(Provisional Translation)

April 3, 2013

Outline of New Regulatory Requirements (Design Basis)

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4. Safety Evaluation

(1) Safety Evaluation

1. General Rules

(1) Definitions of Terms

The definitions of terms as used in the draft shall be as stated under their respective numbers below (Corresponding to the definitions of terms in Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities):

(a) "Safety functions" refer to the functions of structures, systems, and components (hereafter referred to as “SSCs”) necessary to ensure the safety of nuclear reactor facilities, which are categorized as follows:

1) Functions that may cause, when lost, anticipated operational occurrences and design basis accidents in nuclear reactor facilities, potentially leading to undue radiation exposure of the public or site personnel.

2) Functions that prevent, in case of anticipated operational occurrences and design basis accidents in nuclear reactor facilities, the escalation of such conditions or put such conditions under control immediately, thereby preventing or mitigating potential undue radiation exposure of the public or site personnel.

(b) "Importance of safety functions" refers to the degrees of the importance of safety functions from the viewpoint of ensuring the safety of nuclear reactor facilities.

(c) "Normal operations" refer to the scheduled operations of nuclear reactor facilities including starting up, shutting down, power operation, hot stand-by and refueling that are performed under the operational conditions within specified limits.

(d) "Anticipated operational occurrences" refer to those conditions deviating from normal operation which are expected to occur once or several times during the operating life of nuclear reactor facilities by single component failures, single component malfunctions or single errors or by disturbances with a similar frequency of occurrence.

(e) " Design basis accidents" refer to those conditions beyond "anticipated operational occurrences", which have quite small frequency of occurrence and yet are postulated in the light of the safety design of nuclear reactor facilities.

(f) "Reactor containment boundary” refers to those provisions which are designed for and limited to such a range that they serve as a pressure barrier in case of the postulated events for reactor containment design and that they form a barrier to the release of radioactive materials to the environment.

(g) "Reactor coolant pressure boundary" refers to those provisions which are designed for and limited to such a range that they contain, during normal operation, the reactor coolant (primary coolant in case of a pressurized water reactor (PWR)) retaining the same pressure as the reactor and form a pressure barrier in case of anticipated operational occurrences and design basis accidents and which cause loss of reactor coolant when damaged.

(h) "Reactor coolant system" refers to those systems of reactor coolant which directly cool the reactor core during normal operation (the primary cooling system in a PWR; the reactor coolant recirculation system, the main steam system and the feed water system in a boiling water reactor (BWR).

(i) "Reactor cooling system" refers to those systems which are used to remove heat from the reactor during normal operation, anticipated operational occurrences or design basis accidents (reactor coolant system, systems for removing residual heat, emergency core cooling system, secondary cooling system (PWR) , systems for transporting heat to an ultimate heat sink, etc.).

(j) "Reactor shutdown system" refers to those provisions which are designed to make the reactor at or beyond criticality become subcritical by inserting negative reactivity, compensate for reactivity changes associated with the transition from hot shutdown to cold shutdown and maintain the reactor in the subcritical state.

(k) "Reactivity control system" refers to those provisions which are designed to control reactivity of the reactor, thereby regulating reactivity changes associated with variations in reactor power, burn up, fission products, etc.

(l) "Safety protection system" refers to those provisions which are designed to detect anticipated operational occurrences or design basis accidents of nuclear reactor facilities and directly initiate the actuation of the reactor shutdown systems, engineered safety features and other systems as required.

(m) "Engineered safety features" refer to those provisions which are designed to prevent or limit the possible significant release of radioactive materials to the environment after fuel damage, etc. resulting from damage or failure in nuclear reactor facilities.

(n) "Single failure" refers to the loss of intended safety functions due to a failure in a single component. Multiple failures due to secondary causes are included in this category. In this case, “secondary causes” refer to the causes which inevitably occur due to a single cause.

(o) "Active component" refers to a component that performs necessary functions actively depending on an external input, such as actuation signals or motive power.

(p) "Passive component" refer to equipment other than active component.

(q) "Redundancy" represents the existence of two or more systems or components with identical attributes to perform an identical function.

(r) "Diversity" represents the existence of two or more systems or components with different attributes to perform an identical function. In this case, “different attributes” means that the operating principle and others are different, so that the functions will not be impaired simultaneously due to common or subordinate causes. Furthermore, "common cause" refers to a cause that simultaneously affects two or more systems or components; such as influential factors induced by environmental temperature, humidity, pressure, radiation, etc. or those induced by electric power, air, oil, cooling water, etc. supplied to systems and components, or influences caused by earthquake, flooding and fires, etc.

(s) "Independence" means that two or more systems or components are free from simultaneous functional impediment due to common or subordinate causes under environmental or operational conditions considered in design.

(t) "Acceptable fuel design limits" refer to the limits within which fuel damage is tolerable in view of the safety and the reactor can continue its operation, related to the reactor design. In this case, "The reactor can continue its operation" does not necessarily mean that the reactor can be operated in the condition where the fuel is damaged and it includes the resumption of reactor operation after the repair of failures and if necessary, after the inspection of fuel and refueling.

1. General Rules

(2) Applied Codes and Standards

[Basic Requirements] The design, selection of materials, fabrication and inspection of SSCs with safety functions shall conform to those codes and standards which are recognized appropriate in the light of the importance of their safety functions. (Corresponding to Guideline 1 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A. The design, selection of materials, fabrication and inspection of SSCs with safety functions shall in general be subject to the codes and standards conformable to existing domestic (Japanese) laws and regulations.

   In case foreign codes and standards or non-ordinary codes or standards are applied, the background and justification for the intended application of such codes and standards and the comparison of such codes and standards with their Japanese counterparts need to be clarified.

B. The phrase, "... shall conform to those codes and standards", implies that it is necessary to identify the codes and standards on which the design, selection of materials, fabrication and inspection of SSCs in question shall be based.

2. General Technical Requirements for Nuclear Reactor Facilities

(1) Design Considerations against Natural Phenomena

[Basic Requirements]

(Earthquake and tsunami (including concomitant events)

1. SSCs with safety functions shall be assigned to appropriate seismic categories, with the importance of their safety functions and possible safety impacts of earthquake-induced functional loss taken into consideration, and be designed to sufficiently withstand appropriate design seismic forces.

(*The above requirement is based on Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities. However, the requirement for the design basis earthquake and tsunami (including concomitant events) is now being studied by an another review team, would be transposed by the study results.)

(Natural phenomena other than earthquakes)

2. SSCs with safety functions shall be so designed that the safety of the nuclear reactor facilities will not be impaired by other postulated natural phenomena than earthquake, tsunami and concomitant events. SSCs with safety functions of especially high importance shall be of the design that reflects appropriate safety considerations against the severest conditions of anticipated natural phenomena or appropriate combinations of natural forces and design basis accidents-induced loads.

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

[Detailed Notes on Requirements]

A. The interpretation of "... designed to sufficiently withstand appropriate design seismic force" is subject to "Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities". "(decision by Nuclear Safety Commission on September 19, 2006)

B. The phrase "... so designed that the safety of the nuclear reactor facilities will not be impaired by...natural phenomena" means that the SSCs involved shall achieve their safety functions under the environmental conditions caused by natural phenomena itself and the environmental conditions which may be subsequently caused at the facility, when they encounter natural phenomena to be considered in design individually or in combination.

C. "SSCs with safety functions of especially high importance" will be specified in consideration of "Regulatory Guide for Reviewing Classification of Importance of Safety Function of Light Water Nuclear Power Reactor Facilities"(decision by Nuclear Safety Commission on August 30, 1990).

D. "Anticipated natural phenomena" refer to on-site natural phenomena possible to occur including flood, wind (typhoon), tornado, freezing, rainfalls, snowing, lightning, landslide, volcanic effects, biological effects, forest fires, etc.

E. "The severest conditions" refer to the conditions assumed to be the severest according to the latest scientific and technological knowledge concerning the natural phenomena under consideration. Note that, in determining such conditions, other types of natural phenomena should be assumed to overlap as appropriate in consultation with past records, results of on-the-spot investigation, latest knowledge of related matters, and the like.

F. "appropriate combinations of natural forces and accident-induced loads" does not necessarily mean a simple addition of the natural forces considered as the severest to the maximum load on an accident. Rather, it refers to appropriate combinations considering causal correlations and their progress.

2. General Technical Requirements for Nuclear Reactor Facilities

(2) Design Considerations against External Man-Induced Events

[Basic Requirements]

(Contingencies)

1. SSCs with safety functions shall be so designed that the safety of the nuclear reactor facilities will not be impaired by postulated contingent external man-induced events.

(Illegal access by the third party and the like)

2. The nuclear reactor facilities shall be so designed that SSCs with safety functions are protected by appropriate means against any illegal access by the third party and the like.

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

[Detailed Notes on Requirements]

A. "contingent external man-induced events" are to be selected in consideration of the circumstances present at the site or in the surrounding areas, and refer to missiles (airplane crash, etc.), dam break, explosion, fires in neighboring factories and the like, toxic gas, collision of ships, electromagnetic interferences, and the like.

B. As for "airplane crash," evaluation for the necessity of protective design shall be conducted following "Evaluation of Probability of Plane Crash on the Commercial Power Reactor Facility" (2009/06/25 NISA No. 1) and the like, which the former Nuclear and Industrial Safety Agency established on July 30, 2002 and revised on June 30, 2009.

C. "Illegal access by the third party and the like" must include illegal relocation of nuclear materials and sabotage by the insider, introduction of explosive or harmful materials from outside the site by mail or the like, cyber terrorism.

2. General Technical Requirements for Nuclear Reactor Facilities

(3) Design Considerations against Internal Missiles

[Basic Requirements]

SSCs with safety functions shall be so designed that the safety of the nuclear reactor facilities will not be impaired by postulated missiles that may take place within the nuclear reactor facilities.

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

[Detailed Notes on Requirements]

A. The "internal missiles" are flying matters resulting from break of valves and pipes containing high energy fluid inside, damages of high-speed rotating components, gas explosions, a heavy components fall, etc. The design considerations shall take into account the secondary impacts of secondary missiles, fire, chemical reaction, electrical damage, pipe rupture and equipment breakdown that may result from the primary incidents.

B. As for evaluation of internally generated missiles, evaluation must be conducted following “Impact Assessment of the Turbine Missiles (Examination Committee on Reactor Safety of Atomic Energy Commission on July 20, 1977)” and the like.

2. General Technical Requirements for Nuclear Reactor Facilities

(4) Design Considerations for Internal Flooding

[Basic Requirements]

SSCs with safety functions shall be so designed that the safety of the nuclear reactor facilities will not be impaired by postulated internal flooding that may take place within the nuclear reactor facilities. Also, leakage from radiation controlled area shall be prevented by design in case of postulated internal flooding with radioactive materials.

(New)

[Detailed Notes on Requirements]

A. “internal flooding that may take place within the nuclear reactor facilities” refer to flooding caused by failure of equipment or piping installed in the nuclear reactor facilities (including ones caused by earthquake), the operation of the fire protection system, or sloshing in the spent fuel pool or spent fuel pit.

B. “the safety of the nuclear reactor facilities will not be impaired” in this requirement means that it must be capable to achieve hot shutdown and to maintain subsequent cold shutdown and radioactive materials confinement functionable, and when in shutdown, to maintain the function to cool and to feed water to the spent fuel pool or spent fuel pit functionable.

2. General Technical Requirements for Nuclear Reactor Facilities

(5) Design Considerations against Fire

[Basic Requirements] The nuclear reactor facilities shall be so designed that their safety will not be impaired by fire considering individual protective measure for preventing, detecting and extinguishing fire, and mitigating its effect. Also, those protective measures shall be so designed as not to impair the safety functions of SSCs with safety functions as a result of its failure or malfunction. (Corresponding to Guideline 5 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A. "designed that their safety will not be impaired by fire considering individual protective measure for preventing, detecting and extinguishing fire, and mitigating its effect" refers the design to be in compliance with a separately established rule (*).

((*) A review guide is to be established by the nuclear regulation authority referring to the guidelines and rules in the U.S. and other countries.)

2. General Technical Requirements for Nuclear Reactor Facilities

(6) Design Considerations against Environmental Conditions

[Basic Requirements] SSCs with safety functions shall be designed to withstand any environmental conditions under which their safety functions are expected to be. (The reactor containment isolation valve shall be in such a way that its isolation function is not lost even in case of a loss of drive power source.

[Detailed Notes on Requirements]

A. "any environmental conditions under which their safety functions are expected to be" refer to any environmental condition to which the SSCs, whose functions are expected to work during normal operations, anticipated operational occurrences and design basis accidents, may possibly be exposed in their service life.

2. General Technical Requirements for Nuclear Reactor Facilities

(7) Design Considerations for Shared Use

[Basic Requirements]

1. SSCs with safety functions of especially high importance shall generally not be shared or interconnected between multiple units unless it contributes to enhance safety.
2. SSCs with safety functions shall be so designed that in case they are shared or interconnected between multiple units , the safety of the reactors will not be impaired by the shared use.

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

[Detailed Notes on Requirements]

A. “SSCs with safety functions of especially high importance” will be specified in consideration of “Regulatory Guide for Reviewing Classification of Importance of Safety Function of Light Water Nuclear Power Reactor Facilities”

B. “contributes to enhance safety" means that advantage can be achieved by sharing while each shared facility satisfies technical requirements; for example, the control room design that shares the same space between twin units has advantage of operators accommodation, satisfying habitability requirements.

C. “Sharing” means shared use of the same SSCs between multiple units.

D. “Interconnected” means a connection of systems or components between multiple units.

2. General Technical Requirements for Nuclear Reactor Facilities

(8) Design Considerations against Operators' Actions

[Basic Requirements] The nuclear reactor facilities shall be designed taking appropriate preventive measures against operator error. SSCs with safety functions comprising the nuclear reactor facilities shall also be designed for operators to operate easily under the environmental conditions under which the operation is required. (Corresponding to Guideline 8 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A. "designed taking appropriate preventive considerations..." refers to that measures, such as to pay attention to the layout of panels or the operability of valves reflecting human engineering-oriented factors, to pay attention so as to be able to grasp the condition of nuclear reactor facilities correctly and quickly through instrumentation and alarms, or to pay attention so as to prevent errors during maintenance and inspection, are taken into the design. And it also refers to that the nuclear reactor facilities is being designed so as to ensure that the necessary safety functions are maintained without operator's actions for a certain length of time after the occurrence of anticipated operational occurrences or design basis accident.

B. "designed for operators to operate easily" refers to that the safety features are being designed so that operators can operate facilities easily, even assuming both environmental condition simultaneously following with a significant possibility the event caused the concerned operation (for example, aftershock etc.) and environmental condition simultaneously brought with a significant possibility at the nuclear reactor facilities.

2. General Technical Requirements for Nuclear Reactor Facilities

(9) Design Considerations for Reliability

[Basic Requirements]

1. SSCs with safety functions shall be so designed that their adequately high reliability will be ensured and maintained as required according to the importance of their safety functions.

2. Systems with safety functions of especially high importance shall be designed to be able to achieve their safety functions even in case that off-site power is unavailable in addition to an assumption of a single failure in any of the components that comprise the systems.

3. The systems referred to in previous provision shall be designed with redundancy or diversity, and with independence considering their structure, principles of the function, and safety functions to achieve, etc.

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

[Detailed Notes on Requirements]

A. "Adequately high reliability will be ensured and maintained as required according to the importance of their safety functions " and "Systems with safety functions of especially high importance" will be specified separately in consideration of "Regulatory Guide for Reviewing Classification of Importance of Safety Function of Light Water Nuclear Power Reactor Facilities"

B. "Single failure" is categorized into a single failure of active component and a single failure of passive component. Any system with safety functions of especially high importance shall be designed to achieve required safety functions on both on an assumption of a single failure of active component in short term and on an assumption of either a single failure of active component or the single failure of assumed passive component in longer term.

C. The boundary between the short-term and the long-term should basically be 24 hours or the point of time on which the operation modes is switched. For example, the injection mode of the emergency core cooling system or the containment heat removal system of a pressurized-water reactor is switched to the recirculation mode

D. In the evaluation of safety functions for long-term, either above mentioned single failure of active component or single failure of assumed passive component is not needed to be assumed, if it is sure to be removed or repaired in a period within which no obstacles for safety occur even under the postulated severest conditions.

E. The requirement of redundancy for the components shall not be applied if the possibility of occurrence of a single failure can be reasonably explained to be extremely small or if it can be proved by safety analysis that the system function whose loss is estimated on the assumption of a single failure can be maintained by using the function of another system.

2. General Technical Requirements for Nuclear Reactor Facilities

(10) Design Considerations for Testability

[Basic Requirements]

SSCs with safety functions shall be designed to be able to be tested or inspected during reactor operation or shutdown to verify their integrity and capability by appropriate methods according to the importance of their safety functions.

(Corresponding to Guideline 10 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A. "Adequate methods" include the use of bypass systems for test in case test or inspection using systems in service is not appropriate.

B. The tests and inspections must be carried out according to the following rules:

(a) SSCs having safety functions, which are on standby during the operation of the reactor, must allow tests and inspections periodically during operation. This rule, however, does not necessarily apply when the tests and inspections during operation can affect the operation of the reactor significantly. Also, each of the systems and components featuring redundancy or diversity must be able to conduct tests and inspections independently.

“Tests and inspections during operation” include tests specified in "Ordinance for Establishing Technical Standards for Nuclear Power Generation Systems of the Ministry of International Trade and Industry No. 62" (plan to be revised as Nuclear Regulatory Authority regulation).

(b) The tests to confirm the function of each channel of safety protection systems during the operation must be conducted such that the functions themselves are retained during the tests and at the same time no unnecessary actions are caused in the reactor shutdown system, the emergency core cooling system, etc.

(c) “Regular tests and inspections during shutdown of the reactor” include tests specified in the rules relevant to the Reactor Regulation Law.

C. The requirements in the right-hand column must be satisfied for the facilities listed in the left-hand column of the following table:

Structure, system, or components	Requirement
Reactivity control system and Reactor shutdown system	The system shall be designed to allow testing
Reactor coolant pressure boundary (RCPB)	RCPB shall be designed to be capable of being tested and inspected throughout the service life of the nuclear reactor.
Residual heat removal system	The system should be designed to allow testing
Emergency core cooling system	The system shall be designed to be capable of being tested and inspected on a periodical basis. The system shall also be designed to allow testing and inspection of each constituent system independently so that the integrity and redundancy of the system can be verified.
System for transporting heat to ultimate heat sink	The system shall be designed to allow testing
Reactor containment	The containment shall be designed that the leakage rate of the entire containment can be measured under a specified pressure on a periodical basis and designed to allow leakage tests at such important portions as penetrations for electric cables, pipelines, etc. and access openings.
Reactor containment isolation valves	The valves shall allow performance testing on a periodical basis, and important ones among them shall be able to conduct leak test.
Reactor containment heat removal system	The system shall be designed to allow testing
Systems for controlling containment facility atmosphere	The system shall be designed to allow testing
Safety protection system	The system shall be designed to be capable of being tested periodically during reactor operation in general and allow testing of each constituent channel independently so that the integrity and redundancy of the system can be verified.
Electrical systems	The electrical systems associated with safety

	functions of high importance shall be designed such that their important portions can be periodically tested and inspected.
Fuel handling systems	Appropriate periodical testing and inspection of SSCs with safety functions shall be possible.

2. General Technical Requirements for Nuclear Reactor Facilities

(11) Design Considerations for Communication Systems

[Basic Requirements]

1. The nuclear reactor facilities shall be provided with warning systems and communication systems that can give necessary instructions to all the people present in the nuclear power plant when design basis accident occurs.
2. The communication systems and data transmission systems to necessary points outside of the nuclear power plant shall be designed dedicated and diversified.
3. Diversified communication systems to necessary points inside of the nuclear power plant shall be provided.

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

[Detailed Notes on Requirements]

A. "Communications systems" refer to facilities that can convey instructions on operation, work, evacuation, etc. to the person in various places inside or outside of the building with voice from the control room and the like or that can convey reports regarding occurrence of accidents etc. to necessary points outside of the nuclear power plant. .
B. "Data transmission systems" refer to facilities to transmit necessary data from the power station (control rooms, etc.) to Emergency Response Support System (ERSS), etc. outside the nuclear power plant.
C. "designed dedicated and diversified" refers to so designed that the circuit can be used without limit by congestion, etc. and is with diversity in communication methods (for example, cable, wireless). This can be achieved by dedicated communication lines installed by reactor licensees on their own terms such as dedicated satellite IP telephone or communication lines that telecommunications carriers provide only for specific customers.
D. The communication systems and so on shall be designed to be operable by means of connecting with the emergency on-site power systems or the uninterruptible power

systems even in case that off-site power supply is uncertain.

2. General Technical Requirements for Nuclear Reactor Facilities

(12) Design Considerations for Escape Routes

[Basic Requirements]

1. The nuclear reactor facilities shall be provided with evacuation lighting that functions even in case that ordinary power for lighting is lost and have safe escape routes provided with simple, clear and long-lasting guide signs.

2. The nuclear reactor facilities shall be provided with lightning and dedicated power supplies that enable on-the-spot works to respond accident separate from the evacuation lightning.

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

[Detailed Notes on Requirements]

A. "provided with lightning and dedicated power supplies that enable on-the-spot works to respond accident separate from the evacuation lightning" must be that the lightning equipment enables on-the-spot works for recovery from any accident, whenever they are necessitated, the daytime or the nighttime, and wherever the location is. It is to be noted, however, that the use of temporary lightning (portable type) can be considered when there is allowable time for the preparation of them in balance with the urgency of the on-the-spot works.

3. Individual Systems of Nuclear Reactor Facility

(1) Reactor Core and Associated Features

[Basic Requirements]

(Reactor core)

1. The reactor core shall be designed to assure, with the aid of the functions of associated reactor cooling system, reactivity control system, instrumentation and control systems, and safety protection system, that the acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences.
2. Components, other than fuel rods, that make up the reactor core or are located in its proximity within the reactor pressure vessel shall be designed to be capable of ensuring safe reactor shutdown and proper core cooling during normal operation, anticipated operational occurrences and design basis accidents.

(Fuel)

3. The fuel assemblies shall be designed not to lose their integrity despite various unfavorable factors that may take place during their use in the reactor.
4. The fuel assemblies shall be designed not to be excessively deformed during transport or handling.

(Reactor characteristics)

5. The reactor core and associated systems shall be designed to have inherent characteristics to suppress the increase of output and well capable of controlling reactor power oscillation if it occurs.

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

[Detailed Notes on Requirements]

(Reactor core)

A. An acceptable fuel design is to be determined based on the maximum temperatures of fuel pellet and fuel cladding, the maximum heat flux, the minimum critical heat flux ratio, the minimum critical power ratio, the maximum enthalpy of fuel pellet, the maximum deformation of fuel assembly, maximum linear power density (BWR), etc.

B. Specific evaluation is based on "Evaluation Guide for Reactivity-initiated Accident of Light Water Nuclear Power Reactor Facilities" (decision by Nuclear Safety Commission on January 19, 1984, Revised partly on August 30, 1990) and the like.

(Fuel)

C. "Unfavorable factors that may take place" include difference between internal and external pressures of fuel rod, irradiation of fuel rod and other materials, fluctuations in pressure and temperature caused by load change, chemical effects, static and dynamic loads, deformation of fuel pellet, composition change in fuel rod filler gas, etc.

D. "designed not to lose integrity", means a design that ensures the confinement function of fuel cladding, the capability of control rods to be inserted into fuel assemblies, and capability of geometry of fuel assemblies to be cooled, during normal operation and anticipated operational occurrences for necessary period.

E. Specific evaluation is based on "Fuel-design method of light water nuclear power reactors" (approval by Nuclear Safety Commission on May 12, 1988) and the like.

(Reactor characteristics)

F. "Have inherent characteristics to suppress the increase of output" means that the reactivity feedback, which is the integration of Doppler coefficient, moderator temperature coefficient, moderator void coefficient, pressure coefficient, etc., quickly works to suppress the increase of output during the transient of reactor output throughout all operational range and thus prevent or mitigate fuel damage."well capable of controlling reactor power oscillation if it occurs" means that adequate attenuation characteristics are provided or reactor power oscillation can be controlled so that the acceptable fuel design limits are not exceeded.

3. Individual Systems of Nuclear Reactor Facility

(2) Reactivity Control System and Reactor Shutdown System

[Basic Requirements]

(Reactivity control system)

1. The reactivity control system shall be designed to be capable of regulating reactivity changes expected to occur during normal operation, thereby capable of maintaining the stable operational conditions.
2. The maximum reactivity worth of control rods and positive reactivity insertion rate shall be such that postulated reactivity-initiated accidents will result in neither a damage of the reactor coolant pressure boundary nor damage of the core, core support structures and reactor pressure vessel internals that may impair core cooling.
3. The reactivity control system shall be designed to have at least two independent systems capable of making the core subcritical from hot standby or hot operational conditions and also capable of maintaining the core subcritical under hot conditions.
4. During normal operation and anticipated operational occurrences, at least one independent system out of the reactivity control system shall be designed to be capable of making the core subcritical under hot conditions and capable of maintaining the core subcritical under hot conditions until the reactivity addition occurs due to the collapse of xenon after the end of the transient, without leading to the acceptable fuel design limits being exceeded.
5. At least one independent system out of the reactivity control system shall be designed to be capable of making the core subcritical under cold conditions and capable of maintaining the core subcritical under cold conditions.
6. In case of a design based accident, at least one independent system out of the reactivity control system shall be designed to be capable of making the core subcritical and at least one independent system out of the reactivity control system shall be designed to be capable of maintaining the core subcritical.

(Reactor shutdown system)

7. The reactor shutdown system consist of control rods shall be designed to be capable of making the core subcritical under hot conditions even when one control rod with the maximum reactivity worth (one set in case that the control rods belong to the same hydraulic control unit) is withdrawn completely out of the core and cannot be inserted. And even when one control rod with the maximum reactivity worth (one set in case of the control rods belonged to the same hydraulic control unit) is withdrawn completely out of the core and cannot be inserted, the system above shall be designed to be capable of making the core subcritical under cold conditions in cooperation with reactivity control system, if necessary.

*The testability is arranged in the general requirements.

(Corresponding to Guideline 14 to 18 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A. In evaluating the "maximum reactivity worth of control rods", if reactivity worth control devices, by which limit the insertion and arrangement of control rods in conjunction with the operational conditions of the reactor are implemented and the like, the effects of them can be taken into account.

B. "Postulated reactivity-initiated accident" refers to an accident in which abnormal positive reactivity insertion takes place in the reactor, which is defined in "Regulatory Guide for Reviewing Safety Assessment of Light Water Nuclear Power Reactor Facilities" and "Regulatory Guide for Evaluating Reactivity-initiated Accidents of Light Water Nuclear Power Reactor Facilities".

C. "Capable of maintaining the core subcritical under hot conditions" refers to the capability of maintaining the subcritical condition during the period that xenon decay add reactivity after the end of a transient and it may be acceptable to expect the function of other systems to maintain subcriticality beyond that period.

D. "Capable of making the core subcritical under cold conditions and of maintaining the core subcritical under cold conditions" refers to achieving a cold subcritical condition from a hot subcritical condition compensating reactivity addition due to xenon decay and reactor coolant temperature change and maintain cold shutdown condition.

E. Systems adopted for light water reactor at the present, which are control rod and the systems by soluble poison (boric acid injection system in boiling light water reactor and boric acid injection system of chemical and volume control system in pressurized water reactor) can be considered as a reactivity control system that satisfied the Clause 3, because of its function.

F. In case the operation of any other system capable of shutting down the reactor can be expected together with the reactivity control system at the time of a design basis accident, its contribution may be taken into account in the design considerations. A typical case would be the contribution of the emergency core cooling system together with the reactivity control system in making and maintaining the core subcritical in the event of a main steam pipe rupture in a PWR.

G. Both control rods and chemical and volume control system are included in reactivity control system and reactor shutdown system for pressurized-water reactor, both control rods and SLC are included in reactivity control system and reactor shutdown system for boiling water reactor, and reactor recirculation flow control system is included in reactivity control system.

3. Individual Systems of Nuclear Reactor Facility

(3) Reactor Coolant Pressure Boundary

[Basic Requirements] 1. The reactor coolant pressure boundary shall so designed that its integrity will be ensured during normal operation, anticipated operational occurrences and design basis accidents. 2. The pipes connected to the reactor coolant system shall be in general installed with isolation valves. 3. The reactor coolant pressure boundary shall be designed not to exhibit brittle characteristics and develop any quickly propagative failure during normal operation, maintenance, testing, anticipated operational occurrences and design basis accidents. 4. Means shall be provided for quick and proper detection of the leakage of the reactor coolant, if any, from the reactor coolant pressure boundary. *The testability is arranged in the general requirements. (Corresponding to Guideline 19 to 22 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A. The "reactor coolant pressure boundary" refers to the equipment and pipes in the following range.
   (a) Reactor pressure vessel and its equipment (those that can be connected directory to the main body, i.e. control rod drive mechanism housing, etc.)
   (b) Equipment and piping composing the reactor coolant system. Note that they refer to a primary coolant pump, the water chamber, plates and pipes of the steam generator, a pressurizer, primary coolant system piping, valves, etc. in PWR. They are limited to a range including the second isolation valve seen from the reactor side among the main steam line and feed water pipe in BWR.

(c) Connected piping

(i) If it includes a valve normally opened and closed at the time of accidents, it is limited to a range including the second isolation valve from the reactor side.

(ii) If it includes a valve normally closed and closed at the time of accidents, and likely opened normally or at the time of accidents, it is limited to a range including the second isolation valve seen from the reactor side.

(iii) If it includes a valve normally closed and opened at the time of accidents other than (ii), it is limited to a range including the first isolation valve from the reactor side.

(iv) The emergency core cooling system, etc. with a valve to be normally and opened at the time of a loss of the reactor coolant accident also conforms to (i).

(v) In the above, the “isolation valve” refers to the automatic isolation valve, check valve, or close valve locked at the normal time or remotely controlled close valve.

B. ”... so designed that its integrity will be ensured” means that the design reflects the consideration such that abrupt cooling or heating of the reactor coolant pressure boundary and abnormal pressure rise within it can be suppressed with the aid of reactor shutdown system, reactor cooling system, instrumentation and control systems, safety protection system, safety valves, etc. and that the reactor coolant pressure boundary itself can sufficiently withstand temperature change and pressure to be experienced with extremely small possibility of failure or of abnormal leakage of reactor coolant.

C. “... in general be fitted with isolation valves” refers to the reactor coolant pressure boundary design in which the pipes connected to the reactor coolant system and forming the boundary in part are fitted with appropriate isolation valves so that loss of reactor coolant can be stopped in case of abnormal leakage from the portions not forming the reactor coolant pressure boundary considering the service conditions and purposes of those pipes during normal operations. “in general” as mentioned above means that for piping to perform important safety related instrumentation, sampling, etc. and that a leakage from the piping is small enough to be fully permitted or that for piping to place safety valves having the overpressure protection function, it is not equipped with the isolation valve.

D. Specific evaluation depends on "Evaluation Guide for Reactivity-initiated Accidents of Light Water Nuclear Power Reactor Facilities" (decision by Nuclear Safety Commission on January 19, 1984, Revision partly on August 30, 1990) and the like.

3. Individual Systems of Nuclear Reactor Facility

(4) Reactor Cooling System

(a) Reactor Coolant Supply System

[Basic Requirements]

The reactor coolant supply system shall be designed to be capable of supplying as much coolant as required at a proper flow rate to restore the necessary inventory of the reactor coolant in case of a small leakage.

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

[Detailed Notes on Requirements]

A. The "reactor coolant supply system" refers to the system that supplies the reactor coolant to the reactor coolant system (control rod drive hydraulic control system and reactor core isolation cooling system (excluding feed water system) in BWR; charging pump in PWR).

B. "A small leakage" refers to coolant leakage through seals of valves or pumps that consists the reactor coolant pressure boundary or through small cracks in the reactor coolant pressure boundary.

3. Individual Systems of Nuclear Reactor Facility

(4) Reactor Cooling System

(b) Systems for Removing Residual Heat

[Basic Requirements]

1. The systems for removing residual heat shall be designed to be capable of removing fission product decay heat and other residual heat from the core during reactor shutdown, thereby preventing to exceed the acceptable fuel design limits and design conditions for the reactor coolant pressure boundary.

2. The systems for removing residual heat shall be designed to fulfill their safety functions even in case that off-site power is unavailable in addition that a single failure of any of the components that consist the systems is assumed.
*The reliability and testability are arranged in the general requirements.

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

[Detailed Notes on Requirements]

A. The “systems for removing residual heat” refer to the systems provided so as to remove residual heat even in case heat removal cannot be achieved by main condenser (reactor core isolation cooling system, residual heat removal system, high pressure core spray system, automatic depressurization system, etc. in BWR; steam generator, main steam relief valve, main steam safety valve, auxiliary feedwater system, residual heat removal system, etc. in PWR). In association with these, BWR has main steam relief safety valve to reduce pressure in the reactor coolant system, and PWR has pressurizer relief valve, etc. for the same purpose.

B. “Other residual heat” refers to heat accumulated in the materials consist of the core, reactor coolant system, etc., in the reactor coolant and in the secondary coolant (in the case of PWR).

3. Individual Systems of Nuclear Reactor Facility

(4) Reactor Cooling System

（c）Emergency Core Cooling System

[Basic Requirements]

1. The emergency core cooling system shall be designed to be capable of preventing serious damage of fuel and of limiting the reaction between fuel cladding metal and water to a sufficiently small amount in case of a loss of reactor coolant accident resulting from a postulated break in piping, etc.

2. The emergency core cooling system shall be designed to fulfill its safety functions even in case that off-site power is unavailable in addition to a single failure of any of the components that comprise the system is assumed.

*The reliability and testability are arranged in the general requirements.

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

[Detailed Notes on Requirements]

A. Specific evaluation depends on “Evaluation Guide for Emergency Core Cooling System Performance of Light Water Nuclear Power Reactors” (decision by Nuclear Safety Commission on July 20, 1981, Revised partly on June 11, 1992) and the like.

B. A “postulated break in piping” refers to an assumed loss of coolant accident that is defined by the “Regulatory Guide for Evaluating Emergency Core Cooling System Performance of Light Water Reactor.”

C. ”A break in piping, etc.” includes the events that cause a loss of reactor coolant without any physical break, such as stick open of a relief valve.

D. The judgment of a “sufficiently small amount” depends on the “Regulatory Guide for Evaluating Emergency Core Cooling System Performance of Light Water Reactor.”

3. Individual Systems of Nuclear Reactor Facility

(4) Reactor Cooling System

(d) Systems for Transporting Heat to Ultimate Heat Sink

[Basic Requirements]

1. The systems for transporting heat to an ultimate heat sink shall be designed to be capable of transferring heat generated or accumulated in SSCs with safety functions of especially high importance to an ultimate heat sink.

2. The systems for transporting heat to an ultimate heat sink shall be designed to fulfill their safety functions even in case that off-site power is unavailable in addition to a single failure of any of the components that comprise the systems is assumed.

3. The systems for transporting heat to an ultimate heat sink shall be designed in consideration of physical protections against design basis tsunami, flooding, and external man-induced events.

*The reliability and testability are arranged in the general requirements.

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

[Detailed Notes on Requirements]

A. ”Ultimate heat sink” refers to the sea, river, pond, lake or open air.

B. The ”systems for transporting heat to an ultimate heat sink” refers to the systems that transport heat from the emergency core cooling system, residual heat removing systems, etc. to the ultimate heat sink (component cooling system, component cooling sea water system, etc.).

3. Individual Systems of Nuclear Reactor Facility

(4) Reactor Cooling System

（e）Steam Turbine and Associated Equipment

[Basic Requirements]

1. The steam turbine and associated components shall be designed not to influence the safety of the nuclear reactor facilities.

2. The steam turbine and associated components shall be designed to be capable of monitoring its conditions to prevent its damage from influencing the safety of the nuclear reactor facilities.

(*Equipment needed new requirements after the unification with the Electricity Utilities Industry Law)

[Detailed Notes on Requirements]

A. “Designed not to influence the safety of the nuclear reactor facilities” means that the materials with resistance against the chemical and physical effect on them under assumed environmental conditions are used and that sufficient structural strength are provided by taking countermeasures against vibration and over speed of steam turbine and that there is no influence the safety of the nuclear reactor facilities even at the time of damage.

3. Individual Systems of Nuclear Reactor Facility

(5) Reactor Containment Facility

(a) Reactor Containment Facility

[Basic Requirements]

(Functions of Reactor Containment)

1. The reactor containment shall be designed to withstand the load (pressure, temperature, dynamic load) resulting from the postulated events and an appropriate seismic load and not to exceed the specified leakage rate with properly operating isolation functions.

* The testability is arranged in the general requirements.

(Prevention of Reactor Containment Boundary Failure)

2. The reactor containment boundary shall be designed not to show brittle behavior and not to develop any quickly propagative failure during normal operation, maintenance, testing, anticipated operational occurrences and design basis accidents.

(Isolation Function of Reactor Containment)

3. The pipes that penetrate the reactor containment walls shall be fitted with containment isolation valves except that they are for the important safety related measurement or sampling and that they are pipes, such as hydraulic pipes of the control rod drive mechanism, through which the amount of leakage is small enough to be acceptable.

4. The containment isolation valves to be fitted in principal pipes shall be designed to be automatically and properly closed in case of design basis accidents that necessitates the retention of isolation function except that those piping systems are required for mitigating accidents.

* The testability is arranged in the general requirements.

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(Reactor Containment Isolation Valves)

5. The containment isolation valves shall be located close to the reactor containment.

6. The installation of the containment isolation valves shall be subject to the following:

(1) Of the pipes that open inside the reactor containment or connect to the reactor coolant pressure boundary, those which are not closed outside the reactor containment shall be provided with one containment isolation valve inside the reactor containment and one outside. Two containment isolation valves outside the reactor containment may be permitted if it is difficult to install one containment isolation valve inside the reactor containment in terms of physical or environmental conditions with reasonable demonstration.

(2) Of other pipes than (1) above, those which are closed inside or outside the reactor containment shall be fitted with one containment isolation valve outside the reactor containment. One containment isolation valve inside the reactor containment may be permitted with reasonable demonstration.

(3) Of other pipelines than (1) and (2) above, those fitted with rupture disk may be provided with one isolation valve, which is usually closed, inside or outside the reactor containment in addition to the pressure relief plate with reasonable demonstration.

(4) The containment isolation valves shall not lose containment function even if driving power supply is lost after its closure.

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

[Detailed Notes on Requirements]

A "The postulated events" refers to the ones for judging the adequacy of the reactor containment design and their evaluated result will be most severe against criteria, evaluated with regard to an increase in the pressure and the temperature, a dynamic load, inflammable gas, and the concentration of radioactive materials which cause a problem in securing the functions of the reactor containment. Specifically, it is stipulated in the "Examination Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities (established by Nuclear Safety Commission on August 30, 1990 and revised on March 29, 2001)"

B "Containment isolation valve" are automatic isolation valves (including check valves designed to adequately work for containment isolation in case of an accident), normally locked shut-off valves and remote-controlled shut-off valves. "Check valves designed to adequately work for containment isolation in case of an accident" as referred to above are the check valves designed to maintain necessary isolation function by means of gravity, etc. even in case the pipe concerned is damaged either inside or outside the reactor containment and the back pressure to the valves is totally lost as a result.

C “Principal pipes” refer to the ones which must be fitted with containment isolation valves and may cause a leakage beyond tolerable limits from the reactor containment if left uncontrolled during normal operational conditions, except that those pipes whose containment isolation valves are closed during hot operation.

D “Be designed to be automatically and properly closed” refers to the capability of containment isolation valves to automatically close in response to the containment isolation signals from the safety protection system, for example, and minimize the leakage of radioactive materials from the reactor containment in conjunction with isolation barriers other than containment isolation valves even in case that off-site power is unavailable in addition to an assumption of a single failure.

E The meaning of " except that those piping systems are required for mitigating accidents " as mentioned here is that those pipes, for example ECCS pipes etc., are not required to be shut off by automatic isolation signals in order not to block such systems' safety functions. Even in that case, however, the loss of the function of containment isolation shall not be lost.

F The reset function shall be considered with the containment isolation valves, which are shut off automatically, for the sake of necessary post-accident activities.

G "those which are not closed outside the reactor containment" are the pipes which will make intolerable release paths of radioactive materials from the atmosphere in the reactor containment to the outside depending on the conditions of the pipes during a design based accident if they are not isolated.

H “Rupture disk” is allowed to be installed only in the case where it is demonstrated that it does not affect the function of the equipment installed separately as a severe accident measure, and in that case, it may be released at the pressure set out much lower than the containment vessel design pressure.

I The specific evaluation is based on “Evaluation guideline on dynamic load added to containment vessel pressure control system “Regulatory Guide for Evaluating Dynamic Loads on BWR MARK-II Containment Pressure Suppression Systems “(authorized by Nuclear Safety Commission on July 20, 1981 and in part revised on August 30, 1990), etc.

3. Individual Systems of Reactor Facility

(5) Reactor Containment Facility

(b) Reactor Containment Heat Removal System and Systems for Controlling Containment Facility Atmosphere

[Basic Requirements]

(Reactor containment heat removal system)

1. The reactor containment heat removal system shall be designed to have the function that sufficiently reduces the containment pressure and temperature resulting from the released energy in case of the postulated events for reactor containment design.

2. The reactor containment heat removal system shall be designed so that the system can fulfill its safety functions even in case that off-site power is unavailable in addition to an assumption of a single failure of any components that comprise the system.

* The reliability and testability are arranged in the general requirements.

(System for controlling containment facility atmosphere)

3. The containment facility atmosphere cleanup system shall be designed to be capable of reducing the concentration of radioactive materials released to the environment in case of the postulated events for reactor containment design.

4. The inflammable gas concentration control system shall be designed to be capable of controlling the concentration of hydrogen or oxygen present in the reactor containment in case of the postulated events for reactor containment design, thereby maintaining the integrity of the containment facility.

5. The systems for controlling containment facility atmosphere shall be designed so that they can fulfill their safety functions even in case that off-site power is unavailable in addition to an assumption of a single failure of any components that comprise the systems.

* The reliability and testability are arranged in the general requirements.

(Equivalent to Guidelines 32 and 33 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A. The “reactor containment heat removal system” refers to the system capable of reducing pressure and temperature in the reactor containment effectively enough in case of the postulated events for reactor containment design, and includes the reactor containment spray system and its heat removal system.
B. The “system for controlling containment facility atmosphere” refers to the containment facility cleanup system and the inflammable gas concentration control system.
C. The “containment facility atmosphere cleanup system” includes standby gas treatment system, standby recirculation gas treatment system and containment spray system of BWR; annulus air recirculation system and containment spray system of PWR.
D. “Controlling the concentration of hydrogen or oxygen” means to maintain inert atmosphere in the reactor containment or to control the concentration of hydrogen or oxygen to the level below the combustible limit by means of re-combiner, etc., if necessary.

3. Individual Systems of Reactor Facility

(6) Instrumentation and Control Systems

(a) Instrumentation and Control Systems

[Basic Requirements]

1. The instrumentation and control systems shall be designed to satisfy the following requirements during normal operations and anticipated operational occurrences.
   (1) The parameters necessary to maintain the integrity of the reactor core, reactor coolant pressure boundary, reactor containment boundary and associated systems shall be controlled and maintained within presumed range.
   (2) Monitoring of the aforementioned parameters shall be possible within presumed range so as to allow necessary countermeasures to be taken as required.

2. The instrumentation and control systems shall be designed to satisfy the following requirements during design basis accidents.
   (1) The systems shall be designed to enable monitoring the parameters necessary to recognize the status of an accident and to take countermeasures over sufficient range and time period under the environment presumed in accidents.
   (2) The systems shall also be designed to enable monitoring or estimating of the condition of reactor shutdown and core cooling in particular through two or more kinds of parameters.
   (3) Necessary parameters shall be recorded and preserved certainly.

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

[Detailed Notes on Requirements]

A “Parameters necessary to maintain the integrity….” include neutron flux of the reactor core, neutron flux distribution, water level of the reactor, such parameters as pressure, temperature and flow rate of the reactor coolant system, water quality of the reactor coolant, and such parameters as pressure, temperature and atmospheric gas concentrations in the reactor containment.

B "Parameters necessary to recognize the status of an accident and to take countermeasures..." include pressure, temperature, hydrogen gas concentration and radioactive materials concentrations of the reactor containment atmosphere etc.

C “Necessary parameters” mentioned in 2. (3) include neutron flux in the core, reactor water level, pressure and temperature of reactor coolant system, etc. those are necessary to monitor the most important three function of reactor shutdown, core cooling, and containing radio activity.

D “be recorded and preserved” means the information about necessary parameters is available following the course of an accident.

E The instrumentation and control system under the design basis accident shall be based on the provisions of “Regulatory Guide for Reviewing Radiation Monitoring in Accidents of Light Water Nuclear Power Reactor Facilities ” (authorized by Nuclear Safety Commission on July 23, 1981 and in part revised on September 19, 2006).

3. Individual Systems of Reactor Facility

(6) Instrumentation and Control Systems

(b)Safety Protection Systems

[Basic Requirements]

1. The safety protection system shall be designed with redundancy so that a single failure of any components or channels that comprise the system or removal from service of any component or channel does not result in loss of safety functions of the system during normal operation, anticipated operational occurrences and design basis accidents.

2. The safety protection system shall be designed such that the channels comprising the system are separated from each other taking into account the independence between them as much as practicable, thereby preventing loss of its safety functions during normal operation, maintenance, testing, anticipated operational occurrences and design basis accidents.

3. The safety protection system shall be designed to detect the abnormal state during anticipated operational occurrences and initiate automatically the operation of appropriate systems including the reactor shutdown system in order to ensure that the acceptable fuel design limits are not exceeded.

4. The safety protection system shall be designed to detect the abnormal state in a design basis accident and initiate automatically the operation of the reactor shutdown system and necessary engineered safety features.

5. The safety protection system shall be designed to allow the nuclear reactor facilities to be settled in a state of safety eventually in case of driving power loss, system cut-off or any other unfavorable situation.

6. The safety protection system shall be designed to take into account cyber security such as the prevention of intrusion from external network.

7. The safety protection systems shall be designed to be functionally separated from instrumentation and control systems in case that the both systems share common elements so that the system does not lose its safety functions by the influence from instrumentation and control systems.

* The reliability and testability are arranged in the general requirements.

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

[Detailed Notes on Requirements]

A A "channel" refers to the constituent elements (resistors, capacitors, transistors, switches, lead wires, etc.) and modules (assemblies of interconnected constituent elements) to produce a single signal necessary for a safety protection action, and covers a range from a detector to logic circuit input terminals.

B "Channels… are separated from each other" means that in case one channel develops an unfavorable condition, the other channel will not develop any unfavorable condition of the same nature or its safety function will not be affected.

C A typical function of the safety protection system during transient is detecting the abnormal state and actuating the reactor shutdown system to scram the reactor in order to prevent the reactor power from exceeding given level or increasing too fast.

D "Driving power loss, system cut-off or any other unfavorable situation " refers to the loss of electric power or instrumentation air or a situation in which the safety protection system has its logic circuit cut off for some reason. The factors to be considered as the "unfavorable situation" shall be determined depending on the respective design, including environmental conditions.

E "Settled in a state of safety eventually" means that even in case of a failure in the safety protection system, the nuclear reactor facility will be settled into a state on the safe side or can be maintained in a safe state despite the failure in the safety protection system being not repaired. .

F "Designed to take into account cyber security" means a design that enables the physical or functional separation of hardware, and that prevents unapproved actions and modifications as a countermeasure against viruses mixed into the system during system introduction stage, system update stage or test stage.

G "... the system does not lose its safety functions" means that, even if any of the components or channels comprising the instrumentation and control systems which are connected to the safety protection system may be subjected to a single failure, mis-operation or removal from service, the safety protection system with its functions not being impaired can fulfill the basic requirements 1 to 6 regarding safety protection system.

3. Individual Systems of Reactor Facility

(6) Instrumentation and Control Systems

(c) Control Room and Others (excluding habitability)

[Basic Requirements]

(Control room)

1. The control room shall be so designed that the operating status and principal parameters of reactor and principal related facilities can be monitored.

2. The control room shall be designed so that it can grasp the situation outside the nuclear reactor facility.

3. The control room shall be designed so that prompt manual control can be performed, whenever required, to maintain safety.

(Reactor shutdown function from outside of control room)

4. The nuclear reactor facilities shall be designed to have the following functions that allow reactor to be shut down from an appropriate location outside the control room.
   (1) Prompt hot shutdown of the reactor together with necessary instrumentation and control in order to maintain the nuclear reactor facility in a safe state.
   (2) Maintaining cold shutdown state of the reactor by appropriate control procedures.

(Corresponding to Guidelines 41 and 42 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A "Principal parameters... can be monitored" means that, of the parameters required to be monitored under the basic requirements regarding instrumentation and control systems, those which need to be monitored on a continuous basis can be monitored in the control room.

B “designed so that it can grasp the situation outside the nuclear reactor facility” means a design that makes it possible, from the control room, to grasp natural phenomena, etc. that may affect the reactor facility.

C “prompt manual control” means the operations necessary to shut down the reactor and maintain the reactor cooling after shutting down.

D “that allow reactor to be shut down from an appropriate location outside the control room” means that measures have been taken against cases where the control room is inaccessible for some reason.

E "prompt hot shutdown of the reactor" refers to shutting down the reactor immediately, removing residual heat and maintaining the reactor in the hot shutdown state safely..

3. Individual Systems of Reactor Facility

(6) Instrumentation and Control System

(d) Control Room and Others (limited to habitability)

[Basic Requirements]

The control room shall be designed to be protected against fire, properly shielded so as to allow site personnel to have access to or stay in the control room for necessary operations in case of an accident, and safeguarded against toxic gases and gaseous radioactive materials likely to be released due to a fire or accident by means of a proper ventilation system.

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

[Detailed Notes on Requirements]

A “Site personnel to have access to or stay in the control room” refers to that it is possible for site personnel to have access to the control room for necessary operations and to stay there for a proper length of time after an accident takes place, and that radiation protection measures for the site personnel will become feasible as time passes and radiation level attenuates when they need to approach the control room for a shift after immediate operations

3. Individual Systems of Reactor Facility

(6) Instrumentation and Control System

(e) Emergency Control Center

[Basic Requirements]

The nuclear reactor facilities shall be designed to be able to install an emergency control center from which necessary instructions will be furnished in case of a design basis accident.
(Regarding specific requirements, see the requirements for severe accident.)

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

3. Individual Systems of Reactor Facility

(7) Electrical Systems

(a) Basic Requirements related to the Safety Design of Electrical Systems for the Nuclear Reactor Facilities

[Basic Requirements]

1. The electrical systems shall be designed so that the SSCs with safety functions of especially high importance can be supplied with electricity by either of off-site power (transmission grid) and emergency on-site power when they need electricity to fulfill their safety functions and be designed so that the reliability of electricity supply can be ensured and maintained sufficiently high. In addition, the electrical systems shall be designed so that the abnormal event can be detected and prevented from its expansion and spread, thereby no loss of required power are ensured due to the failure of the related electrical system equipment, such as a main generator, external power supply, emergency auxiliary power supply, or disturbance of the off-site power (transmission grid) and the like.

2. The off-site power system shall be connected to the electric power system with two or more power transmission lines which are connected to two or more independent substations or switchyards and which at least one line out of these lines is physically separated from other lines. Also, in the case of the multiple reactor siting nuclear power station, it shall be designed so that the loss of any two lines of power transmission lines may not cause the loss of off-site power at a same time in these nuclear reactor facilities.

3. The emergency on-site power systems shall have enough capacity and function sure to accomplish the following even with an assumption of a single power systems failure.

(1) Shutting down and cooling the reactor without the acceptable fuel design limits and design conditions for the reactor coolant pressure boundary being exceeded in case of anticipated operational occurrences.

(2) Cooling the reactor core and ensuring the integrity of the reactor containment and

the safety functions of other necessary systems and components in case of a design basis accidents, such as a loss of reactor coolant.

(3) Not depending on sharing the emergency on-site power systems among two or more nuclear reactor facilities.

4. The emergency on-site power systems shall be designed so that they can supply required power during a loss of off-site power for a certain period of time.

＊ The reliability and testability are arranged in the general requirements.

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

[Detailed Notes on Requirements]

A “be designed so that the reliability of electricity supply can be ensured and maintained sufficiently high” refers to a design in which the electricity system achieves system separation in its bus-configuration so that the SSCs with especially high significance may not lose redundancy, and also the individual equipment comprising the electrical system is highly reliable, the bus can be easily switched to be supplied by the emergency on-site power source and, etc.

B “Off-site power (transmission grid)” refers to the electric power system outside the switchyard of nuclear power station, excluding the main generators in the reactor facility concerned and neighboring other reactor facilities.

C “designed so that the abnormal event can be detected and prevented from its expansion and spread,” refers to not only the design that can detect the short-circuit and ground fault of electrical system equipment, low voltage in bus, overcurrent, etc., isolate faulty equipment by circuit breakers and the like, and localize the effects of failure, but also refers to the design that can confine the effects of failure on safety function.

D “The off-site power system” refers to a series of provisions used to supply power to the nuclear reactor facility from an off-site power (transmission grid) system or main power generator.

E “two or more independent substations or switchyards” refers to two or more different substations or switchyards which are not connected to a single substation or switchyard upstream, thereby its failure do not lead all the power transmission lines connected to a nuclear power plant being shut down.

F “Two or more” power transmission lines in the off-site power system are realized by preparing two or more off-site power receiving circuit that connect the off-site power lines with emergency on-site power distribution, which are the combination of lines capable that transmit power and exclusively receive power. “Physically separated” means that power transmission lines are not to set up a single tower etc..

G For the switchyards, and the facilities for power transmission and power receiving, which are installed from the switchyards to main generators, inside the site of a nuclear power station, they shall be built on the soil structure with sufficient supportive properties not to cause an unequal subsidence and a dip, and highly earthquake-resistant materials shall be used for their insulators and breakers. They shall be also segregated from the effect of tsunami and protected, and salt damage shall be considered as well.

H When multiple reactor facilities are established, the external power supply systems shall have a composition of off-site power system with tie lines connected to each reactor facility so that a loss of any two lines may not cause a loss of off-site power at the same time in multiple reactor facilities by connecting to off-site power supply systems with three or more power transmission lines.

I “Emergency on-site power system" refers to emergency on-site power generation facilities (emergency diesel generator, batteries, etc.) and power supply equipment to supply power to the facilities with safety functions of especially high importance including the engineered safety features (emergency bus switch gears, cables, etc.).

J “Not depending on sharing the emergency on-site power systems among two or more nuclear reactor facilities“ means that enough capacity of the emergency on-site power systems shall be provided in own reactor facilities. It is not allowed to be designed that the demand of emergency on-site power are supplied by two or more reactor facilities.

K “supply required power during a loss of off-site power for a certain period of time.” by emergency on-site power systems (emergency diesel generator, etc.) means that power can be supplied by continuous operation of the emergency diesel generator, etc. even if a loss of off-site power that continues for 7 days is assumed. The facility (aseismic design level S) storing the fuels for the emergency diesel generator, etc. shall be designed so that it can store the fuels more than that necessary for 7 days continuous operation within the site.

L Concerning to “safety functions of especially high importance”, it will set with “Regulatory Guide for Reviewing Classification of Importance of Safety Function of Light Water Nuclear Power Reactor Facilities”.

3. Individual Systems of Reactor Facility

(7) Electrical systems

(b) Basic Requirements related to Electric Facilities for Nuclear Power Generation

[Basic Requirements]

1. The switchyard, large transformers and main generators of the electric facilities for nuclear power generation shall be designed so that they may not cause damages to other equipment, taking the isolation of electric circuits, prevention of disconnection, grounding, ground fault protection, over current protection, heat resistance and mechanical impact caused by short-circuit current into considerations.

2. The compressor systems and gas insulated circuit breakers shall be designed so as to be able not only to monitor and control the working pressure, but also to adequately withstand it, and to be corrosion resistant.

3. The rotating part of the main generator shall be designed to have a sufficient mechanical strength. Furthermore, for the main generators of the hydrogen-cooling type, they shall be designed that the leak of hydrogen and mixing in air can be protected and that the leak of hydrogen can be detected if it occurs, alarm goes off, the leak are sealed and the leakage are discharged to the outdoors.

4. The lightning arrester and the like shall be installed with the electric facilities so that their electric circuit may not be damaged by lightning strike.

(Newly set out)

[Detailed Notes on Requirements]

A “Electric facilities for nuclear power generation” refer to the electric facilities for electricity generation from nuclear energy (According to the provision of Article 106 of the Electricity Utility Industry Law). In this case, prescribed here are equipment in the switch yard of the nuclear power station (breaker, disconnecting switch, lighting arrester, insulator, etc.), large transformer for power transmission and reception, main generator, electric circuit to connect these electric facilities to each other, and communication systems for the safety of electric power system.

3. Individual Systems of Reactor Facility

(8) Design Consideration for Total Loss of AC Power Supply

[Basic Requirements]

The nuclear reactor facilities shall be so designed that safe shutdown, proper cooling of the reactor after shutting down and the integrity of reactor containment can be ensured in case of the station blackout for a given time.

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

[Detailed Notes on Requirements]

A To prepare for the station blackout (loss of both off-site power and emergency on-site AC power), emergency DC power shall be designed so that the reactor can safety be shut down and required power for cooling after shutdown can be secured in a given time.

B “the design that safe shutdown, proper cooling of the reactor after shutting down and the integrity of reactor containment can be ensured” refers to the design that sufficient capacity to the load are so assumed in its design as to be supplied by the emergency DC power equipment, to ensure the function to shut down reactor, to cool down after shutdown, and to ensure the integrity of reactor containment.

3. Individual Systems of Reactor Facility

(9) Radioactive Waste Treatment Systems

[Basic Requirements]

(Radioactive gaseous waste and radioactive liquid waste treatment systems)

1. The treatment systems for radioactive gaseous wastes and liquid wastes generated in the course of reactor operation shall be so designed that the quantity and concentration of radioactive materials released to the environment can be reduced as low as reasonably achievable.

2. The radioactive liquid waste treatment systems and associated systems shall be designed to reflect preventive considerations against the leakage of liquid radioactive materials from the systems and uncontrolled release of those materials to the outside of the site.

(Radioactive solid waste treatment systems and storage system)

3. The treatment systems for radioactive solid wastes generated from nuclear reactor facilities shall be designed to reflect preventive considerations against the dispersion of radioactive material, etc.

4. The radioactive solid waste storage systems shall have enough capacity to store radioactive solid wastes generated by nuclear reactor facilities and be designed to reflect preventive considerations against the spread of contamination from the wastes.

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

[Detailed Notes on Requirements]

(Radioactive gaseous waste and radioactive liquid waste treatment systems)

A “that the quantity and concentration of radioactive materials released to the environment can be reduced as low as reasonably achievable” means to achieve it by filtration, storage, decay and control at gaseous waste treatment facility, by filtration, evaporation, ion-exchange, storage, decay and control at liquid waste treatment facility.

B “Can be reduced as low as reasonably achievable” refers to the design that, under the principle of “as low as reasonably available(ALARA)”, can achieve the dose objective <Note: 50 mSv/y> stipulated in Regulatory Guide for Annual Dose Target for the public in the Vicinity of Light Water Power Reactor Facilities (authorized by Nuclear Safety Commission on May 13th, 1975) as concerned nuclear power reactor facility.

C The above-mentioned evaluations of the dose objective shall be based on Regulatory Guide for Reviewing Evaluation of Dose Target for Surrounding Area of Light Water Nuclear Reactor Facilities (authorized by Nuclear Safety Commission on September 28th, 1976).

D “The radioactive liquid waste treatment systems” refers to such facility as to separate and collect liquefied radioactive waste including radioactive liquid waste with solid materials like sludge associated with the operation of nuclear reactor facility and, in accordance with the state of liquid waste, to carry out appropriate filtration, evaporation treatment, ion-exchange, storage, attenuation and control.

E “Associated systems” means the building or the area housing the treatment facility.

F As for “be designed to reflect preventive considerations against the leakage of liquid radioactive materials from the systems and uncontrolled release of those materials to the outside of the site”, it shall be based on Regulatory Guide for Fundamental Policy to be Considered in Reviewing of Liquid Radioactive Waste Treatment Facilities (authorized by Nuclear Safety Commission on September 28th, 1981).

(Radioactive solid waste treatment systems and storage system)

G “the dispersion of radioactive materials, etc.” includes scattering during the treatment processes of the waste such as crushing, compaction, incineration, solidification and others.

H “have enough capacity to store” means that it has the capacity which is able to store and control radioactive solid waste, considering the amount of radioactive solid waste generated in the nuclear reactor facility and shipped from it in the future.

3. Individual Systems of Reactor Facility

(10) Fuel Handling System

[Basic Requirements]

1. The storage and handling systems for fresh and spent fuels shall be designed so as to meet the following requirements.
   (1) The storage systems shall have appropriate containment and air purification systems.
   (2) The storage systems shall have appropriate storage capacity.
   (3) The handling systems shall have capability to prevent the dropping of fuel assemblies during its operation.

* The testability is arranged in the general requirements.

2. The storage and handling systems (except for dry storage casks) for spend fuels shall be designed so as to meet the following requirements, in addition to the aforementioned.
   (1) Proper shielding for radiation protection shall be implemented.
   (2) The storage systems shall have the system capable of fully removing decay heat and transporting it to an ultimate heat sink, and with its associated purification system.
   (3) Prevention of excessive decrease of cooling water inventory in the storage systems and proper leakage detection shall be possible.
   (4) The storage systems shall not lose their safety functions even in case of postulated dropping of fuel assemblies during handling.

3. The storage systems (if using dry storage casks) shall be designed so as to meet the following requirements, in addition to the first paragraph. (Excluding the requirement on the air clean up system, in the case that the lid part of the dry storage cask is not opened in the facility and it is possible to assure the containment of the radioactive materials by the dry storage cask):
   (1) Proper shielding for radiation protection shall be implemented.
   (2) The storage facilities shall appropriately remove decay heat.

(3) The spent fuel can confine the radioactive materials appropriately and the function can be monitored.

4. The fuel storage and handling systems shall be so designed that criticality can be prevented

5. The storage and handling systems for fresh and spent fuels shall be designed so that the water level and water temperature of the spent fuel storage facility except using dry storage casks, the radiation level in the fuel handling area and other abnormalities can be detected and that such a situation can be properly communicated to the site personnel or corrective measures can be automatically taken. Also, they shall be designed so that the situation of any events can be monitored through more than one parameter even in case of loss of off-site power.

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

[Detailed Notes on Requirements]

A A "dry storage cask" means a cask which dries inside after putting spent fuels in, enclosed with inert gasses and stores spent fuels. It consists of a main body, a lid part (doubly capped), a basket and other parts.

B "can be properly communicated to the site personnel" refers to that the monitoring is possible in the control room even if the access to the facilities is restricted in an emergency.

C The validity of the dry storage cask design shall be verified according to the instructions given in the guide "On-site storage of spent fuels at nuclear power stations using dry casks" (approved by NSC on Aug. 27, 1993, partly revised on March 29, 2002, and on September 19).

3. Individual Systems of Reactor Facility

(11) Radiation Management

(a) Environmental Radiation Protection (during normal operation)

[Basic Requirements] The nuclear reactor facilities shall be so designed that the dose rate around the site by direct and sky-shine gamma rays generated during normal operation can be reduced as low as reasonably achievable. (Corresponding to Guideline 56 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A. “reduced as low as reasonably achievable” in this regulation means that the design and the management of the facilities are to be carried out so that the target of not more than 50 μGy per annum in air kerma may be achieved under the concept of ALARA based on the “Dose assessment of the general public in the safety review of light water nuclear power reactor facilities” (approved by the Nuclear Safety Commission on March 27, 1989). It is not required to evaluate the radiation dose if it is designed and managed in this manner.

3. Individual Systems of Reactor Facility

(11) Radiation Management

(b) Radiation Protection and Control Facilities

[Basic Requirements]

(Radiation protection for radiation workers)

1. The nuclear reactor facilities shall be so designed as to reflect necessary considerations on radiation protection in order to fully reduce the dose in the areas accessible to radiation workers.

2. The nuclear reactor facilities shall incorporate radiation protection measures that will allow radiation workers to perform necessary operations during anticipated operational occurrences and design basis accidents.

(Radiation control for radiation workers)

3. The nuclear reactor facilities shall be provided with radiation control systems that adequately monitor and control radiation exposure in order to protect radiation workers.

4. The radiation control systems referred to in the preceding paragraph shall be so designed that necessary information can be displayed in the control room or in other appropriate places.

(Corresponding to Guideline 57 and 58 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

(Radiation protection for radiation workers)

A “be designed as to reflect necessary considerations on radiation protection” in paragraph 1 refers to the resign that necessary measures are taken regarding such as shielding, layout of equipment, remote control, prevention of leakage of radioactive materials and ventilation in view of the workability of radiation workers under the concept of ALARA.

(Radiation control for radiation workers)

B. “Radiation control systems” refers to the facilities for controlling the access of radiation workers, the control of contamination and decontamination, etc. in order to monitor and manage radiation exposure.

C. “necessary information can be displayed in the control room or in other appropriate places” means the display in the control room of the dose rates measured by the area radiation monitor necessary for radiation management and the display at an appropriate place of dose rates in radiation controlled areas, the concentration of radioactive materials in the air and the surface density of radioactive materials on the floors, etc., respectively.

3. Individual Systems of Reactor Facility

(11) Radiation Management

(c) Monitoring Equipment

[Basic Requirements]

The nuclear reactor facilities shall be designed to enable proper radiation monitoring and surveillance against the release of radioactive materials and measurement of the dose rate can be performed so that it allows necessary information to be displayed in the control room or in other appropriate places during normal operations, anticipated operational occurrences and design basis accidents.

(Corresponding to Guideline 59 of Regulatory Guide for Reviewing Safety Design of Light Water Nuclear Power Reactor Facilities)

[Detailed Notes on Requirements]

A “proper radiation monitoring and surveillance…and measurement” means the measurement and monitoring of the concentration of radioactive materials and the dose rates through sampling and monitoring of radiation in the atmosphere within the reactor containment vessel and the peripheral monitoring areas and the measurement and monitoring of appropriate places such as radiation sources, points of release, peripheries of power stations, and the expected route of release of radioactive materials so that prompt countermeasures may be taken in case of design basis accidents.

B The requirement to measure and monitor the environmental discharge of gaseous and liquid waste during normal operation shall be stipulated in the “Regulatory Guide for Radiation Monitoring of Effluent Released from Light Water Nuclear Power Reactor Facilities ” (authorized by the Atomic Energy Commission on September 29, 1978).

C The measurement and monitoring in case of a design basis accident shall be stipulated in the “Regulatory Guide for Regulatory Guide for Reviewing Radiation Monitoring in Accidents of Light Water Nuclear Power Reactor Facilities ” (authorized by the Atomic Energy Commission on July 23, 1981).

D The monitoring post shall be designed to be functionable until power supply is restored through an uninterruptible power supply system, etc. after on-site emergency power is lost. And the transmission system of the monitoring post shall be designed to have diversity.

3. Individual Systems of Reactor Facility

(12) Others

(a) Basic Requirements for Auxiliary Boilers

[Basic Requirements]

1 Auxiliary boilers shall be capable of supplying the required steam under the assumed conditions.

2 Auxiliary boilers shall be designed so as not to affect the safety of nuclear reactor facilities.

(*Auxiliary boiler which requires to establish new criteria .by integration with regulation of NPP by Electricity Utility Industry Law)

[Detailed Notes on Requirements]

A “Supplying the required steam” mean sufficiently supplying the steam required by SSCs with safety functions.

B “Does not affect the safety of nuclear reactor facilities” mean not to affect the safety of nuclear reactor facilities in case the auxiliary boilers are damaged.

4. Safety Assessment

(1) Safety Assessment

[Basic Requirements]

1. In order to verify that the basic policy of safety design of nuclear reactor facilities conforms to validity, the safety evaluation against anticipated operational occurrences and design basis accidents must be conducted.

2. It shall be verified that the safety evaluation referred to in the preceding paragraph pertaining to anticipated operational occurrences satisfy the pertinent criteria specified below.
   (1) The minimum critical heat flux ratio or the minimum critical power ratio shall be larger than the acceptable limit.
   (2) Fuel cladding shall not be mechanically damaged.
   (3) Fuel enthalpy shall not exceed the acceptable limit.
   (4) Pressure on the reactor coolant pressure boundary shall not exceed 110% of the maximum allowable working pressure.

3. It shall be verified that the safety evaluation referred to in paragraph 1 pertaining to design basis accidents satisfy the pertinent criteria specified below.
   (1) The core shall be without significant damage, and adequate coolable configuration of the core shall be maintained.
   (2) Fuel enthalpy shall not exceed the specified limit.
   (3) Pressure on the reactor coolant pressure boundary shall not exceed 120% of the maximum allowable working pressure.
   (4) Pressure on the reactor containment boundary shall not exceed the maximum allowable working pressure.
   (5) The accident shall not give significant radiological risk to the off-site public.

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

[Detailed Notes on Requirements]

A. "safety evaluation of Anticipated Operational Occurrences (AOOs) and Design Basis Accidents (DBA)" shall be conducted under the "Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities" and "Meteorological Guide for Safety Analysis of Nuclear Power Reactor Facilities" (authorized by the Nuclear Safety Commission on January 28, 1982)."