is a measure taken by improvement of procedures using existing equipment.

Alternative re-circulation:

During an accident, the ECCS injects water into the core from the external tank with boric acid water stored, and when the water in the tank runs out, the ECCS circulates the water collected in the recirculation sump (the primary coolant that flowed out during a pipe rupture etc. and the ECCS water that was injected by the ECCS and flowed out of the pipe rupture opening) to cool the core, but in case the switching to the recirculation operation fails, the connecting line is newly provided between the residual heat removal pumps and the containment spray water pumps to inject the water into the core. This is a measure taken by addition of equipment and improvement of procedures.

(3) Radioactive material confinement function

Natural convection cooling in the primary containment vessel:

In the case that the containment atmosphere pressure goes up due to a failure of the containment spray system actuation, in order to prevent a failure of the containment, the reactor auxiliary equipment cooling water is fed to the normal containment recirculation units to lower the temperature and pressure in the containment by natural circulation within the containment to maintain the containment integrity.

Water injection into the containment:

In order to prevent a pressure rise in the containment that could lead to its failure during a loss of function of the emergency containment cooling system, a line is provided so that the water of the fresh water tank can be injected into the containment via the containment spray by fire pumps to remove heat from the containment.

3.2 Implementation of Accident Management

Accident management (AM) measures explained in "3.1 Accident Management" are measures to prevent a severe accident using all available equipment under the assumption that the safety systems do not function properly because of multiple failures. Therefore, in case AM has to be implemented, its operation shall be clearly specified, and its organization, operation manuals, communication system, etc. shall be made well-known to the persons involved. At nuclear power stations in Japan, the organization to implement AM and its supporting organizations including duty allocations, responsible persons, etc. have been established.

(1) Organization

To implement the accident management, operator supporting organizations in addition to operators are provided for technical evaluation, communication, dose assessment, restoration, etc.

Operational actions are carried out mainly at the discretion of a central control room. Furthermore, when an accident has escalated to core damage and requires overall judgment, the supporting organization provides the technical evaluation to support the decision-making of operators.

(2) Procedures

Electric utilities have prepared procedures for operators and for supporting organizations, according to the event progression. For operators, in addition to emergency operation manuals (for BWR plants) and emergency operation procedures (part II) (for PWR plants) for preventing core damage, emergency operation manuals (for BWR plants) and emergency operation manuals (part III) (for PWR plants) for post-core damage have been established to respond to a severe accident. For supporting organizations, in order to judge the mitigation measures in a comprehensive manner after core damage, the accident management guide (for BWR plants) and the assessment manual for actions to mitigate the consequences of an accident (for PWR plants), including procedures and criteria, knowledge bases such as technical data, and projected consequences have been prepared.

3.3 Education, Training and Communication System

Electric utilities have their programs in place for periodic education on knowledge base to enhance knowledge on the plant behavior during a severe accident, and drills for examining the effectiveness of the organizations for implementing accident management. The knowledge base is the organized information including types of information and criteria for identifying operating status and making correct decisions during the implementation of the AM measures.

Furthermore, systems have been developed to provide a training to notify to and communicate with the internal and external organizations about the plant conditions and the status of implementation of accident management measures, and also to exchange information and to receive guidance and advices.

4. Conclusion

Electric utilities in Japan have developed accident management as one of the efforts to ensure safety. It can be said that this is the preemptive effort of the electric utilities to achieve safety goals. Accident management measures are the arrangements taken by the electric utilities according to the sequences to further enhance the safety after performing the probabilistic safety assessment based on the detailed design of their nuclear power reactor facilities and understanding the safety characteristics of those from viewpoints of the "core integrity" and "containment integrity".

As mentioned in this text, nuclear safety regulation can be effectively implemented based on more scientific and rational methods as well as judgments.

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