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February 6, 2013

Outline of New Safety Standard (Design Basis)
(DRAFT)

For Public Comment

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Table of Contents

(Caution)

- This document contains the new safety standards (Severe Accident Management) outline compiled based on the discussions by the "The Study Team on the New Safety Standards for Light Water Reactor for Electric Power Generation".
- This document compiles regulatory requirements in a free format. Legislative structure, definitions of terms, and detailed provisions shall be determined in the future in accordance with legal examples.
- Review of the "Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities" (Nuclear Safety Commission Decision August 30, 1990) is planned. Item "2. (7) Design considerations for common use" and "Item 2. (9) Design considerations for reliability" are planned for review in advance of developing future regulatory guides.

1. General

(1) Definitions of terms

The terms in the following items as used in this draft outline are defined according to the provisions of each of the relevant items (corresponding to the definitions in the design guidelines).

①"Safety function" is the function of necessary structures, systems, and components to ensure safety of the reactor facility and may be categorized as following.

1) Those that, if lost, may lead to abnormal transients during operation of the reactor facility and design basis accident, leading to excessive radiation exposure to the general public or workers.

2) Those that prevent the expansion of or quickly resolve abnormal transients during operation of the reactor facility or design basis accidents thereby preventing or mitigating excessive radiation exposure to the general public or workers and radioactive contamination of the environment outside of the reactor site premises.

②"Importance of safety functions" is the level of importance of the safety function from the perspective of ensuring the safety of the reactor facility.

③"Normal operation" is planned startup, shutdown, power operation, hot standby, refueling and other operations of the reactor facility that are within designated limits of the operating conditions.

④"Abnormal transients during operation" are abnormal conditions that arise due to foreseen equipment single failures or spurious actions, single operator erroneous operations during the lifetime of the reactor facility or caused by disturbances predicted to occur at similar frequencies.

⑤"Design basis accident" (DBA) is an abnormal condition exceeding an "abnormal transient during operation" which has an occurrence frequency that is rare but is assumed to occur from the perspective of the safety design of the reactor facility.

⑥"Reactor containment vessel boundaries" are the equipment that are designed to be the pressure barrier against assumed reactor containment vessel events and which form the barrier against release of radioactive materials.

⑦"Reactor coolant pressure boundaries" are the equipment that contain reactor coolant (primary coolant for pressurized water reactors) during normal reactor operation, and which constitute the pressure barrier during abnormal transients during operation and design basis accidents, leads to loss of coolant accident, if damaged

⑧"Reactor coolant systems" are systems for reactor coolant which directly cool the core during normal reactor operation (primary cooling system in pressurized water reactors, primary loop recirculation system, main steam system, and feedwater system for boiling water reactors).

⑨"Reactor cooling systems" are the systems to remove heat from the reactor during normal reactor operation, abnormal transients during operation, and design basis accidents (including reactor coolant system, systems to remove residual heat, emergency core cooling system, secondary cooling systems (for pressurized water reactors), and systems to transfer heat to the ultimate heat sink).

⑩"Reactor shutdown systems" are the systems designed to bring the reactor subcritical by injecting negative reactivity to the reactor from critical or supercritical conditions.

⑪"Reactivity control systems" are the systems designed to adjust reactivity change depending on the reactor output, burn-up, fission products, temperature, and other elements by controlling the reactivity of the reactor.

⑫"Safety protection systems" are the systems designed to detect abnormal transients during operation of reactor facilities and design basis accidents, and, if necessary, directly actuate the reactor shutdown systems, engineered safety facilities, and other systems.

⑬"Engineered safety facilities" are the equipment designed to constrain or prevent massive release of radioactive materials in the unlikely event of failure of fuel in the reactor caused by damage or failure of the reactor facility.

⑭"Single failure" refers to the loss of prescribed safety functions due to failure of one component. It also includes multiple failures based on dependent causes. "Dependent causes" refers to causes that occur inevitably due to a single cause.

⑮"Active components" refer to the components that actively execute prescribed functions in response to actuation signals or inputs from components, such as actuators, other than the said one.

⑯"Passive components" refer to the components that are not active components.

⑰"Redundancy" refers to having two or more systems or components that have the same properties with the same functions.

⑱"Diversity" refers to having two or more systems or components that have the different properties with the same function. "Different properties" as referenced herein refers to having different operation principles and functions that are not simultaneously hindered by common causes or dependent causes. "Common cause" refers to the causes that act simultaneously on two or more systems or components such as impact factors of environmental temperature, humidity, pressure, or radiation; and, impact factors of power, air, oil, cooling water supplied to systems or components, in addition to, impact of earthquakes, flooding, or fire.

⑲"Independence" refers to when the functions of two or more systems or components are not simultaneously hindered due to common causes or dependent causes during environmental and operating conditions considered in design.

⑳"Allowable design limit of fuel" is the allowable level of fuel damage in terms of safety with regard to reactor design and is the limit at which the reactor may continue operation. "Reactor may continue operation" herein does not necessarily refer to operation of the reactor in an as-is condition, but includes restart of operations after repairing the relevant failure and inspect/replace the fuel, if necessary.

1. General

(2) Applicable standards

[Basic requirement]

Design, selection of materials, production, and inspection of structures, systems, and components with safety functions shall be according to the codes and standards recognized as appropriate in consideration of the level of importance of the intended safety function.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 1)

[Requirement Details]

A In principle, design, selection of materials, production, and inspection of structures, systems, and components with safety functions shall be according to the codes and standards based on current domestic laws. However, if the applied codes and standards are from other countries or if the codes and standards are not generally used, it is necessary to clarify the basis for applying these codes and standards, to compare them to domestic laws, and to explain the validity of their application.

B "Shall be according to the codes and standards" means that the applicable codes and standards shall be clarified regarding the design, selection of materials, production, and inspection of applicable structures, systems, and components.

2. Common technical requirements for reactor facilities

(1) Design considerations for natural phenomena (Guide 2)

[Basic requirement]

(Earthquakes, tsunamis (including accompanying events)

1. Structures, systems, and components with safety function shall be classified for seismic design considering the importance level of its safety function and the impact on safety if functions were lost due to earthquake, and it shall be designed to sufficiently withstand the design seismic force as considered appropriate.

(* The above is written based on the current Safety Design Regulatory Guide, but shall be replaced by the results of the on-going separate study team examining design basis earthquakes/ tsunamis (including accompanying events).)

(Natural phenomena other than earthquakes)

2. Structures, systems, and components with safety functions shall be designed so as to not impair safety of the reactor facilities due to assumed natural phenomena other than earthquakes, tsunamis, and accompanying events. For structures, systems, and components with particularly high level of importance in their safety functions, the design shall consider the severest conditions among the predicted natural phenomena and an appropriate combination of the natural forces and accident loads.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 2)

[Requirement Details]

A "Design to sufficiently withstand the design seismic force as considered appropriate" shall be according to provisions in the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities (Nuclear Safety Commission Decision September 19, 2006).

B "Design so as to not impair safety of the reactor facilities due to natural phenomena" refers to, in case natural phenomena necessary to be considered for design or a combination of such natural phenomena occurs, the safety function of the equipment shall be achieved under the environmental conditions brought on by the natural phenomena and the environmental conditions arising at the facility as a result.

C “Structures, systems, and components with particularly high level of importance in their safety functions” shall be separately provided for based on the Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities (Nuclear Safety Commission Decision August 30, 1990).

D “Predicted natural phenomena” are those applicable phenomena based on the natural environment of the site including flooding, wind (typhoon), tornado, freezing, precipitation, snow accumulation, lightning, landslide, volcanic effects, biological events, forest fires and others.

E “The most severe conditions among the natural phenomena” are those that may be predicted based on the newest scientific and technological knowledge regarding that natural phenomenon. Based on the past records, results of field investigations, the newest knowledge, the combination with other natural phenomena shall all be considered.

F “A combination of the natural force and accident load” does not necessarily require that the natural force considered the most severe and the accident load to be the maximum load during an accident. Instead it refers to an appropriate combination considering respective cause and chronological relationships between these forces and loads.

2. Common technical requirements for reactor facilities

(2) Design considerations for external human events

[Basic requirement]

(Random event)

1. Structures, systems, and components with safety functions shall be designed so that assumed random external human events do not impair safety of the reactor facility.

(Illegal approach by third parties)

2. The reactor facility shall be designed with the appropriate measures to protect against illegal approach by third parties to those structures, systems, and components with safety functions.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 3)

[Requirement Details]

A "Random external human events" are selected based on the conditions of the site and the site location and refers to missiles (airplane crashes), dam collapses, explosions, fires in nearby factories, toxic gases, ship collisions, electromagnetic interference, and others.

B For airplane crashes, the necessity for a protective design shall be confirmed based on the "Assessment of airplane crash probability for commercial power reactor facility (2009.06.25 NISA-1)" stipulated as of July 30, 1995 and revised as of June 30, 2009 by the former Nuclear and Industrial Safety Agency.

C "Illegal approach by third parties" includes: illegal transport of nuclear material by people on-site, sabotage, transport of explosives or hazardous materials onto the site including using mail and cyber terrorism.

2. Common technical requirements for reactor facilities

(3) Design considerations for internally generated missiles

[Basic requirement]

Structures, systems, and components with safety functions shall be designed so that the safety of the reactor facility is not impaired by missiles that may be assumed to be generated within the reactor facility.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 4)

[Requirement Details]

A “Internally generated missiles” refer to the missiles generated by breakage of valves or pipes containing fluid with high internal energy, breakage of high-speed rotating equipment, gas explosions, and the dropping of heavy equipment. Secondary missiles, fires, chemical reactions, electrical damage, piping breakage, equipment failures and other secondary impacts shall also be considered.

B Assessment of internally generated missiles shall be according to “Assessment of turbine missiles” (Nuclear Safety Commission, Reactor Safety Dedicated Review Committee, July 20, 1977) and other appropriate documents.

2. Common technical requirements for reactor facilities

(4) Design considerations for internal flooding

[Basic requirements]

Structures, systems, and components with safety functions shall be designed so that the safety of the reactor facility is not impaired by flooding that is assumed to occur internally in the reactor facility.

(New)

[Requirement Details]

A “Flooding that is assumed to occur internally in the reactor facility” refers to the flooding caused by breakage of components and piping installed within the reactor facility (including seismically-induced breakage), actuation of fire protection systems, overflow or sloshing of the water in the spent fuel pool or spent fuel pit.

B “Design so that the safety of the reactor facility is not impaired” in these regulations refers to the ability to bring the reactor to hot shutdown, continue to cold shutdown, and maintain confinement functions of radioactive material. If the plant is in a shutdown condition, to the plant must be able to continue to maintain such conditions when internal flooding is assumed to occur. For the spent fuel pool or spent fuel pit, the capability to maintain pool cooling and water supply to the pool must be preserved.

2. Common technical requirements for reactor facilities

(5) Design considerations for fire

[Basic requirement] Reactor facilities shall be designed considering protection measures such as prevention of fires, fire detection and suppression, and mitigation of impacts of fires so the safety of the reactor facilities is not impaired by fires. The protection measures shall be designed so as not to impair the safety functions of structures, systems, and components important to safety due to failure or spurious actuation of the protection measures. (Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 5)

[Requirement Details]

A. "Design considering protection measures such as prevention of fires, fire detection and suppression, and mitigation of impacts of fires" refers to the design that complies with separately defined requirements (※).

((※) Assessment guide is to be developed by the Nuclear Regulation Authority referencing the US and other specification rules.)

2.Common technical requirements for reactor facilities

(6) Design considerations for environmental conditions

[Basic requirement]

Structures, systems, and components with safety functions shall be designed to comply with all environmental conditions for which the safety function is requred.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 6)

[Requirement Details]

A. "All environmental conditions for which the safety function is required" refers to all the environmental conditions to which the structures, systems, and components may be exposed, and for which this equipment is expected to be operable during normal operation, abnormal transients during operation, and during design basis accidents.

2.Common technical requirements for reactor facilities

(7) Design considerations for common use

[Basic requirement]

In principle, among structures, systems, and components with safety functions of particular importance shall not be shared or interconnected between two or more units of reactor facilities. However, this shall not apply if safety is enhanced through shared use or interconnections.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 7)

[Requirement Details]

A Items applicable under “among structures, systems, and components with safety functions of particular importance” shall be determined based on Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities.

B “If safety is enhanced” refers to cases in which consideration is given to benefits resulting from shared use. Nevertheless, the technical requirements for each of the facilities to be shared must be met. For example, control rooms can be designed to be commonly used by the two units at a twin unit plant to enable the share of operators as long as habitability requirements are met.

C “Common use” refers to using the same structures, systems, and components at two or more reactor facilities.

D “Interconnection” refers to connection of systems or components between two or more reactor facilities.

2. Common technical requirements for reactor facilities

(8) Design considerations for operator manipulations

[Basic requirements]

Reactor facilities shall be designed to take appropriate measures to prevent erroneous operations by operators. The safety equipment constituting the reactor facility shall be designed so that the operators can easily operate under environmental conditions in which operation is required.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 8)

[Requirement Details]

A "Design taking into account appropriate measures" refers to the design features that incorporate ergonomics such as: considerations for the location of panels and operability of tools and valves, consideration so the conditions of reactor facility can be accurately and quickly understood through instrument indications and alarm indications, and considerations so errors in maintenance and inspection will be less likely. In addition, it also refers to design that ensure necessary safety functions without expecting operator actions up to a certain period of time after the occurrence of an abnormal transient during operation or design basis accident.

B "Design so that operators can easily operate" refers to the designs that allow operators to easily operate equipment even assuming environmental conditions due to an abnormal event (for example, aftershocks) and environmental conditions that are likely to occur simultaneously with other abnormal conditions assumed for the facility.

2. Common technical requirements for reactor facilities

(9) Design considerations for reliability

[Basic requirement]

1 Structures, systems, and components with safety functions shall be designed to ensure sufficiently high reliability and to allow it to be maintained corresponding to the classification of importance of such safety function.

2 For systems with safety functions of particular importance, it shall be designed to achieve the safety function of the system even when offsite power is unavailable in addition to assuming a single failure of a component that constitutes the system.

3 Therefore, the system in the above paragraph shall be designed with redundancy or diversity and independence considering its structure, operation principles, and nature of safety function it fulfills.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 9)

[Requirement Details]

A. “Ensure sufficiently high reliability corresponding to the classification of importance of such safety function” and “systems with safety functions of particular importance” shall be stipulated separately based on “Regulatory Guide for Reviewing Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities.”

B. “Single failure” can be separated into a single failure of active equipment and a single failure of passive equipment. For systems with safety functions of particular importance, it is necessary to design so that the prescribed safety functions are achieved even when postulating either a single failure of active equipment in the short-term or single failure of active equipment or an assumed single failure of passive equipment in the long-term.

C. The boundary between short and long terms shall basically be 24 hours, and, if operation mode is switched, that timing shall be the boundary between short and long terms. For example, switching of the operating mode for PWRs would be switching from injection mode using emergency core cooling system or containment heat removal system to recirculation mode.

D. For safety function assessment in the long term that should postulate either a single failure of active equipment or an assumed single failure of passive equipment as indicated above, if it is certain that the single failure can be eliminated or remedied within a time period that would not hinder safety even under the severest assumed conditions, it is acceptable not to postulate this single failure.

E. If it is possible to rationally explain that the possibility of occurrence of a single failure is extremely low, or if it can be confirmed through safety analysis and other methods that there is an alternative to that function using other systems even in case system functions are lost when postulating a single failure, requirements for redundancy shall not apply to the relevant component.

2. Common technical requirements for reactor facilities

(10) Design considerations for testability

[Basic requirement] Structures, systems, and components with safety functions shall be designed to allow testing and inspection (hereafter "testing") during reactor operation or shutdown using the appropriate methods corresponding to the classification of importance of the safety function to confirm its integrity and capability. (Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 10)

[Requirement Details]

A. "Appropriate methods" refer to the case when testing or inspection using actual system is inappropriate and use of bypass systems for testing is allowed.

B. "Testing" shall be according to the following items:
   (a) Structures, systems, and components with safety function in standby condition during reactor operation may be tested periodically during operation. However, if testing during operation has a major impact on operation, this shall not apply. In addition, testing can be conducted independently for individual systems and components with redundancy or diversity. "Periodical testing during operation" shall include tests stipulated in "Ordinance of Establishing Technical Requirements (Standards) for Nuclear Power Generation Equipment" (Ministry of International Trade and Industry Ordinance No. 62 (planned to be revised as Nuclear Regulation Authority Rules).
   (b) Regarding the functional test of each channel of safety protection systems during operation, even when conducting the test, the safety protection system function itself shall be maintained and any unnecessary operation of the reactor shutdown system, emergency core cooling system or others shall not occur at the same time.
   (c) "Periodical testing during reactor shutdown" shall include tests stipulated by the laws and regulations related to Reactor Regulation Law.

C. For the facilities listed in the left column in the table below, the requirements in the right column shall be met.

Structures, systems, components	Requirements
Reactivity control system	Design allows testing
Reactor coolant pressure boundary	Design allows testing and inspection while reactor is in service
System to remove residual heat	Design allows testing
Emergency core cooling system	Design allows testing and inspection periodically and testing and inspection can be conducted for each system independently to check that integrity and diversity is maintained
System to transport heat to ultimate heat sink	Design allows testing
Reactor containment vessel	Design allows periodical leak rate test of the overall reactor containment vessel using the prescribed pressure Testing can be conducted for leaks from important areas such as penetrations for cables and piping and access ways
Reactor containment vessel isolation valve	Periodical operation test of the reactor containment vessel isolation valve can be conducted and leak tests of major valves can be conducted
Reactor containment vessel heat removal system	Design allows testing
Systems to control containment facility atmosphere	Design allows testing
Safety protection system	In principle, the design allows periodical testing during reactor operation and each channel can be tested independently to check that its integrity and redundancy is maintained
Electrical systems	Electrical systems related to safety functions of particular importance shall be designed to allow periodical testing and inspection of the important portions of the system in an appropriate manner.
Fuel storage and handling facilities	Structures, systems, and components with safety functions shall allow periodical testing and inspection in an appropriate manner.

2. Common technical requirements for reactor facilities

(11) Design considerations for telecommunication systems

[Basic requirement]

1 The reactor facility shall have telecommunication facilities and alarm systems in place to provide proper instruction to all of the people at the nuclear power station in case of a design basis accident.

2 The telecommunication facilities to offsite locations and telecommunication lines for data transfer facilities shall be dedicated lines and shall be designed with diversity.

3 The telecommunication facilities used between onsite locations shall be designed with diversity.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 45)

[Requirement Details]

A “Telecommunication facilities” refers to the facilities that allow communication such as providing verbal instructions on operation, work, or evacuation from the main control room to personnel on locations inside and outside of the building.

B “Data transfer facilities” refers to the facilities to transfer necessary data from onsite (main control room or other location) to off-site emergency response support system (ERSS).

C “Telecommunication lines shall be dedicated lines and shall be designed with diversity” refers to the design of communication lines with diversity in communication methods (for example cables and wireless). This includes lines that can be used without limitations such as congestion. They include satellite-specific IP phones and other dedicated telecommunication lines developed independently by the reactor licensee or telecommunication lines dedicated for special customers provided by the power and communication operator.

2. Common technical requirements for reactor facilities

(12) Design considerations for evacuation routes

[Basic requirement]

1 Reactor facilities shall be designed with evacuation lighting equipment that will remain functioning when normal lighting power supplies are lost and with safety evacuation routes with simple, clear, and lasting signs.

2 Reactor facilities shall have lighting and dedicated power supply separate from evacuation lighting to allow for work activities in case field work arises for accident response.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 46)

[Requirement Details]

A "Reactor facilities shall have lighting and dedicated power supply to allow for work activities and field work required for accident response" refers to the need to have lighting systems to enable the work activities required for accident response in the reactor facility at any time of the day or night, or at any location. In terms of the emergency of the field work, it is acceptable to consider the use of temporary lighting (portable) as time allows.

3. Individual systems within the reactor facility

(1) Core, etc.

[Basic requirements]

(Core)

1 The core shall be designed in combination of the functions of the reactor cooling systems, reactor shutdown systems, reactivity control systems, instrument control systems and safety protection systems to ensure that allowable design limit of fuel is not exceeded during normal operation or abnormal transients during operation.

2 The composition elements of the core excluding fuel rods, as well as composition elements in the vicinity of the core within the RPV, shall be designed to ensure safe reactor shutdown and core cooling during normal operation, abnormal transients during operation, or design basis accidents.

(Fuel)

3 Fuel assemblies shall be designed to ensure integrity is not lost, even when considering factors which could conceivably occur within the reactor during its period of usage.

4 Fuel assemblies shall be designed to prevent occurrence of excessive deformation during transport or handling.

(Reactor characteristics)

5 The core and relevant systems shall be designed to possess specific output control characteristics, and be able to easily control any output fluctuation that may occur.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 11, 12, 13)

[Requirement Details]

(Core)

A Allowable design limits of fuel shall be set using factors such as fuel pellet maximum temperature, fuel cladding pipe maximum temperature, maximum heat flux, minimum critical heat flux, minimum critical power ratio, fuel pellet maximum enthalpy, and fuel cladding maximum deformation as bases for determination.

B Specific methods of the evaluation shall be as per the "Power generating light water reactor facility reactivity insertion event evaluation policy (determined by Nuclear Safety Commission of Japan on Jan. 19, 1984, partially revised on Aug. 30, 1990)."

(Fuel)

C "Factors which could conceivably occur" refer to the factors such as changes in pressure or temperature, chemical effects, static or active load, fuel pellet deformation, and composition of gas sealed within fuel rods due to the difference between internal and external pressure of fuel rods; as well as irradiation or load on fuel rods and other materials.

D "Design to ensure integrity is not lost" refers to the design which ensures functions such as the confinement function of fuel cladding, as well as insertability and geometries of control rods maintained during both normal operation and abnormal transients during operation for the prescribed operation period.

E Specific methods of the evaluation shall be as per the "Power generating light water reactor fuel design method," (Approved by Nuclear Safety Commission of Japan on May 12, 1988)."

(Reactor characteristics)

F "Possess specific output control characteristics" refers to the reactivity feedback effects (comprising of the Doppler coefficient, moderator temperature coefficient, moderator void coefficient, pressure coefficient, etc.) to control output in a responsive manner for all foreseeable scopes of operation and to prevent or mitigate fuel damage due to excessive changes in reactor output.

G "Easily control any output fluctuations that may occur" refers to the sufficient attenuation capability to prevent allowable design limit of fuel from being exceeded or ability to control output fluctuations.

3. Individual systems within the reactor facility

(2) Reactivity control systems and reactor shutdown systems

[Basic requirements]

(Reactivity control systems)

1 Reactivity control systems shall be designed to maintain stable operation by adjusting foreseeable reactivity changes that are expected to take place during normal operation.

2 Reactivity control systems shall be designed so that the maximum reactivity worth and reactivity insertion rate of control rods will not cause damage to the reactor coolant pressure boundary in case of a postulated reactivity insertion event, and also ensure that destruction of the core, core support structures, or RPV internal structures which would impair core cooling do not occur.

3 Reactivity control systems shall be designed to have two independent systems at least to bring the core subcritical from a hot standby or hot operation state, and also maintain subcriticality in a high temperature state.

4 Among the independent systems constituting the reactivity control systems, at least one system shall be designed to bring the core subcritical in a high temperature state during normal operation or abnormal transients during operation without exceeding allowable design limit of fuels, and also maintain subcriticality after a transient in a high temperature state has been terminated until Xenon decay allows addition of reactivity.

5 Among the independent systems constituting the reactivity control systems, at least one system shall be designed to bring the core subcritical while in a low temperature state, as well as maintain subcriticality in a low temperature state.

6 Among the independent systems constituting the reactivity control systems, at least one system shall be designed to bring the core subcritical during design basis accident. Among the independent systems constituting the reactivity control systems, at least one system shall be designed to maintain core subcriticality.

(Reactor shutdown systems)

7 Reactor shutdown systems by way of control rods shall be designed to bring the core subcritical in a high temperature state, even if a single control rod with the highest reactivity worth (or a cluster of control rods when the concerned control rod belongs to the same hydraulic control unit) has been completely withdrawn out of the core and cannot be inserted. Reactor shutdown systems consisting of control

rods shall be designed to bring the core subcritical in a low temperature state, in combination with the reactivity control systems if necessary, even if a single control rod with the highest reactivity worth (a cluster of control rods when the concerned control rod belongs to the same hydraulic control unit) has been completely withdrawn out of the core and cannot be inserted.

※The testability shall be summarized in the common matters.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 14, 15, 16, 17, 18)

[Requirement Details]

A. Evaluation of the "control rod maximum reactivity worth" may take into consideration the effects of equipment installed to control reactivity value, such as those limiting the extent of control rod insertion or positions depending on reactor operation status.

B. "Foreseeable reactivity insertion events" refer to the events where abnormal reactivity is introduced into the reactor. These are as stipulated in the "Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities " and "Guideline for Evaluation of Reactivity Insertion Events at Light Water Nuclear Power Reactor Facilities"

C. "Maintain subcriticality in a high temperature state" refers to maintaining subcriticality in the period after a transient has been terminated until reactivity is added due to Xenon decay. The actions of other systems may be relied upon or expected when maintaining subcriticality for greater lengths of time after this period.

D. "Ability of bring the core subcritical in a low temperature state, and maintain subcriticality in a low temperature state" refers to compensating for the reactivity added due to Xenon decay or reactor coolant temperature changes during high temperature subcriticality, while also achieving and maintaining low temperature subcriticality.

E. The control rod systems and soluble poison systems currently used in light water reactors (standby liquid control system for BWR, standby boric acid injection systems as part of chemical and volume control system for PWR) can be considered to be reactivity control systems which satisfy Item 3 above when looking at their functions.

F. The abilities of reactivity control systems during design basis accident may take the contributions of systems which possess reactor shutdown ability into consideration if their operation can be expected. An example would be the reactivity control systems working in combination with the emergency core cooling system during PWR main steam pipe rupture to induce subcriticality in the core and maintain core subcriticality.

G. Regarding the equipment included in the reactor shutdown systems or reactivity control systems, the control rod/chemical and volume control systems are both included in the reactor shutdown systems or reactivity control systems for PWR, while the control rod and SLC systems are both included in the reactor shutdown systems or reactivity control systems for BWR. The reactor recirculation flow control system is included in the reactivity control systems.

3. Individual systems within the reactor facility

(3) Reactor coolant pressure boundaries

[Basic requirements]

1. Reactor coolant pressure boundaries shall be designed to ensure its integrity during normal operation, abnormal transients during operation, or DBA.
2. Piping systems connected to the reactor coolant system shall be designed to have isolation valves in principle.
3. Reactor coolant pressure boundaries shall be designed so as not to show any brittle behavior during normal operation, abnormal transients during operation, or DBA, and not to cause sudden propagative fractures.
4. Reactor coolant pressure boundaries shall be designed to ensure that, in the event of reactor coolant leakage from the reactor coolant pressure boundary, leak can be swiftly and accurately detected.
   ※The reliability and testability are to be summarized in the common matters.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 19, 20, 21, 22)

[Requirement Details]

A. The reactor coolant pressure boundary refers to the equipment or piping within the scope covered below.

(a) RPV and associated components (components directly attached to the RPV, CRDM housing, etc.).

(b) Equipment and piping comprising the reactor cooling system. For PWR, this refers to the primary coolant pump, steam generator channel head, tubesheet and tubes, pressurizer, primary cooling system pipes, and valves. For BWR, the scope of equipment covered includes the main steam pipes and feedwater pipes up to the secondary isolation valves as seen from the reactor side.

(c) Connection piping

   i) Including the pipes that are equipped with valves which are normally open and closed in case of an accident up to the secondary isolation valves as seen from the reactor side.

   ii) Including the pipes that are equipped with valves which are normally closed and closed in case of an accident up to the secondary isolation valve as seen from the reactor side.

iii) as Also including the pipes of the emergency core cooling system which are equipped with valves that are normally closed in and opened when reactor coolant is lost as stipulated in i) above.

iv) "Isolation valves" above refers to the automatic isolation valves, check valves, normally locked shut-off valves, and remote control shut-off valves.

B. "Design which ensures integrity" refers to designs where the functions of the reactivity control systems, reactor cooling systems, instrumentation and control systems, safety protection systems, and safety valves are able to prevent sudden cooling/heating or abnormal pressure increase of the reactor coolant pressure boundary. The reactor coolant pressure boundary shall be designed to ensure that it can sufficiently withstand such temperatures or pressure changes, and incorporate design considerations to minimize abnormal reactor coolant leakage or damage to the reactor coolant pressure boundary.

C. "Design that is provided with isolation valves in principle" refers to the design where sufficient consideration is given to the objective of use and conditions of piping systems during normal operation, and appropriate isolation valves have been installed. The purpose thereof is to terminate the loss of reactor coolant in the event that abnormal leakage takes place at piping systems which comprise the reactor coolant pressure boundary, and specifically occurs between the piping systems which are connected to the reactor cooling system and the piping systems are not connected. The phrase "as a general rule" here refers to piping where measurement or sampling vital to reactor safety are performed, and assumes that the leakage which takes place at these pipes is so minor as to be sufficiently allowable. Piping which does not comprise the reactor coolant pressure boundary shall not have isolation valves installed.

D. Specific methods of the evaluation shall be as per the "Guideline for Evaluation of Reactivity Insertion Events at Light Water Nuclear Power Reactor Facilities.
(Determined by Nuclear Safety Commission of Japan on Jan. 19, 1984, partially revised on Aug. 30, 1990)"

3. Individual systems within the reactor facility

(4) Reactor cooling system

① Reactor coolant makeup system

[Basic requirements]

The reactor coolant makeup system shall be designed to have the ability to feed coolant at an appropriate flow rate so that the reactor coolant inventory can be restored even in the event of a minor reactor coolant leakage.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 23)

[Requirement Details]

A. The "reactor coolant makeup system" refers to the system which feeds the reactor cooling system with reactor coolant (the CRD hydraulic system and reactor isolation cooling system (excluding feedwater system) for BWR, and the systems feeding coolant by utilizing charge pumps for PWR).

B. "Minor reactor coolant leakage" refers to the reactor coolant leakage from cracking and form seals of valves and pumps constituting the reactor coolant pressure boundary.

3. Individual systems within the reactor facility

(4) Reactor cooling system

② RHR system

[Basic requirements]

1. The RHR system shall be designed to have the ability to remove fissile product decay heat and other residual heat from the core to prevent the allowable design limit of fuel and reactor coolant pressure boundary design conditions being exceeded during reactor shutdown.
2. The RHR system shall be designed to achieve its safety function in the event that external power sources are not available, in addition to an assumed single failure of the equipment which constitutes the RHR system.
   ※The reliability and testability are to be summarized in the common matters.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 24)

[Requirement Details]

A. The "RHR system" refers to the system designed to remove residual heat in the event that heat removal by the main condenser fails (reactor core isolation cooling system, RHR system, HPCS system, and automatic depressurization system for BWR; steam generator, main steam relief valves, main steam safety valves, auxiliary feedwater equipment, and residual heat removal system for PWR).In addition, systems are in place to depressurize the reactor cooling system for BWR (main steam safety relief valves) and PWR (pressurizer relief valves).

B. "Other residual heat" refers to the heat stored within areas such as the core, equipment, and materials such as the reactor cooling system, reactor coolant, and secondary coolant (for PWR) during normal operation.

3. Individual systems within the reactor facility
(4) Reactor cooling system
③ Emergency core cooling system

[Basic requirements]

1. The emergency core cooling system shall be designed to prevent extensive fuel damage in the event LOCA caused by assumed piping rupture and other events, and to minimize the interaction between fuel cladding metal and water.
2. The emergency core cooling system shall be designed to achieve its safety function in the event external power supplies are not available in addition to an assumed single failure of equipment which constitutes the system.
※The reliability and testability are to be summarized in the common matters.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 25)

[Requirement Details]

A. Specific methods of the evaluation shall meet the "Guideline for Evaluation of Performance of Emergency Core Cooling System of Light Water Nuclear Power Reactor Facilities (determined by Nuclear Safety Commission of Japan on Jul. 20, 1981, partially revised on Jun. 11, 1992)."

B. "assumed piping rupture" refers to "assumed LOCA accidents" described in the "Guideline for Evaluation of Performance of Emergency Core Cooling System of Light Water Nuclear Power Reactor Facilities"

C. "Piping ruptures etc." shall include accidents where LOCA is caused by e.g. stuck open of relief valves which doesn't accompany actual physical break.

D. Decisions regarding "sufficiently minor amount" shall be made according to the "Guideline for Evaluation of Performance of Emergency Core Cooling System of Light Water Nuclear Power Reactor Facilities ".

3. Individual systems within the reactor facility

(4) Reactor cooling system

④System to transfer heat to the ultimate heat sink

[Basic requirements]

1. The system to transfer heat to the ultimate heat sink shall be designed to transfer heat generated or accumulated in structures, systems or equipment having safety functions of particular importance to the ultimate heat sink.
2. The system to transfer heat to the ultimate heat sink shall be designed to achieve its safety functions event if external power sources are not available in addition to an assumed single failure of equipment which constitutes the system.
3. The system to transfer heat to the ultimate heat sink shall be designed taking into account the physical protection against design basis tsunamis, flooding, missiles, and all other external human events.

   ※The reliability and testability are to be summarized in the common matters.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 26)

[Requirement Details]

A. "Ultimate heat sink" refers to the seas, rivers, ponds, lakes, and atmosphere.

B. "System to transfer heat to the ultimate heat sink" refers to the system to transfer heat from the emergency core cooling system and RHR system (e.g., reactor auxiliary component cooling equipment, reactor auxiliary component cooling seawater equipment) to the ultimate heat sink.

3. Individual systems within the reactor facility

(4) Reactor cooling system

⑤Steam turbines

[Basic requirements]

1. Steam turbines and their associated components shall be designed so that they will not cause any adverse effect on the safety of reactor facilities.

2. Steam turbines and their associated components shall be designed to monitor parameters necessary to prevent reactor facility safety from being impaired by damage to turbines.

(※Equipment requiring establishment of new standards due to unification with Electricity Business Act)

[Requirement Details]

A. "Design not causing any adverse effect on the safety of reactor facilities" refers to the use of materials with resistance against chemical and physical effects under assumed environmental conditions, and to the sufficient structural strength including countermeasures against steam turbine vibration and over speed to ensure the safety of the reactor facility is not affected in the event of damage to steam turbines.

3. Individual systems within the reactor facility

(5) Reactor containment vessel

① Reactor containment vessel

[Basic requirements]

(Functions of the reactor containment vessel)

1 The reactor containment vessel shall be designed to withstand loads(pressure, temperature, dynamic load) arising from an anticipated event as well as relevant seismic loads, and to function in combination with an appropriate isolation function to keep the amount of leakage within a specified limit.

※The reliability and testability are to be summarized in the common matters.

(Preventing damage to the reactor containment vessel boundary)

2 The reactor containment vessel boundary shall be designed not to exhibit brittle behavior and develop any quickly propagative failure during normal operation, maintenance, testing, abnormal transient and design basis accident.

(Isolation function of the reactor containment vessel)

3 The pipes that penetrate through the walls of the reactor containment vessel shall be provided with containment isolation valves except for those pipes used for measurement or sampling important to the reactor safety and CRDM hydraulic pipes for which leakage through the pipe walls is small enough to be allowable.

4 The containment isolation valves to be installed on main pipe systems shall be designed to automatically and reliably close in the event of a design basis accident which requires the secured isolation function except for the pipes in the systems needed to bring the accident under control.

※The reliability and testability are to be summarized in the common matters.

(Containment isolation valves)

5 Containment isolation valves shall be installed close to the reactor containment vessel.

6 Containment isolation valves shall be installed in the following manners:

  a For the pipes that either open inside of the reactor containment vessel or connected to the reactor coolant pressure boundary, and are not closed on the outside of the reactor containment vessel, install one valve on the inside and another valve on the outside of the reactor containment vessel. If it is difficult to install isolation valves for physical reasons or due to environmental conditions, two isolation valves may be installed either on the inside or the outside of the reactor containment vessel provided that such installation is shown to be valid.

b For the pipes other than those described in the preceding paragraph, that are closed either on the inside or the outside of the reactor containment vessel, install one isolation valve on the outside of the reactor containment vessel. Alternatively, one isolation valve may be installed on the inside provided that such installation is shown to be valid.

c Notwithstanding the provisions of the two preceding paragraphs, for the pipes that have a rupture disk, it is allowed to install one normally closed isolation valve, in addition to the rupture disk, either on the inside or the outside of the reactor containment vessel.

d Containment isolation valves shall not lose its isolation function even when their source of drive force is lost after their closure.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 28-31)

[Requirement details]

A "Anticipated event" refers to an event anticipated for determining the validity of reactor containment vessel design. It encompasses the most severe conditions against relevant parameters out of evaluated parameters that could hinder the assurance of containment vessel functions such as pressure / temperature increase, dynamic load development, combustible gas generation, and concentration of radioactive materials. Specific details are defined in the Regulatory Guides for Reviewing Safety Evaluation of Light Water Nuclear Power Reactor Facilities (adopted by the Nuclear Safety Commission on August 30, 1990 and partially revised on March 29, 2001).

B "Containment isolation valves" refer to automatic isolation valves (including check valves designed to deliver a sufficient isolation function at the time of a design basis accident), normally-locked shut-off valves and remote-controlled shut-off valves. "Check valves designed to deliver a sufficient isolation function at the time of a design basis accident" refer to check valves designed to sustain their isolation function by way of gravitational force, etc., even in the loss of all counter pressure to the applicable check valve following damage to the applicable pipe system penetrating through containment vessel walls either on the inside or the outside of the primary containment vessel.

C "Main pipe systems" refer to pipe systems that must have containment isolation valves installed and could cause an unallowable level of leakage from the containment vessel if left in the state of normal operation, excluding those designed to have containment isolation valves closed in high temperature operation.

D "Function to automatically and reliably close" refers to the function of automatically closing in response to a reactor containment vessel isolation signal from the safety protection system, and reducing the leakage of radioactive materials from the reactor containment vessel in combination with isolation barriers other than containment isolation valves even in the event of a single failure when off-site power is not available.

E "Except for the pipes in the systems needed to bring the accident under control" refers to the exclusion of the pipes in the Emergency Core Cooling System, etc. that do not have to be closed in response to an automatic isolation signal so as not to compromise the safety functions expected of the applicable system. However, these pipes must not cause the loss of the reactor containment vessel's isolation function.

F The containment isolation valves that are automatically closed shall take into account the cancellation of isolation to allow necessary actions to be taken after an accident.

G "Pipes that are not closed on the outside of the reactor containment vessel" refer to the pipes that could form a channel for discharging an unallowable level of radioactive materials from the reactor containment vessel atmosphere to the outside if not isolated, in the event of a design basis accident.

H "Installation of a rupture disk" can occur only when it is shown not to adversely affect the safety functions of sever accident management equipment, which is to be separately installed. If installed, a rupture disk may be opened at a pressure setting sufficiently below the containment vessel's design pressure.

I Specific evaluation is performed in accordance with the "Regulatory Guide for Evaluating Dynamic Load on BWR Mark II Containment Pressure Suppression Systems (adopted by the Nuclear Safety Commission on July 20, 1981 and partially revised on August 30, 2000).

3. Individual systems within the reactor facility

(5) Reactor containment vessel

② Reactor containment vessel heat removal systems, systems controlling the containment atmosphere

[Basic requirements]

(Reactor containment vessel heat removal system)

1 The reactor containment vessel heat removal system shall be designed to have sufficient functionality for reducing the pressure and temperature inside the reactor containment vessel, generated from the energy released in the event of a design basis accident.

2 The reactor containment vessel heat removal system shall be designed to achieve its safety in the event that the external power sources are not available in addition to an assumed single failure of equipment constituting the system.

※The reliability and testability are to be summarized in the common matters.

(Systems controlling the containment atmosphere)

3 The containment atmosphere purifier system must be designed to have the function of lowering the concentration of radioactive materials, released into the atmosphere in the event of a design basis accident.

4 The flammability control system shall be designed to control the concentration of hydrogen or oxygen inside the reactor containment vessel, generated in the event of a design basis accident, in order to maintain the integrity of the containment facility.

5 The containment atmosphere control system shall be designed to achieve its safety in the event that the external power sources are not available in addition to an assumed single failure of equipment constituting the system.

※The reliability and testability are to be summarized in the common matters.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 32-33)

[Requirement details]

A "Reactor containment vessel heat removal system" refers to the system that sufficiently reduces the pressure and temperature inside the reactor containment vessel in the event of a design basis accident. This includes the Containment Spray System and its heat removal systems.

B “Systems controlling the containment atmosphere” refer to the containment atmosphere purifier system and flammability control system.

C “Containment atmosphere purifier system” refer to the emergency Gas Treatment System, Filtration Recirculation and Ventilation System, Containment Spray System, etc. in a BWR and the Annulus Air Recirculation System, Containment Spray System, etc. in a PWR.

D “Controlling the concentration of hydrogen or oxygen” refers to keeping the atmosphere inside the reactor containment vessel inert or keeping the concentration of hydrogen or oxygen below the combustible limit by means of hydrogen recombiners or other methods as necessary.

3. Individual systems within the reactor facility

(6) Instrumentation and control system

① Instrumentation and control system

[Basic requirements]

1 The Instrumentation and control system shall be designed to fulfill the following requirements during normal operation and abnormal transients during operation:

  a All the parameters required for securing the integrity of the reactor core, reactor coolant pressure boundary, containment vessel boundary, fuel storage facilities and associated systems, shall be maintained and controlled within the assumed range of fluctuation.

  b The parameters listed in the preceding paragraph shall be monitored within the assumed range of fluctuation to facilitate necessary response.

2 The instrumentation and control system shall be designed to fulfill the following requirements in the event of a design basis accident:

  a All the parameters required for identifying and countering accident conditions shall be monitored in an environment anticipated in an accident over a sufficient range and period.

  b The reactor's shutdown state and core cooling state shall be monitored or estimated based on at least two of these parameters.

  c Readings of required parameters must be kept in records and archives.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 47)

[Requirement details]

A "All the parameters required for securing the integrity" refer to in-core neutron flux, neutron flux distribution, reactor coolant inventory, reactor coolant system's pressure / temperature / flow volume, reactor coolant quality, pressure / temperature / atmospheric gas concentration in the primary containment vessel, and pool level / temperature at fuel storage facilities.

B "All the parameters required for identifying and countering accident conditions" refer to the pressure, temperature, hydrogen gas concentration, radioactive material concentration, etc. in the atmosphere inside the containment vessel.

C "Required parameters" in Paragraph 2 Item 3 refer to items listed in Article 20 Paragraph 1 Item 1 and from Item 3 to Item 14 of the government directive defining the technological standards concerning nuclear facilities for power generation.

D "Records and archives" refer to the state whereby necessary parameters "can be referenced" after an event.

E The requirements for the measurement and control systems in the event of a design basis accident are as defined in the Regulatory Guide for Reviewing Radiation Measurement during Accidents at Light Water Nuclear Power Reactor Facilities (adopted by the Nuclear Safety Commission on July 23, 1981 and partially revised on September 19, 2006).

3. Individual systems within the reactor facility

(6) Instrumentation and control systems

② Safety protection systems

[Basic requirements]

1 The safety protection systems shall be designed with redundancy so that the safety protection functionality is maintained during normal operation, abnormal transient during operation and a design basis accident even if a single device or channel comprising the applicable system is removed.

2 The safety protection systems shall be designed to have channels comprising the applicable system mutually separated to achieve each channel's independence, so that the safety protection function is maintained in normal operation, maintenance, testing, and abnormal transient during operation and a design basis accident.

3 The safety protection systems shall be designed to detect an abnormal status at the time of abnormal transient during operation, and automatically activating appropriate systems including the reactor shutdown systems, so as to keep the fuel within its allowable design limit.

4 The safety protection systems shall be designed to detect an abnormal status in the event of a design basis accident, and automatically actuate the reactor shutdown systems and other necessary systems including engineered safety facilities.

5 The safety protection systems shall be designed to bring the reactor facility to a safe state even in the loss of drive power source, system shutdown or any other adverse state.

6 The safety protection systems shall be designed to take into account unauthorized system breach from an external network and other cyber security issues.

7 The safety protection systems shall be designed to be functionally separated from the measurement and control systems, if they are partially shared, so that the measurement and control systems would not disrupt the functionality of the safety protection systems.

※The reliability and testability are to be summarized in the common matters.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 34-39)

[Requirement details]

A “Channels” refer to arrays of components (resister, condenser, transistor, switch, conductor, etc.) and modules (assemblies of components that are internally communicating) from a detector to the inlet of a logical circuit, required to generate a stand-alone signal for activating safety protection.

B “Having channels mutually separated” refers to a mechanism for preventing an adverse condition on one channel from inducing a similar adverse condition on another channel, or safeguarding a channel from influence that could hinder its safety functionality.

C An example of the safety protection systems’ function in abnormal operational transient is to detect the abnormal state and activate the reactor shutdown system and other relevant systems to initiate emergency shutdown, in order to prevent excessive reactor output and sudden output rise.

D “Loss of drive power source, system shutdown or any other adverse state” refers to the shutoff of a safety protection system’s logic circuit due to the air loss of instrumentation or some other cause. An adverse state includes environmental conditions, but the state that needs to be taken into account is determined on a case-by-case basis for individual designs.

E “To bring the reactor facility to a safe state” refers to the reactor facility settling and remaining in a safe condition even when a safety protection system fails, or the reactor facility maintaining a state with no safety issues even if a safety protection system fails.

F “Design that takes into account cyber security issues” refers to a design approach that prevents unauthorized actions and changes via physical hardware separation, functionality separation, and preventing computer viruses on the stages of system introduction, update, and testing.

G “Will not lose the functions of the safety protection systems” means, even if a single failure, erroneous operation or single removal from the service occurred in any component or channel of the connected instrumentation and control system, the portions of the safety protection system that immune to such failure/removal would satisfy the basic requirements 1 to 6 for the safety protection systems.

3. Individual systems within the reactor facility

(6) Instrumentation and control systems

③Control room, etc. (excluding habitability)

[Basic requirements]

(Control room)

1 The control room shall be designed to monitor the operation status and main parameters of the reactor and its associated facilities.

2 The control room shall be designed to be able to identify the status outside the reactor facility.

3 The control room shall be designed to enable quick manual operation for assuring safety if such operations are required.

(Reactor shutdown function from outside the control room)

4 The reactor facility shall be designed to have the following functions so that the reactor can be shut down from an appropriate location outside the control room:

  a Being able to bring the reactor to a rapid hot shutdown with necessary instrumentation and control systems to maintain the reactor facility in a safe state

  b Being able to bring the reactor to a cold shutdown by way of following appropriate procedures

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities41-42)

[Requirement details]

A "Able to monitor main parameters" refers to the ability to monitor, from the control room, the parameters that are subject to monitoring as [Basic requirements] for the instrumentation and control systems and need to be monitored continuously.

B "Designed to be able to identify the status outside the reactor facility" refers to the ability to identify, from the control room, any natural phenomena, etc. that could affect the reactor facility.

C "Quick manual operation" refers to the operation for shutting down a reactor and ensuring the cool-down of the reactor after its shutdown.

D "The reactor can be shut down from an appropriate location outside the control room" means that some countermeasures are in place when personnel cannot approach the control room for some reason.

E "Bringing the reactor to a rapid hot shutdown" refers to the ability to shut down a reactor immediately, remove residual heat, and safely maintain the state of hot shutdown.

3. Individual systems within the reactor facility

(6) Instrumentation and control systems

④ Control room, etc. (habitability)

[Basic requirements]

The control room shall be designed to have a fire protection mechanism, shield facilities so that personnel can access or stay in the control room even in the event of a design basis accident to carry out accident response operations, and a ventilation system to provide adequate protection against toxic gas and radioactive materials that could be discharged in a fire or an accident.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 43)

[Requirement details]

A "So that personnel can access or stay in the control room" means having an access route secured for personnel in charge of accident response operations to reach the control room when an accident occurs, facilitating personnel's stay in the control room for an appropriate period of time, and enabling the implementation of exposure protection measures following accident response operations once the radiation level attenuates after a certain period of time, so as to allow replacement personnel to approach the control room.

3. Individual systems within the reactor facility

(6) Instrumentation and control systems

⑤ Emergency response center

[Basic requirements]

The reactor facility shall be designed so that the emergency response center for issuing necessary orders/commands on countermeasures in the event of a design basis accident can be installed on site.

(See the requirements for severe accidents for specific requirements.)

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 44)

3.Individual systems within the reactor facility

(7) Electric systems

① Basic requirements concerning the safety design of electric systems for the reactor facility

[Basic requirements]

1 The structures, systems and components with safety functions of particular importance shall be designed to be able to receive power from both an off-site power source (electric system) and emergency power source on site, with the power supplies designed to secure and sustain a sufficiently high level of reliability when they need electricity to fulfill their functions. Also, in order to ensure that they would not lose required power supplies as a result of a fault in electric system equipment such as a main generator, off-site power system, emergency on-site power system, etc. or disruption to the off-site power (electric system), these structures, systems and components should be designed to detect an abnormal status and prevent its expansion or propagation.

2 The off-site power system shall be connected to the reactor facility's electric system via at least 2 transmission lines, which are connected to at least 2 separate, independent substations or switchyards, wherein at least one of the transmission lines must be physically separate from other lines. A nuclear power station that has multiple reactor facilities shall be designed so that the reactor facilities would not lose off-site power simultaneously even if any two of the transmission lines are lost.

3 The emergency on-site power systems shall be designed to have sufficient capacity and functionality to ensure the following matters even if any one of the systems is lost:

a. Shutting down and cooling the reactor in the event of an abnormal transient during operation without exceeding the allowable design limit for fuel or design conditions for reactor coolant pressure boundary
b. Cooling the core in the event of a design basis accident such as the loss of reactor coolant, ensuring the integrity of the reactor containment vessel and securing the safety functions of other relevant systems and equipment
c. Ensuring that emergency on-site power facilities do not rely on shared use between at least two reactor facilities

4 Emergency on-site AC power facilities shall be designed to continuously supply required electricity in the event of the loss of off-site power for a cerain period of time.

※The reliability and testability are to be summarized in the common matters.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 48)

[Requirement details]

A "Power supplies designed to secure and sustain a sufficiently high level of reliability" means that electric systems' bus is configured in view of system separation so as not to compromise the redundancy for the structures, systems and components with safety functions of particular importance, and that individual components comprising the electric systems are highly reliable to make it easy to switch bus, for example, to receive power from the emergency on-site power system.

B "Off-site power source (electric system)" refers to an electric system outside the power station's switchyard and does not include the applicable reactor facility's main generator or the main generators of other reactor facilities within the same nuclear power station.

C "Designed to detect an abnormal status and prevent its expansion or propagation" refers to a design that detects the short circuiting / grounding of devices in the electric systems as well as low voltage / over voltage of bus and uses a circuit breaker, etc. to isolate the fault location to localize the impact or limit its impact on other safety functions.

D "Off-site power system" refers to the off-site power source (electric system) and a series of components for supplying electricity from the applicable reactor facility's main generator to the reactor facility.

E "At least 2 separate, independent substations or switchyards" refer to at least 2 different substations or switchyards that are connected to a single substation or a switchyard upstream of the electric system, so that the shutdown of one of these substations or switchyards would not entirely halt power transmission on the transmission lines connected to the nuclear power station.

F "At least 2 transmission lines" in the off-site power system shall be achieved by combining lines that can send and receive power and lines dedicated to receiving power, provided by establishing at least two lines connecting the electric system to emergency on-site power distribution facilities for receiving power from an off-site power source. "Physically separate" refers to the mechanism of preventing simultaneous transmission cut by laying transmission lines through multiple transmission line towers to avoid the situation in that the collapse of a transmission tower that carries all the transmission lines would result in simultaneous transmission cut.

G Switchyards and power transmission / receiving facilities at a nuclear power station must be built on the ground foundation that has sufficient support to resist uneven subsidence or inclination, and use insulators, circuit breakers, etc. that are highly earthquake resistant. These facilities shall be isolated or protected from the impact of possible tsunami, with considerations given to salt damage.

H In the case of a nuclear power station that has multiple reactor facilities, the off-site power system shall be connected to the electric systems of each of the reactor facilities via at least three transmission lines in a tie-line design so that the multiple reactor facilities would not experience the loss of off-site power simultaneously even if any two of those lines fail.

I “Emergency on-site power system” refers to facilities (emergency bus switch gear, cables, etc.) supplying power to the equipment with safety functions of particular importance, including emergency on-site power source facilities (e.g., emergency diesel generators, batteries) and engineered safety features.

J “Loss of off-site power for a certain period of time” for emergency on-site AC power supplies (emergency diesel generators, etc.) refers to the assumption of a 7-day loss of off-site power, during which the emergency diesel generators, etc. must operate continuously to supply electricity. The facilities for storing the fuel for the emergency diesel generators, etc. (seismic S class) shall be designed to have a capacity to store enough fuel required to operate them for 7 consecutive days on site.

K “Safety function of particular importance” is to be defined separately based on the “Review Guide for Classification of Safety Function Importance in Light Water Nuclear Power Reactor Facilities”.

3. Individual systems within the reactor facility

(7) Electric systems

②Basic requirements concerning electrical facilities at a nuclear power station

[Basic requirements]

1 Of electrical facilities at a nuclear power station, switchyards, large transformers, main generators, etc. shall be designed considering electric circuits' electric insulation, open circuit prevention, grounding measures, earthing measures, over-current measures, heat resistance and mechanical impact caused by short-circuit current.

2 Compressed air equipment used in circuit breakers and gas-insulated circuit breakers shall be designed to facilitate the monitoring and control of pressure used, sufficiently withstand the pressure applied and have anti-corrosive property.

3 The rotating part of the main generators shall be designed to have sufficient mechanical strength. Of the main generators, those in the hydrogen cooling system must be designed to prevent hydrogen leakage or air entry, detect any hydrogen leakage, sound an alarm in the case of leakage, stop the leakage and release the leaked hydrogen outside.

4 Lightning arresters shall be installed so that lightning would not damage electric circuits' electric facilities.

(New)

[Requirement Details]

A. "Electrical facilities at a nuclear power station" refer to electrical facilities for power generation purposes, driven with nuclear energy (according to Article 106 of the Electricity Business Act), and represent equipment at a switchyard within a nuclear power station (circuit breakers, line switches, lightning arresters, insulators, etc.), large transformers for sending / receiving electricity to/from off-site power sources, main generators, electric circuits that mutually connect these electrical facilities, electric circuits connecting to off-site power sources, and security communications system.

3.Individual systems within the reactor facility

(8) Design considerations against station blackout

[Basic requirements]

Reactor facilities shall be designed to shut down the reactor safely, cool it down after shutdown, and secure the integrity of the reactor containment vessel in the event of station blackout that lasts for a certain period of time.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 27)

[Requirement Details]

A To be prepared for station blackout (combination of the loss of off-site power and the loss of emergency on-site AC drive power), emergency on-site DC power supply shall be designed to serve as the source of power required for safe reactor shutdown and subsequent cooling for a certain period of time.

B "Designed to shut down the reactor safely, cool it down after shutdown, and secure the integrity of the reactor containment vessel" refers to the design of the emergency on-site DC power supply to have sufficient capacity to handle the power load assigned to these facilities to maintain the functions associated with reactor shutdown, subsequent reactor cooling, and securing the integrity of the containment vessel.

3.Individual systems of a reactor facility

(9) Radioactive waste treatment facilities

[Basic requirements]

(Treatment facilities for radioactive gaseous waste and radioactive liquid waste)

1. The treatment facilities for radioactive gaseous waste and radioactive liquid waste, generated during the operation of a reactor facility, shall be designed to sufficiently lower the concentration and volume of radioactive materials discharged into the surrounding environment.

2. The treatment facilities for radioactive liquid waste and associated facilities shall be designed to prevent the leakage of liquid radioactive materials from these facilities or their discharge to the outside of the plant site.

(Treatment and storage facilities for radioactive solid waste)

3. The treatment facilities for radioactive solid waste, generated from a reactor facility, shall be designed with considerations to prevent the possible dispersion of radioactive materials.

4. The storage facilities for radioactive solid waste shall have sufficient capacity for storing radioactive solid waste, generated from a reactor facility, and shall be designed with considerations to prevent the spread of contamination attributable to waste.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 52-55)

[Requirement Details]

(Treatment facilities for radioactive gaseous waste and radioactive liquid waste)

A. "Design to sufficiently lower the concentration and volume of radioactive materials discharged" refers to the use of filtration, retention, attenuation, management, etc. for gaseous waste treatment facilities, and filtration, evaporation, ion exchange, retention, attenuation, management, etc. for liquid waste treatment facilities.

B. "Sufficiently lowering" refers to lowering radiation dose to the target level defined in the "Annual Dose Target for the public in the Vicinity of Light Water Power Reactor Facilities" (adopted by the Nuclear Safety Commission of Japan on May 13,

1975) <Note: 50 μSv/year> under the concept of ALARA (As Low As Reasonably Achievable).

C. The evaluation of abovementioned dose target shall be conducted according to the "Regulatory Guide on the Assessment of Exposure Dose of the Public in Safety Examination of Power Generating Light Water Reactors" (adopted by the Nuclear Safety Commission of Japan on September 28, 1976).

D. The "treatment facilities for radioactive liquid waste" refer to facilities that separate and collect radioactive liquid waste as well as radioactive waste in the liquid state containing sludge and other solid matter, and apply filtration, evaporation, ion exchange, retention, attenuation, management, and other appropriate processing according to the properties of the waste liquid.

E. "Associated facilities" refer to buildings or areas for housing treatment facilities.

F. "Design to prevent the leakage of radioactive liquid materials from these facilities or their discharge to the outside of the plant site" is determined according to "Considerations and Basic Approach to the Safety Review of Radioactive Liquid Waste Processing Facilities" (adopted by the Nuclear Safety Commission of Japan on September 28, 1981).

(Treatment and storage facilities for radioactive solid waste)

G. “Dispersion of radioactive materials” includes the spread of waste through the processes of crushing, compaction, incineration, and solidification.

H. The storage facilities for radioactive solid waste shall be capable of storing and managing the volume of radioactive solid waste generated at or transported into the reactor facility in the future.

3. Individual systems of a reactor facility

(10) Fuel handling systems

[Basic requirements]

1. The new and spent fuel storage and handling facilities shall be designed to meet the following requirements:
  a. Storage facilities shall have appropriate containment systems and air purification systems.
  b. Storage facilities shall have an appropriate storage capacity.
  c. Handling facilities shall be able to prevent the fall of fuel assemblies during transfer operation.

  <u>※ The testability is to be summarized in the common matters.</u>

2. The storage facilities (excluding the cases where dry storage casks are used) and handling facilities for spent fuel shall be designed to meet the following requirements in addition to the requirements listed in the preceding paragraph:
  a. The facilities shall have appropriate shields for radiological protection.
  b. The facilities shall have the systems for removing decay heat from spent fuel and transfer heat to the ultimate heat sink, as well as associated systems for purification.
  c. The facilities shall be able to prevent significant reduction in coolant inventory and carry out appropriate leakage detection.
  d. The facilities shall be able to maintain their safety functions even in an anticipated event during the handling of fuel assemblies, such as dropping of the assemblies or falling of a heavy item.

3. The storage facilities for spent fuel (only for the cases where dry storage casks are used) shall be designed to meet all the following requirements in addition to the requirements listed in Paragraph 1 (excluding air purification systems if the lid for dry storage casks are not to be opened in the applicable facility, and if the containment of radioactive materials can be secured only with the use of the dry storage casks):
  a. The facilities shall have appropriate shields for radiological protection.
  b. The facilities shall be able to remove decay heat properly.
  c. The facilities shall be able to properly contain radioactive materials included in spent fuel, and monitor their functions appropriately.

4. The storage facilities and handling facilities for fuel shall be designed to prevent criticality.

5. The storage and handling facilities for new and spent fuel shall be designed to be able to measure the water level and temperature of spent fuel storage facilities and radiation level of the location of fuel handling, detect abnormal readings, and inform workers of abnormal readings reliably or address them automatically. Furthermore, they shall be designed to monitor multiple parameters even when off-site power is not available.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 49-51)

[Requirement Details]

A. “Dry storage casks” refer to containers in which spent fuel is loaded with inert gas for storage, consisting of the main cask unit, a (double) lid, a basket, etc.

B. “Informing workers of abnormal readings” means the readings can be monitored from the control room even if access to the location of fuel handling is restricted due to an abnormal state.

C. The design of dry storage casks is verified according to “Dry cask storage of spent fuel at nuclear power stations” (adopted by the Nuclear Safety Commission of Japan on August 27, 1992 and partially amended on September 29, 2001).

3.Individual systems of a reactor facility

(11) Radiation control

① Surrounded radiological protection facilities (during normal operation)

[Basic requirements]

The reactor facility shall be designed to sufficiently lower the space dose rate in surrounding areas during normal operation, caused by direct gamma rays and skyshine gamma rays.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 56)

[Requirement Details]

A. “Sufficiently lowering” means the facilities shall be designed and managed to keep the air kerma no more than 50 micro gray per annum based on the concept of ALARA and according to the “Dose Evaluation for Members of the Public during Safety Review of Light Water Nuclear Power Reactor Facilities” (adopted by the Nuclear Safety Commission of Japan on March 27, 1989). If the facilities are designed and managed accordingly, there is no need for dose evaluation.

3. Individual systems of a reactor facility

(11) Radiation control

② Protection and management facilities

[Basic requirements]

(Radiological protection for radiation workers)

1. The reactor facility shall be designed with radiological protection measures so as to sufficiently reduce radiation dose in areas accessed by radiation workers.

2. The reactor facility shall be designed with radiological protection measures so that radiation workers can conduct necessary operations during an abnormal operational transient and a design basis accident.

(Radiation control for radiation workers)

3. The reactor facility shall be designed with radiation control facilities to protect radiation workers from radiation.

4. The aforementioned radiation control facilities shall be designed to display necessary information in the control room or other appropriate locations.

(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 57 & 58)

[Requirement Details]

(Radiological protection for radiation workers)

A. "Design with radiological protection measures" in Paragraph 1 refers to the design that incorporates relevant radiological protection measures such as shielding, equipment layout, remote control, prevention of radioactive material leakage, and ventilation under the concept of ALARA and in consideration for the workability of radiation workers.

(Radiation control for radiation workers)

B. "Radiation control facilities" refer to the facilities for radiation workers' access control, contamination management and decontamination, established for monitoring and managing radiation exposure.

C. "Displaying necessary information in the control room or other appropriate locations" refers to the ability to display the space dose rate, measured with area radiation monitors for radiation control, in the control room, as well as the space dose rate in controlled areas, in-air concentration of radiation materials and surface concentration of contamination of radioactive materials on floors at other appropriate locations.

3. Individual systems of a reactor facility

(11) Radiation control

③ Monitoring facilities

[Basic requirements]
The reactor facility shall be designed to appropriately measure and monitor the discharged radiation and radioactive materials during normal operation, abnormal operational transient and design basis accident, and display necessary information in the control room or other suitable locations.
(Corresponds to Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities Guide 59)

[Requirement Details]

A. "Appropriately gauging and monitoring" means the reactor facility can measure and monitor the concentration of radioactive materials and space dose rate through sampling and radiation monitors for the atmosphere inside the primary containment vessel and in areas under surveillance surrounding the reactor facility and can also gauge and monitor relevant locations such as radiation sources, discharge points, around the nuclear power station, and at anticipated discharge channels of radioactive materials, so as to facilitate quick response in the event of a design basis accident.

B. The gas and liquid waste discharged into the environment during normal operation should be measured and monitored according to the "Guidelines on the measurement of radioactive materials discharged from light water nuclear reactor facilities" (adopted by the Atomic Energy Commission on September 29, 1978).

C. The measurement and monitoring during a design basis accident should be conducted according to the Review Guide for Radiation Measurement During Accidents from Light Water Nuclear Power Reactor Facilities (adopted by the Atomic Energy Commission on July 23, 1981).

D. Monitoring posts shall be designed, if not connected to an emergency electrical power distribution system, to secure the functionality until the restoration of power supplies through the use of uninterruptible power supplies, etc. Also, the transmission system of monitoring posts shall be designed to have diversity.

(12) Miscellaneous

① Basic requirements on auxiliary boilers

[Basic requirements]

1. Auxiliary boilers shall have the capacity to supply required steam under anticipated usage conditions.
2. Auxiliary boilers must be designed to not compromise the safety of the reactor facility.

(※Electrical facilities that need to have new standards defined due to the incorporation of the Electricity Business Act)

[Requirement Details]

A. "Capacity to supply required steam" refers to the ability to supply enough steam for structures, systems, and components with safety functions.

B. "Design to not compromise the safety of the reactor facility" refers to design that would not affect the safety of the reactor facility even when the auxiliary boiler becomes damaged.

4. Safety evaluation

(1) Safety evaluation

[Basic requirements]

1. Analysis and assessment shall be carried out against abnormal operational transients and design basis accidents in order to confirm that the basic policies of safety design for a reactor facility meet the requirements defined in 1 – 3.

2. The aforementioned analysis and assessment concerning abnormal operational transient shall confirm the following requirements are met. [Judging criteria for the Evaluation Guideline 4.1]

a. The minimum critical power ratio or minimum critical output ratio shall exceed the allowable limit.
b. Fuel cladding shall not be mechanically damaged.
c. Fuel enthalpy shall not exceed the allowable limit.
d. The pressure applied to the reactor coolant pressure boundary shall be no more than 1.1 times the maximum service pressure.

3. The aforementioned analysis and assessment concerning a design basis accident shall confirm that the following requirements are met. [Judging criteria for the assessment guideline 4.2]

a. The reactor core can be sufficiently cooled down without resulting in significant damage.
b. Fuel enthalpy shall not exceed the limit.
c. The pressure applied to the reactor coolant pressure boundary shall be no more than 1.2 times the maximum service pressure.
d. The pressure and temperature at the containment vessel boundary shall not exceed the maximum service pressure and temperature.
e. The reactor facility shall not cause significant exposure risk to the general public in the surrounding areas

(Corresponds to Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities)

[Requirement Details]

A. "Analysis and assessment against abnormal operational transient and design basis accident" shall be performed based on the “Review Guidelines on the safety evaluation of light water nuclear reactor facilities” and the “Meteorological Guidelines on the safety analysis of light water nuclear reactor facilities” (adopted by the Nuclear Safety Commission of Japan on January 28, 1982).