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Review Guide for Safety Design of Light Water Nuclear Power Reactor Facilities

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I. Introduction

This guide, "Review Guide for Safety Design of Light Water Nuclear Power Reactor Facilities" (hereinafter referred to as the "Review Guide for Safety Design"), sets forth the bases of review in the safety review of an installation permit (or a modification permit) for a light water nuclear power reactor facility (hereinafter referred to as "LWR facility") to determine the adequacy of the design from the viewpoint of ensuring safety.

The Review Guide for Safety Design, which is used in the safety review of an LWR facility, was originally instituted in April 1970 by the then Atomic Energy Commission. It was revised thoroughly in June 1977 also by the then Atomic Energy Commission. More than 10 years have passed since the first revision of the Guide. In the meantime, remarkable technological innovation and progress has been made. Useful experiences have also been accumulated through lessons learned from various events that took place at home and abroad such as the TMI accident in the U.S.A. In recognition of these, another set of thorough review and updating of the Guide has been made to provide more definite and systematic arrangements.

Concurrently with this revision, another review guide entitled the "Review Guide for Classification of Importance of Safety Functions for Light Water Nuclear Power Reactor Facilities," hereinafter referred to as the "Safety Importance Classification Guide," was set forth to provide guidance in determining the importance of safety functions of structures, systems and components of reactor facilities and to provide a method to apply this guide. Thus, the application of the Review Guide for Safety Design should make reference to the Safety Importance Classification Guide.

II. Position of the this guide and its scope of application

This guide, benefiting from the LWR experiences to date and the latest technological knowledge, establishes the fundamental principles for safety design that should be confirmed in the review process of the LWR installation permit and is not intended to establish general design standards for nuclear reactor facilities.

In this review process, it should be confirmed that the safety design of the subject reactor facility fully meets the requirements of this guide. Even if part of the safety design failed to comply with the requirements of this guide, however, the deviation is not necessarily ruled out insofar as the design incorporates technological improvement or advancement and is justifiable that the safety can be ensured as equal as or more than the safety that has met the requirement of this guide.

While this guide primarily addresses LWRs, it is believed that this guide will be helpful for the safety review of other types of nuclear reactor facilities.

III. Definitions of Terms

The following are definitions of the terms used in this guide.

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

1) Functions, when lost, that cause, abnormal conditions in nuclear reactor facilities and could potentially lead to undue radiation exposure of the public or site personnel.

2) Functions that prevent, in the case of occurrence of abnormal conditions in nuclear reactor facilities the escalation of such conditions or the immediate regaining of control of normal conditions, thereby preventing or mitigating potential undue radiation exposure of the public or site personnel.

(2) “Importance of safety functions” refers to the relative degree of the importance of safety functions from the viewpoint of ensuring the safety of nuclear reactor facilities.

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

(4) “Abnormal conditions” refer to those conditions in which nuclear reactor facilities suffer such disturbances as may cause them to deviate from normal operations. Anticipated operational occurrences and accidents fall within this category.

(5) “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 misoperations or by disturbances with a similar probability of occurrence.

(6) “Accidents” refer to those conditions beyond “anticipated operational occurrences”, which have quite small probabilities of occurrence and yet are postulated in the light of the safety design of nuclear reactor facilities.

(7) “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.

(8) “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 under abnormal conditions and which cause loss of reactor coolant when damaged.

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

(10) “Reactor cooling system" refers to those systems which are used to remove heat from the reactor during normal operation or abnormal conditions (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.).

(11) “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.

(12) “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, burnup, fission products, etc.

(13) “Safety protection system” refers to those provisions which are designed to detect abnormal conditions of nuclear reactor facilities and directly initiate the operation of the reactor shutdown systems, engineered safety features and other systems as required.

(14) “Engineered safety features” refer to those provisions which are designed to prevent or limit the possible heavy release of radioactive materials to the environment by fuel damage, etc. resulting from damage or failure in nuclear reactor facilities.

(15) “Single failure” refers to the loss of intended safety functions of a component by a single cause. Multiple failures due to secondary causes are included in this category.

(16) “Active component” refers to a component that performs necessary functions actively depending on an external input.

(17) “Redundancy” represents the existence of two or more systems or components with identical attributes to perform an identified function.

(18) “Diversity” represents the existence of two or more systems or components with different attributes to perform an identified function.

(19) “Independence” represents the freedom of two or more systems or components from simultaneous functional impediment caused by common or subordinate factors under design-based environmental and operational conditions.

(20) “Acceptable fuel design limits” refer to the limits of safety tolerance for reactor fuel damage, set forth in association with the reactor design, within which the reactor can be

allowed to continue its operation.

IV. General Requirements for Nuclear Reactor Facilities

G1. Applied Codes and Standards

(In the English version, Guiding Principle is referred to as G)

The design, selection of materials, fabrication and inspection of structures, systems and components with safety functions shall conform to those codes and standards which are recognized as appropriate in the light of the importance of their safety functions.

G2. Design Considerations Against Natural Phenomena

(1) Structures, systems and components 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 the most appropriate design ground motion.

(2) Structures, systems and components with safety functions shall be so designed that the safety of the nuclear reactor facilities will not be impaired by other postulated natural phenomena other than earthquake.

Structures, systems and components 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 accident-induced loads.

G3. Design Considerations Against External Man-induced Events

(1) Structures, systems and components with safety functions shall be so designed that the safety of the nuclear reactor facilities will not be impaired by postulated external man-induced events.

(2) The nuclear reactor facilities shall be so designed that structures, systems and components with safety functions are protected by appropriate means against any unjustifiable access by third persons.

G4. Design Considerations Against Internal Missiles

Structures, systems and components 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.

G5. Design Considerations Against Fire

The nuclear reactor facilities shall be so designed that their safety will be protected against fire by an appropriate combination of three measures of fire prevention, fire detection and extinguishment and mitigation of fire effects.

G6. Design Considerations Against Environmental Conditions

Structures, systems and components with safety functions shall be designed to withstand all the environmental conditions under which their safety functions are expected.

G7. Design Considerations for Shared Use

Structures, systems and components with safety functions shall be so designed that in case they are shared by two or more nuclear reactor facilities, the safety of the reactors will not be impaired by the shared use.

G8. Design Considerations Against Operator's Actions

The nuclear reactor facilities shall be designed to reflect appropriate preventive considerations against operators' errors.

G9. Design Considerations for Reliability

(1) Structures, systems and components 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 with redundancy or diversity and independence considering their physical make–up, working principles, assigned safety functions, etc.

(3) The systems referred to in item (2) above shall be designed to be capable of fulfilling their safety functions even in case of unavailability of off-site power in addition to an assumption of a single failure of any of the components that comprise the systems.

G10. Design Considerations for Testability

Structures, systems and components with safety functions shall be designed to be capable of being tested or inspected to verify their integrity and capability by adequate methods consistent with the importance of their safety functions during reactor operation or shutdown.

V. Reactor and the Reactor Shutdown System

G11. Reactor Core Design

(1) The reactor core shall be designed to assure, with the aid of the functions of associated reactor cooling system, reactor shutdown 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 proximity to it within the reactor pressure vessel shall be designed to be capable of ensuring safe reactor shutdown and proper core cooling during normal operation and abnormal conditions.

G12. Fuel Design

(1) The fuel assemblies shall be designed so as not to lose their integrity despite various unfavorable factors that may take place during their use in the reactor.

(2) The fuel assemblies shall be designed so as not to be excessively deformed during transport or handling.

G13. Reactor Characteristics

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

G14. 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 maintaining the necessary operational conditions.

(2) The maximum reactivity worth of control rods and reactivity insertion rate shall be such that postulated reactivity-initiated events will not result in damage of the reactor coolant pressure boundary nor in destruction of the core, core support structures and reactor pressure vessel internals that may impair core cooling.

G15. Independence and Testability of Reactor Shutdown System

The reactor shutdown 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 maintaining the core subcritical under hot conditions. They shall also be designed to allow testing with respect to their functional capability・

G16. Reactor Shutdown Margin by Control Rods

The control rod-dependent system out of the reactor shutdown system shall be designed to be capable of making the core subcritical under hot and cold conditions even when one control rod with maximum reactivity worth is withdrawn out of the core and cannot be inserted.

G17. Shutdown Capability of Reactor Shutdown System

(1) At least one independent system out of the reactor shutdown system shall be

designed to be capable of making the core subcritical under hot conditions during normal operation and anticipated operational occurrences without leading to the acceptable fuel design limits being exceeded and capable of maintaining the core subcritical under hot conditions.

(2) At least one independent system out of the reactor shutdown system shall be designed to be capable of making the core subcritical under cold conditions and of maintaining the core subcritical under cold conditions.

G18. Reactor Shutdown System Capability in Case of Accident

At least one independent system included in the reactor shutdown system shall be designed to be capable of making the core subcritical in case of an accident, and at least one independent system included in the reactor shutdown system shall be designed to be capable of maintaining the core subcritical in case of an accident.

VI. Reactor Cooling System

G19. Integrity of Reactor Coolant Pressure Boundary

(1) The reactor coolant pressure boundary shall be so designed that its integrity will be ensured during normal operation and abnormal conditions.

(2) The pipelines connected to the reactor coolant system shall in general be fitted with isolation valves.

G20. Prevention of Reactor Coolant Pressure Boundary Failure

The reactor coolant pressure boundary shall be designed not to exhibit brittle behavior and develop any quickly propagating failure during normal operation, maintenance, testing and abnormal conditions.

G21. Detection of Reactor Coolant Pressure Boundary Leaks

The means shall be provided for quick and proper detection of the leakage of the reactor coolant, if any, from the reactor coolant pressure boundary.

G22. In-Service Test and Inspection of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed to be capable of being tested and inspected to verify its integrity throughout the service life of the nuclear reactor.

G23. Reactor Coolant Supply System

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

G24. Systems for Removing Residual Heat

(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 the acceptable fuel design limits and design conditions for the reactor coolant pressure boundary from being exceeded.

(2) The systems for removing residual heat shall be properly provided with redundancy or diversity and independence so that they can fulfill their safety functions even in case of unavailability of off-site power in addition to an assumption of a single failure of any of the components that comprise the systems. They shall also be designed to allow testing with respect to their functional capability.

G25. Emergency Core Cooling System

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

(2) The emergency core cooling system shall be designed with redundancy or diversity and independence so that the system can fulfill its safety functions even in case of unavailability of off-site power in addition to an assumption of a single failure of any of the components that comprise the system.

(3) The emergency core cooling system shall be designed to be capable of being tested and inspected on a periodical basis. The emergency core cooling system shall also be designed to allow testing and inspection of each constituent system independently so that the integrity and redundancy of the emergency core cooling system can be verified.

G26. Systems for Transporting Heat to Ultimate Heat Sink

(1) The systems for transporting heat to an ultimate heat sink shall be designed to be capable of transferring heat generated or accumulated in structures, systems and components 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 properly provided with redundancy or diversity and independence so that they can fulfill their safety functions even in case of unavailability of off-site power in addition to an assumption of a single failure of any of the components that comprise the systems. They shall also be designed to allow testing with respect to their functional capability.

G27. Design Considerations Against Loss of Power

The nuclear reactor facilities shall be designed that safe shutdown and proper cooling of the reactor after shutting down can be ensured incase of a short-term total AC power loss.

VII. Reactor Containment

G28. Functions of Reactor Containment

(1) The reactor containment shall be designed to withstand the load (pressure, temperature, dynamic load) resulting from postulated events for reactor containment design and an appropriate seismic load and prevent the specified leakage rate from being exceeded with the aid of properly operating isolation functions.

(2) The reactor containment shall be so designed that the leakage rate of the entire containment can be measured under a specified pressure on a periodical basis.

(3) The reactor containment shall be designed to allow leakage tests at such important portions as penetrations for electric cables, pipelines, etc. and access openings.

G29. Prevention of Containment Boundary Reactor Containment Boundary Failure

The reactor containment boundary shall be designed not to exhibit brittle behavior and develop any quick propagating failure during normal operation, maintenance, testing or abnormal conditions.

G30. Isolation Function of Reactor Containment

(1) The pipelines that penetrate the reactor containment walls shall in general be fitted with containment isolation valves.

(2) The containment isolation valves to be fitted in principal pipelines shall in general be designed to be automatically and properly closed in case of an accident that necessitates the retention of isolation function.

G31. Reactor Containment Isolation Valves

(1) The containment isolation valves shall be located as close to the reactor containment as practicable.

(2) The installation of the containment isolation valves shall be subject to the following:

1) Of the pipelines that open inside the reactor containment or communicate with the reactor coolant pressure boundary, those which are not closed outside the reactor containment shall in general be provided with one containment isolation valve inside the reactor containment and one outside.

2) Of pipelines other than 1) above, those which are closed inside or outside the reactor containment shall in general be provided with one containment isolation valve outside the reactor containment.

3) The containment isolation valves shall not lose their isolation function due to loss of driving power after they are closed.

4) The containment isolation valves shall allow performance tests to be conducted on a periodical basis, of which important ones shall be testable for leakage.

G32. Reactor Containment Heat Removal System

(1) The reactor containment heat removal system shall be designed to sufficiently reduce the containment pressure and temperature resulting from the release of energy in case of postulated events for reactor containment design.

(2) The reactor containment heat removal system shall be designed with redundancy or diversity and independence so that the system can fulfill its safety functions even in case of unavailability of off-site power in addition to an assumption of a single failure of any of the components that comprise the system. The system shall also be designed to allow testing with respect to its functional capability.

G33. Systems for Controlling Containment Facility Atmosphere

(1) 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 postulated events for reactor containment design.

(2) The flammable 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.

(3) The systems for controlling containment facility atmosphere shall be designed with redundancy or diversity and independence so that they can fulfill their safety functions even in case of unavailability of off-site power in addition to an assumption of a single failure of any of the components that comprise the systems. They shall also be designed to allow testing with respect to their functional capability.

VIII. Safety Protection System

G34. Redundancy of Safety Protection System

The safety protection system shall be designed with redundancy so that any single failure of any of the components or channels which comprise the system or removal from service of any component or channel does not result in loss of safety functions of the system.

G35. Independence of Safety Protection System

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 and abnormal conditions.

G36. Function of Safety Protection System During Transient

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.

G37. Function of Safety Protection System in Case of Accident

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

G38. Function of Safety Protection System in Case of Failure

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.

G39. Separation of Safety Protection System from Instrumentation and Control Systems

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

G40. Testability of Safety Protection System

The safety protection system shall be designed to be capable of being tested in general during reactor operation on a periodical basis and shall allow testing of each constituent channel independently so that the integrity and redundancy of the system can be verified.

IX. Control Room and Emergency Provisions

G41. Control Room

The control room shall be so designed that the operating status and principal parameters of reactor and principal related facilities can be monitored and that prompt manual control can be performed, whenever required, to maintain safety.

G42. Reactor Shutdown Function from Outside of Control Room

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) To have 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) To have the maintenance of cold shutdown state of the reactor by appropriate control procedure.

G43. Design Considerations for Control Room Protection

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 geseous radioactive materials likely to be released due to fire or accident by means of a proper ventilation system.

G44. Emergency Station in Nuclear Power Plant

The nuclear reactor facilities shall be designed to allow installation, in the nuclear power plant, of an emergency station from which necessary instructions will be furnished in case of an accident.

G45. Design Considerations for Communication Means

The nuclear reactor facilities shall be Provided with adequate warning systems and communication means that allow necessary instructions and messages to be given properly to all the people present in the nuclear power plant in case of an accident. The communication means between nuclear power plant and necessary outside points shall be Provided with redundancy or diversity.

G46. Design Considerations for Escape Routes

The nuclear reactor facilities shall be provided with evacuation lighting that functions even in case of ordinary lighting power loss and shall have safe escape routes provided with concise and permanent guide signs.

X. Instrumentation and Control Systems and Electrical Systems

G47. Instrumentation and Control Systems

(1) The instrumentation and control systems shall be designed with adequate considerations for 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 the appropriate predicted limits.

2) Monitoring of the aforementioned parameters within predicted variation limits shall be possible so as to allow necessary measures to be taken as required.

(2) The instrumentation and control systems shall be designed to enable monitoring, and recording as required, of the parameters necessary to recognize the status of an accident and take measures according to an adequate method over sufficient range in case of an accident. The systems shall also be designed to enable monitoring or estimation of the status of reactor shutdown and core cooling in particular by use of two or more kinds of parameters.

G48. Electrical Systems

(1) The electrical systems shall be designed to allow the structures, systems and components with safety functions of especially high importance to be fed by either the off-site power or the emergency on-site power when they need electric power to fulfill their safety functions.

(2) The off-site power system shall be connected to the electric power system with two or more power transmission lines.

(3) The emergency on-site power system shall incorporate redundancy or diversity and independence and have enough capacity and capability to accomplish the following properly even with the assumption of a single failure of its components.

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 safety functions of other necessary systems and components in case of an accident, such as loss of reactor coolant.

(4) The electrical systems associated with safety functions of high importance shall be designed such that their important portions can be tested and inspected on a periodic basis.

XI. Fuel Handling Systems

G49. Fuel Storage and Handling Systems

(1) The storage and handling systems for fresh and spent fuels shall be designed so as to meet the following requirements.

1) Appropriate periodical testing and inspection of structures, systems and components with safety functions shall be possible.

2) The storage systems shall have appropriate containment and air purification systems.

3) The storage systems shall have appropriate storage capacity.

4) The handling systems shall have capability to prevent the dropping of fuel assemblies during transit operation.

(2) The storage and handling systems 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 available.

2) The storage systems shall have the system capable of fully removing decay heat and transporting it to an ultimate heat sink with an 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.

G50. Fuel Criticality Prevention

The fuel storage and handling systems shall be so designed that criticality can be prevented in any postulated case by use of a geometrical safety layout or other appropriate means.

G51. Monitoring of Fuel Handling Area

The fuel handling area shall be so designed that the state of events leading to the loss of decay heat removal capability and excessive radiation levels can be detected and that such a state can be properly communicated to the site personnel or corrective measures can be automatically taken against such a state.

XII. Radioactive Waste Treatment Systems

G52. Radioactive Gaseous Waste Treatment Systems

The treatment systems for radioactive gaseous waste generated with the reactor operation shall be so designed that the quality and concentration of radioactive materials released to the environment can be reduced as low as reasonably achievable through proper filtration, retention, attenuation, management, etc.

G53. Radioactive Liquid Waste Treatment Systems

(1) The treatment systems for radioactive liquid waste generated with 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 through

proper filtration, evaporation process, ion exchange, retention, attenuation, management, etc.

(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 outside the site.

G54. Radioactive Solid Waste Treatment Systems

The treatment systems for radioactive solid waste generated from nuclear reactor facilities shall be designed to reflect preventive considerations against the dispersion of radioactive materials in the process of crushing, compression, burning, solidification, etc. of the radioactive waste.

G55. Radioactive Solid Waste Storage Systems

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

XIII. Radiation Control

G56. Radiation Protection in the Vicinity of the Site

The nuclear reactor facilities shall be so designed that the dose rate by direct and skyshine gamma rays generated during normal operation around the site can be reduced as low as reasonably achievable.

G57. Radiation Protection for Radiation Workers

(1) The nuclear reactor facilities shall be designed so as to reflect necessary considerations for radiation protection in order to reduce the dose equivalent rate in the areas accessible to radiation workers as low as reasonably achievable by means of shielding, component layout, remote handling, prevention of the leakage of radioactive materials, ventilation, etc., taking the work efficiency of radiation workers into account.

(2) The nuclear reactor facilities shall incorporate radiation protection measures that will allow radiation workers to perform necessary operations during abnormal conditions.

G58. Radiation Control for Radiation Workers

The nuclear reactor facilities shall be provided with radiation control systems that adequately monitor and control radiation exposure in order to protect radiation workers from radioactivity. The radiation control systems shall be designed so that necessary information can be displayed in the control room or in other appropriate places.

G59. Radiation Monitoring

The nuclear reactor facilities shall be designed to enable proper radiation monitoring over at least reactor containment atmosphere the environmental monitoring area surrounding the nuclear reactor facility and release paths of radioactive materials and to allow necessary information to be displayed in the control room or in other appropriate places.

Commentary

In order to ensure that this guide is applied properly, specific explanations below are provided pertaining to what needs particular attention in the Practical usage of the individual requirements and what is deemed necessary to further clarify with respect to their significance and interpretation.
The items in the text of the guide chosen for specific explanations are here listed below.

III. Definitions of Terms

(1) Safety functions
(2) Importance of safety functions
(6) Accidents
(7) Reactor containment boundary
(8) Reactor coolant pressure boundary
(13) Safety protection system
(14) Engineered safety features
(15) Single failure
(18) Diversity
(19) Independence
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III. Definitions of Terms

(1) "Safety Functions"

Structures, systems and components with "safety functions" are specified separately in the "Safety Importance Classification Guide."

(2) "Importance of Safety Functions"

The importance of safety functions is specified separately in the "Safety Importance Classification Guide."

(6) "Accidents"

"Postulated" signifies that a certain event is believed to take place with a probability to be taken into consideration from the view point of safety design of nuclear reactor facilities. This interpretation also applies to "postulated missiles," "a postulated single failure of any passive component," etc. mentioned in this guide.

(7) "Reactor Containment Boundary"

"The postulated events for reactor containment design" are the events which are postulated for the purpose of evaluating the adequacy of the reactor containment design and which encompass the severest conditions with respect to parameters in question on the basis of the evaluation of pressure rise, temperature rise, dynamic load, flammable gas generation and concentration of radioactive materials that have potential to impede the maintenance of the functions of the reactor containment. For further details, refer to the "Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities."

(8) "Reactor Coolant Pressure Boundary"

The reactor coolant pressure boundary includes the components and pipelines within and limited to the following ranges.

1) Reactor pressure vessel and associated items (those which can be directly attached to the body, such as the control rod drive mechanism housing).

2) Components and pipelines that comprise the reactor coolant system; they include the primary coolant pump, such components as water chamber, tube plates, tubes and pressurizer of steam generator, primary cooling line, valves, etc. in a PWR, and main steam and feedwater pipelines up to and including the second isolation valves as viewed from the reactor side in a BWR.

3) Connecting pipelines

(a) Pipelines provided with valves that are normally open and closed in case of accident up to and including the second isolation valves as viewed from the reactor.

(b) Pipelines provided with valves that are normally closed and opened in case of an accident up to and including the first isolation valves as viewed from the reactor.

(c) The above (a) also applies to the emergency core cooling system, etc.

provided with valves that are normally closed and opened in case of loss of reactor coolant.

The isolation valves mentioned above refer to automatic isolation valves, check valves, normally locked shut-off valves or remote-controlled shut-off valves.

(13) “Safety Protection System”
The safety protection system includes signal circuits that actuate reactor shutdown systems in case of emergency and signal circuits that actuate engineered safety features, etc. and either system extends from the detectors to the input terminals of the operating devices.

(14) “Engineered Safety Features”
The engineered safety features include emergency core cooling system, reactor containment (including isolation valves), containment facility atmosphere cleanup system, etc.

(15) “Single Failure”
“Secondary cause” refers to a factor necessarily generated by a single cause.

(18) “Diversity”
“Different attributes” mean that the mode of total or partial loss of necessary function is not the same.

(19) “Independence”
“Common factor” is a factor that simultaneously affects two or more systems or components; influential factors induced by environmental temperature, humidity, pressure, radiation, etc. and those induced by electric power, air, oil, cooling water, etc. supplied to systems and components are common factors.

(20) “Acceptable Fuel Design Limits”
“The reactor can be allowed to continue its operation” does not necessarily mean that the reactor is operated from the state in which fuel is damaged but includes the resumption of reactor operation after the problem is remedied with fuel which is inspected or replaced where necessary. Maximum temperature of fuel pellets, maximum temperature of fuel cladding, maximum heat flux, minimum critical heat flux ratio, minimum critical power ratio, maximum enthalpy of fuel pellets, maximum amount of deformation of fuel cladding, etc. serve as the bases of judgment to determine the acceptable fuel design limits.

IV. General Requirements for Nuclear Reactor Facilities

G1. Applied Codes and Standards

The design, selection of materials, fabrication and inspection of structures, systems and components 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.

By saying "...shall conform to those codes and standards", the guide implies that it is necessary to identify the proper codes and standards on which the design, selection of materials, fabrication and inspection of structures, systems and components in question shall be based.

G2. Design Considerations Against Natural Phenomena

The interpretation of "...designed to sufficiently withstand the most appropriate design ground motion" is subject to the "Review Guide for Seismic Design of Nuclear Power Reactor Facilities."

The phrase "... so designed that the safety of the nuclear reactor facilities will not be impaired by ...natural phenomena" means that the structures, systems and components involved shall be designed to maintain their capability of fulfilling their safety functions even in the case that they meet with natural phenomena against which design considerations are required, individual or combined.

"Structures, systems and components with safety functions of especially high importance" are separately specified in the "Safety Importance Classification Guide".

"Anticipated natural phenomena" refer to on-site natural phenomena possible to occur including flood, tsumani, strong wind, freezing, snowing, landslide, etc.

"The severest conditions" refer to the conditions not less severe than the past records of the natural phenomena in question that are considered to be reliable and statistically reasonable.

An assumption of simultaneous occurrence of different natural phenomena should be taken into consideration if deemed necessary upon the review of past records or field investigation results.

"Appropriate combinations of natural forces and accident-induced loads" do not necessarily mean the simple addition of maximum accident-induced load to natural force considered to be the severest of all but their combination in the proper way with the relation of their cause and effect and their change with time taken into account.

G3. Design Considerations Against External Man-Induced Events

"External man-induced events" as referred to in the guide include airplane crash, breaking of dam, explosion, etc.

G4. Design Considerations Against Internal Missiles

The "internal missiles" refer to flying matter as a result of the breaking of valves and pipelines containing fluid with high internal energy, damage of high-speed rotating equipment, a gas explosion, a heavy equipment fall, etc. The design considerations shall take into account the secondary impacts of secondary missiles, fire, flood, chemical reaction, electrical damage, pipe rupture and equipment breakdown that may result from the above incidents.

G5. Design Considerations Against Fire

"Designed that their safety will be protected against fire" refers to the design conforming to "Guide for Fire Protection in Light Water Nuclear Power Reactor Facilities".

G6. Design Considerations Against Environmental Conditions

"Environmental conditions under which their safety functions are expected" refer to all the environmental conditions to which the structures, systems and components whose functions are expected to work during normal operations and abnormal conditions may possibly be exposed throughout their service life.

G7. Design Considerations for Shared Use

"...So designed that ...the safety of the reactors will not be impaired ..." as mentioned here means that the safety functions required under abnormal conditions are not impeded by shared use of structures, systems and components, that, under abnormal conditions involving one of the reactors in concern, shutting down and removing residual heat of the other reactors) can be achieved, and that a possible failure of any of the shared structures, systems and components will not cause an accident involving two or more reactors at a time.

G8. Design Considerations Against Operator's Actions

"Designed to reflect appropriate preventive considerations ..." refers to the design that reflects human engineering-oriented considerations to ensure efficient panel layout, operational efficiency of apparatus and valves, capability of monitoring reactor status accurately and quickly through meters, lamps and alarms and prevention of errors during maintenance and inspection, and that allows the necessary safety functions to be maintained without the operator's actions being expected for a certain length of time after the occurrence of an abnormal condition.

G9. Design Considerations for Reliability

"...Adequately high reliability ...as required according to the importance of their safety functions" and "systems with safety functions of especially high importance" are specified separately in the "Safety Importance Classification Guide."

"Single failure" is categorized into two kinds i.e., single failure of active component and single failure of passive component.

Systems with safety functions of especially high importance shall be designed so that they can fulfill their expected safety functions even with an assumption of a single failure of any active component during a short term and with an assumption of either as a single failure of any active component or a postulated single failure of any passive component during a long term. In evaluating the long-term safety functions for which either a single failure of any active component or a postulated single failure of any passive component is to be assumed, the assumption of a single failure in particular components can be exempted if it is assured that such a single failure can be removed or remedied within a period of time not being detrimental to safety.

G10. Design Considerations for Testability

"Adequate methods" include the use of testing bypass systems in a case test or inspection using systems in actual service is inadequate.

V. Reactor and Reactor Shutdown System

G12. Fuel Design

"Various 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 changing load, chemical effects, static and dynamic loads, deformation of fuel pellet, composition change in fuel rod filler gas.

G13. Reactor Characteristics

"Have inherent characteristics to suppress the reactor power rise" means that the reactivity feedback as an inherent total effect of Doppler coefficient, moderator temperature coefficient, moderator void coefficient, pressure coefficient, etc. which quickly works to suppress the reactor power rise during power transient over all operational range and thus prevent or mitigate fuel damage.

"Be well capable of controlling reactor power oscillation if it occurs" means that adequate attenuation characteristics or controllability against reactor power oscillation is provided so that the acceptable fuel design limits are not exceeded.

G14. Reactivity Control System

In evaluating the "maximum reactivity worth of control rods," the effects of reactivity controlling device that limits the insertion and arrangement of control rods in conjunction with the operational conditions of the reactor, if available, may be taken into account.

"Postulated reactivity-initiated event" refers to an event in which abnormal reactivity insertion takes place in the reactor, which is defined in the "Review Guide for Safety Evaluation of Light Water Nuclear Power Reactor Facilities" and the "Guide for Evaluation of Reactivity Initiated Events in Light Water Nuclear Power Reactor Facilities."

G15. Independence and Testability of Reactor Shutdown System

The control rod-dependent system and the soluble poison-dependent system (boric acid injection system in BWRs, boric acid injection system of the chemical and volume control system in PWRs) employed in present LWRs are considered to conform to paragraph 15 with respect to their performance as the reactor shutdown system.

In case a control rod-dependent system itself has multiple independent shutdown functions that are sufficient enough in number for a hot shutdown, such functions may be regarded as multiple independent shutdown systems practically.

“Capable of maintaining the core subcritical under hot conditions” refers to the capability of maintaining the subcritical state of the core during a Period from the end of a transient state to the addition of reactivity due to xenon decay and means that the operation of other systems, if any, may be expected to maintain the core subcritical beyond that period. In case the operation of other systems capable of reactor shutdown can be expected depending on the kind of event, their contribution may be taken into account in the design considerations.

G17. Shutdown Capability of Reactor Shutdown System

“Capable of making the core subcritical under cold conditions and of maintaining the core subcritical under cold conditions" refers to achieving a cold subcritical state from a hot subcritical state and maintaining it while compensating for reactivity addition due to xenon decay and reactor coolant temperature change.

G18. Reactor Shutdown System Capability in Case of Accident

In case the operation of any other system capable of shutting down the reactor can be expected together with the reactor shutdown system at the time of an 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 reactor shutdown system in making and maintaining the core subcritical in the event of a main steam pipe rupture in a PWR.

VI. Reactor Cooling System

G19. Integrity of Reactor Coolant Pressure Boundary

“...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.

“Be fitted with isolation valves” refers to the reactor coolant Pressure boundary design in which the pipelines connected to the reactor coolant system and forming the boundary in part are fitted with appropriate isolation valves considering the service conditions and purposes of those pipelines during normal operations so that loss of reactor coolant can be stopped in case of abnormal leakage from the portions not forming the reactor coolant pressure boundary.

G23. Reactor Coolant Supply System

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 the feed water system) in a BWR; charging Pump-aided supply system in a PWR).

“A limited leakage” refers to coolant leakage through seals of valves or pumps that make up the reactor coolant pressure boundary or through small cracks in the reactor coolant pressure boundary.

G24. Systems for Removing Residual Heat

The “systems for removing residual heat” refer to the systems provided so as to remove residual heat even in case heat removal is unachievable by main condenser （reactor core isolation cooling system, residual heat removal system, high pressure core spray system, automatic depressurization system, etc. in a BWR; steam generator, main steam relief valve, main steam safety valve, auxiliary feedwater system, residual heat removal system, etc. in a PWR).　In association with these, the BWR has a main steam relief safety valve to reduce pressure in the reactor coolant system, and PXVR has pressurizer relief valve, etc. for the same purpose.

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

“Properly provided with ...”means that redundancy or diversity and independence are required for achieving the functions of the system during abnormal conditions.

G25. Emergency Core Cooling System

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

G26. Systems for Transporting Heat to Ultimate Heat Sink

“Ultimate heat sink" refers to the sea, river, pond, lake or open air.

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

“Properly provided with ....” means that redundancy or diversity and independence are required for achieving the functions of the systems during abnormal conditions.

G27. Design Considerations Against Loss of Power

No particular considerations are necessary against long-term total AC power loss because the repair of troubled power transmission line or emergency AC power system can be expected in such case.

The assumption of total AC power loss is not necessary if the emergency AC power system is reliable enough by means of system arrangement or management (such as maintaining the system in operation at all times).

VII. Reactor Containment

G28. Functions of Reactor Containment

"Such important portions as penetrations for electric cables, pipelines, etc. and access openings" refer to significant portions from the viewpoint of leakage. Such examples are penetrations and access openings utilizing resilient seals or expansion bellows.

G30. Isolation Function of Reactor Containment

"Containment isolation valves" 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 the necessary isolation function by means of gravity, etc. even in a case in which the concerned pipeline is damaged either inside or outside the reactor containment and the back pressure to the valves is totally lost as a result.

"...In general be fitted with containment isolation valves" means that the pipelines for sampling or instrumentation important to reactor safety, the hydraulic pipelines for control rod drive, or other similar pipelines need not be provided with containment isolation valves if the leakage through those pipelines is as small as tolerable.

"Principal pipelines" refer to the pipelines 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 for those pipelines whose containment isolation valves are closed during hot operation.

"...in general 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 of unavailability of off-site power in addition to an assumption of a single failure.

The meaning of “in general” as mentioned here is that those pipelines which belong to principal pipelines but are part of the systems necessary to control accidents are not required to be shut off by automatic isolation signals in order to save such systems’ safety functions. Even in such a case, however, the loss of the containment isolation function shall be prevented.

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

G31. Reactor Containment Isolation Valves

“Those which are not closed outside the reactor containment” are the pipelines which will make intolerable release paths of radioactive materials in the atmosphere within the reactor containment to the outside depending on the conditions of the pipelines during an accident if the isolation is not applied.

“...In general be provided with one containment isolation valve inside the reactor containment and one outside” means that the design with two isolation valves outside the reactor containment can be accepted if the design is justifiable from the viewpoint of adequate safety considerations other than reactor containment isolation function.

“...In general be provided with one containment isolation valve outside the reactor containment” means that the design with one isolation valve either inside or outside the reactor containment can be accepted if the pipelines do not communicate to the outside of the reactor containment considering their functional state.

G32. Reactor Containment Heat Removal System

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 postulated events for reactor containment design, and includes the reactor containment spray system and its heat removal system.

G33. Systems for Controlling Containment Facility Atmosphere

The “systems for controlling containment facility atmosphere” refers to the containment facility atmosphere cleanup system and the flammable gas concentration control system.

The “containment facility atmosphere cleanup system” includes the standby gas treatment system, standby recirculation gas treatment system and containment spray system in the case of a BWR; annulus air recirculation system and containment spray system in the case of a PWR.

“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 a level below the combustion limit by means of a recombiner etc., if necessary.

VIII. Safety Protection System

G34. Redundancy of Safety Protection System

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 the safety protection action, and covers a range from a detector to logic circuit input terminals.

G35. Independence of Safety Protection System

“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 and its safety function will not be affected.

G36. Function of Safety Protection System During Transient

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 a given level or increasing too fast.

G38. Function of Safety Protection System in Case of Failure

“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.

“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 unrepaired.

G39. Separation of Safety Protection System from Instrumentation and Control Systems

“...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, misoperation or removal from service, the safety protection system with its functions not being impaired can fulfill the requirements in paragraphs 34 through 38.

G40. Testability of Safety Protection System

“Capable of being tested ...during reactor operation on a periodic basis” means that the safety protection system can be tested to verify the maintenance of its proper functions at appropriate time intervals during reactor operation and that even during the in-operation functional verification test, the functions themselves are maintained without unnecessary actuation of the reactor shutdown system, emergency core cooling system, etc. being caused.

IX. Control Room and Emergency Provisions

G41. Control Room

"Principal parameters ...can be monitored" means that, of the parameters required to be monitored under paragraph 47 are those which need to be monitored on a continuous basis can be monitored in the control room.

"Prompt manual control" means the operations necessary to shut down the reactor and maintain the reactor cooling after shutting down.

G42. Reactor Shutdown from Outside of Control Room

"...Allow reactor to be shut down from an appropriate location outside the control room" means that appropriate measures are taken in the event where access to the control room is prevented for some reason.

"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.

G43. Design Considerations for Control Room Protection

"Site personnel to have access to or stay in the control room" means that such passages as will allow site personnel to have access to the control room for necessary operations and are allowed to stay in there for a proper length of time after an accident takes place, and that radiation protection measures which will become feasible with time and attenuation of radiation level can be properly taken for the site personnel who approach the control room for the shift after immediate operations are accomplished.

X. Instrumentation and Control Systems and Electrical Systems

G47. Instrumentation and Control Systems

"Parameters necessary to maintain the integrity ...." include neutron flux of the reactor core, neutron flux distribution, reactor water level, 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.

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

"Recording" includes the availability of the information about necessary parameters following the course of an accident.

G48. Electrical Systems

"Off-site power system" refers to a series of provisions used to supply power to the nuclear reactor facility from an external electric power system or main power generator.

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

"Safety functions of especially high importance" and "safety functions of high importance" are specified separately in the "Safety Importance Classification Guide."

XII. Radioactive Waste Treatment Systems

G52. Radioactive Gaseous Waste Treatment Systems

The radioactive gaseous and liquid waste treatment systems must be so designed that the dose equivalent rate to the public can be maintained as low as reasonably achievable and therefore should meet the requirements separately specified in the "Guide for Dose Objective around Light Water Nuclear Power Reactor Facilities."

G53. Radioactive Liquid Waste Treatment Systems

"Radioactive liquid waste treatment systems" are the systems in which radioactive liquid waste generated with reactor operation together with radioactive waste in liquid form containing such solids as sludge are separated, collected and properly treated through filtration, evaporation, ion exchange, retention, attenuation, etc. depending on the properties of the waste.

"Associated systems" refer to the buildings or areas that accommodate treatment systems.

Detailed requirements as to being "designed to reflect preventive considerations against leakage of liquid radioactive materials from the systems and uncontrolled release of those materials to the outside of the site" is separately specified in "Considerable Items or Fundamental Principles for Licensing Review of Radioactive Liquid Waste Treatment Systems."

XIII. Radiation Control

G58. Radiation Control for Radiation Workers

"Necessary information can be displayed in the control room or in other appropriate places" means that the dose rate readings of an area radiation monitor necessary for radiation control can be displayed in the control room and that the dose rate, concentrations of radioactive materials in the air and surface densities of radioactive materials on the floor and elsewhere in radiation controlled areas can be displayed in appropriate places.

G59. Radiation Monitoring

"Radiation monitoring" is measurement and surveillance of radioactive materials concentrations, etc. by means of sampling or radiation monitor.

"Enable proper radiation monitoring" means that surveillance for the release of radioactive materials and measurement of the dose rate can be performed during normal operations and abnormal conditions and that radiation sources, release points, vicinity of the nuclear power plant, anticipated release routes of radioactive materials and other necessary places can be monitored so that prompt measures can be taken in case of an accident.

Detailed requirements as to monitoring during normal operations are separately specified in the "Guide for Measurement of Radioactive Materials Released from Light Water Nuclear Power Reactor Facilities."

Detailed requirements as to the monitoring during an accident are separately specified in the "Guide for Radiation Measurement during Accidents in Light Water Nuclear Power Reactor Facilities."